Decommissioning Standard Review Plans and Risk-Informing Decommissioning Regulation

Selected 1999 Industry/NRC Decommissioning Licensing Interactions

TR-109460

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EPRI Project Manager
C. Wood

EPRI • 3412 Hillview Avenue, Palo Alto, California 94304 • PO Box 10412, Palo Alto, California 94303 • USA
800.313.3774 • 650.855.2121 • askepri@epri.com • www.epri.com

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This report was prepared by

Duke Engineering & Services
Solomon Pond Park
400 Donald Lynch Boulevard
Marlborough, MA 01752

Principal Investigators

W. Metevia
J. Oddo
T. O'Hara
J. Sutton

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REPORT SUMMARY

This report describes the technical support EPRI provided the Nuclear Energy Institute (NEI) Decommissioning Working Group in 1999. This volume includes two initiatives that produced four draft Decommissioning Standard Review Plans (DSRPs). It also includes an evaluation entitled Spent Fuel Pool Seismic Failure Frequency in Support of Risk-Informed Decommissioning – Emergency Planning.

Background

In late 1998, the NEI formed a Decommissioning Working Group to work with the Nuclear Regulatory Commission to clarify the regulations dealing with nuclear power plant facility closure and decommissioning. EPRI provides technical support to the Working Group. The Working Group also works with the NRC and senior NRC management to ensure that the strategic objectives of the industry are met.

Objectives

To document the technical support EPRI provided the NEI Decommissioning Working Group.

Results

In early 1999, the NEI Decommissioning Working Group proposed an initiative to NRC management to assist them in appropriating resources to expedite process reviews of exemption requests relating to the decommissioning of permanently defueled facilities. The Group drafted four DSRPs in the following areas:

- Emergency Plan Exemption Request for Permanently Defueled Facilities
- Request for Exemption of Permanently Defueled Facilities from Certain Requirements of 10CFR73, Physical Protection of Plants and Materials
- Financial Protection Requirements Limits: Exemption Request for Permanently Defueled Facilities
- Facility Personnel Training Requirements for Permanently Defueled Facilities

The Working Group did not submit the DSRPs to NRC for formal review, because of changes in NRC decommissioning management and their approach to the development of decommissioning regulations. The proposed DSRPs are included in draft form in this report for the industry's information, and to assist licensees in preparing those exemption requests still required.

In the second quarter of 1999, also because of the previously mentioned changes at the NRC, a series of interactions between the NEI Decommissioning Working Group and the NRC focused on risk informing decommissioning regulation. In particular, the Working Group provided a risk-

informed justification to exclude evaluations of beyond design basis accidents for permanently defueled plants. For example, a risk-informed analysis concluded that an exemption request from the offsite emergency planning requirements should not require evaluation of “beyond design basis” accidents, such as the “zircaloy fire” scenario.

EPRI Perspective

NRC staff has proposed the completion of a spent fuel pool risk assessment aimed at developing a single, integrated risk-informed decommissioning rule for Emergency Planning, Insurance, Safeguards, Operator Training/Staffing, and Backfit. This rule will use criteria developed from the Spent Fuel Pool assessment as appropriate, rather than undertaking multiple rulemakings. NRC staff also intends to develop a rulemaking plan to consolidate decommissioning regulations for nuclear power plants into a separate part of Title 10. EPRI continues to provide technical support to the NEI Working Group, as it interacts with the NRC on the development of these activities.

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Key Words

Decommissioning

ABSTRACT

This volume is the EPRI technical report on the support provided for two initiatives developed by the NEI Decommissioning Working Group in 1999.

The first initiative produced four draft Decommissioning Standard Review Plans (DSRPs), which are listed below. The Working Group proposed that the industry develop a set of draft Decommissioning Standard Review Plans (DSRPs) for NRC review, anticipating the DSRPs would be published by EPRI/NEI and endorsed by the NRC. Selected members of the NEI Decommissioning Licensing Issues Task Force, which reports to the NEI Decommissioning Working Group, reviewed the draft DSRPs. The DSRPs were not submitted to NRC for formal review because of changes in NRC decommissioning management and the changes in NRC approaches to development of decommissioning regulations. The proposed DSRPs, though still in draft form, are included in this EPRI Technical Report for industry's information and use. Licensees may find the draft DSRPs a valuable resource when preparing exemption requests. The four DSRPs are:

1. Emergency Plan Exemption Request for Permanently Defueled Facilities
2. Request for Exemption of Permanently Defueled Facilities from Certain Requirements of 10CFR73, Physical Protection of Plants and Materials
3. Financial Protection Requirements Limits: Exemption Request for Permanently Defueled Facilities
4. Facility Personnel Training Requirements for Permanently Defueled Facilities

The second initiative produced a report entitled "Evaluation of Spent Fuel Pool Seismic Failure Frequency in Support of Risk-Informed Decommissioning – Emergency Planning." This study developed risk-informed evidence that exemption requests from offsite emergency planning requirements, for permanently defueled facilities, should not require evaluation of "Beyond Design Basis" accidents (i.e., it is not necessary to evaluate "Beyond Design Basis" accidents, in general, and the "zircaloy fire" scenario, in particular).

CONTENTS

LIST OF FIGURES

LIST OF TABLES

1
SUMMARY: SELECTED 1999 INDUSTRY/NRC DECOMMISSIONING LICENSING INTERACTIONS

1.1 Background

1.1.1 EPRI/NEI Initiative

In late 1998, the Nuclear Energy Institute (NEI) formed a Decommissioning Working Group to work with NRC to clarify the regulations dealing with facility closure and decommissioning. EPRI provides technical support to the Working Group. The Working Group provides executive oversight for strategic aspects of decommissioning. In addition to providing direction on these issues, the group works with the Nuclear Regulatory Commission, senior NRC management, and members of Congress to ensure that the industry's strategic objectives are met.

These objectives include:

- Safe, timely decommissioning,
- Site cleanup and license termination,
- Deployment of dry cask storage in time to meet industry's decommissioning needs, and
- Safe, reliable, economical low-level waste disposal for decommissioning.

The Working Group was established because current regulations are not risk-informed, nor do they effectively address many issues associated with permanent shutdown and decommissioning. The group is working to ensure that the NRC devotes resources to performing critical decommissioning reviews in a timely manner.

1.2 Chronology

1.2.1 Industry Draft DSRPs

In early 1999, the NEI Decommissioning Working Group proposed an initiative to assist the NRC in devoting appropriate resources and expeditiously processing reviews of exemption requests relating to the decommissioning of permanently defueled facilities. The Working Group proposed that the industry develop a set of draft Decommissioning Standard Review Plans (DSRPs) for submittal to NRC for review. It was intended that, after NRC review and comment resolution, the DSRPs would be published by EPRI/NEI and endorsed by the NRC.

Four DSRPs in the following areas were drafted:

1. Emergency Plan Exemption Request for Permanently Defueled Facilities

2. Request for Exemption of Permanently Defueled Facilities from Certain Requirements of 10CFR73, Physical Protection of Plants and Materials

3. Financial Protection Requirements Limits: Exemption Request for Permanently Defueled Facilities

4. Facility Personnel Training Requirements for Permanently Defueled Facilities

The four draft DSRPs were reviewed by selected members of the NEI Decommissioning Licensing Issues Task Force which reports to the NEI Decommissioning Working Group. Comments were resolved and the DSRP for "Emergency Plan Exemption Request for Permanently Defueled Facilities" was provided to NRC management for preliminary review.

Because of changes in NRC decommissioning management and changes in NRC's approaches to development of decommissioning regulations, the DSRPs were not submitted to NRC for formal review. As a result of the modified direction taken by the NRC in resolving decommissioning concerns with the regulations, the draft DSRPs were not further developed after the first quarter of 1999. Nevertheless, the proposed DSRPs are included in this EPRI Technical Report in draft form for the industry's information and use. The draft DSRPs are intended to assist licensees preparing those exemption requests still required. Further explanation is provided in the Appendices which contain the draft DSRPs, along with a description of their format, content, bases, and guidance for their current use.

1.2.2 Risk-Informed, Spent Fuel Pool - Seismic Evaluation

In the second quarter of 1999, also as a result of the changes in decommissioning management and in the approaches to development of decommissioning regulation at the NRC mentioned above, a series of interactions between the NEI Decommissioning Working Group and the NRC focussed on risk informing decommissioning regulation. In particular, the Working Group provided a risk-informed justification to exclude evaluations of beyond design basis accidents for permanently defueled plants. The plan for the performance of this evaluation was presented to NRC staff at a briefing by stakeholders on Part 50 decomissioning issues, held on March 17, 1999. The results of this evaluation were provided to NRC staff at an NRC workshop on June 16, 1999. The workshop was held to allow for stakeholder comment on a "Draft Technical Study of Spent Fuel Accidents for Decommissioning Plants," dated June 1999.

A summary of the report "Evaluation of Spent Fuel Pool Seismic Failure Frequency in Support of Risk-Informed Decommissioning – Emergency Planning" is provided in Section 2.0. The full risk-informed evaluation is provided in Section 3.0 of this EPRI Technical Report.

1.3 Current Status

Rather than undertaking multiple rulemakings, the NRC staff has proposed to complete a spent fuel pool risk assessment, aimed at developing a single, integrated risk-informed decommissioning rule for Emergency Planning, Insurance, Safeguards, Operator Training/Staffing, and Backfit that uses criteria developed from the Spent Fuel Pool assessment as appropriate. On July 12, 1999, the NRC staff indicated in SECY 99-168 their recommendation to develop a rulemaking plan to consolidate decommissioning regulations for nuclear power plants into a separate part of Title 10.

The NEI, with EPRI technical support, is continuing to interact with the NRC on the development of these NRC activities.

2
RISK-INFORMED DECOMMISSIONING LICENSING

2.1 Risk-Informed Decommissioning - Emergency Planning

As part of the initiative to risk inform decommissioning regulation, a study was performed to provide risk-informed evidence that exemption requests, for permanently defueled facilities, from offsite emergency planning requirements should not require evaluation of “Beyond Design Basis” accidents (i.e., it is not necessary to evaluate “Beyond Design Basis” accidents, in general, and the “Zircaloy Fire” scenario, in particular).

Risk-informed approaches supplement traditional engineering analyses and are evaluations from a diverse perspective. Risk-informed evaluations use the results and insights derived from probabilistic risk analyses (Evaluation of Probability and Consequence of Events) to determine relative risk and make comparisons to acceptance criteria (e.g., the Safety goals; cost/benefit from backfit analyses – including intermediate criteria such as initiator frequency).

This risk-informed evaluation is based on the results of previous NRC analyses, but includes updates of key contributors to the NRC analyses. These updates further validate the conclusions reached.

2.2 Spent Fuel Pool Evaluation

In 1989, the NRC published NUREG 1353 “Regulatory Analyses for the Resolution of Generic Issue 82, ‘Beyond Design Basis Accidents in Spent Fuel Pools’ ” which provided risk results for spent fuel pools and comparisons to the Safety Goal Policy statement and to the Backfit Criteria. Risk-informed insights from NUREG-1353 include the results that Seismic Events and the Seismic Hazard are the dominant events and key contributors, respectively, to beyond design accidents in spent fuel pools.

The EPRI technical evaluation in Section 3.0 consists of a requantification of the NUREG-1353 seismic hazard frequency [key contributor] using both the more recent Lawrence Livermore National Laboratory (LLNL) and the EPRI site specific results, which show a reduction in risk. Note also that the LLNL/EPRI results have converged since NUREG-1353 was issued.

The EPRI technical evaluation contained in Section 3.0 concluded that the risk-informed evaluation previously performed by NRC, provides a sufficient rationale for not having to evaluate beyond design basis accident, particularly a zircaloy oxidation reaction [“zirc fire”], in connection with Emergency Plan exemption requests. The information provided in this report validates and further reinforces this conclusion.

3

EVALUATION OF SPENT FUEL POOL SEISMIC FAILURE FREQUENCY IN SUPPORT OF RISK-INFORMED DECOMMISSIONING – EMERGENCY PLANNING

3.1 Executive Summary

This report presents the results of a risk-informed evaluation of a zircaloy oxidation reaction event to determine whether it qualifies as a “beyond design basis” accident. Licensees may find this information particularly useful for decommissioning emergency planning purposes.

In 1989, NUREG-1353 [1] estimated the probability of a zircaloy cladding fire that may result from a spent fuel pool drain down event, for either a PWR or BWR spent fuel pool, as 2 x 10^{-6} per reactor year. NUREG/CR-5176 [5] derived the annual probability of a seismicly induced spent fuel pool (SFP) failure, by convolving a family of seismic hazard curves with a family of fragility curves. The family of seismic hazard curves was based on preliminary results published by Lawrence Livermore National Laboratory (LLNL) in 1989.[2] LLNL based the curves on estimates of the seismic capacity of typical BWR and PWR spent fuel pools. Since the publication of NUREG-1353, EPRI published seismic hazard results for 61 nuclear power plant (NPP) sites[4] and LLNL updated its seismic hazard results.[3] The purpose of this study is to recalculate the annual probability of a zircaloy cladding fire, originally calculated in NUREG-1353, using more recent seismic hazard results.

Recalculating the annual probability of a zircaloy cladding fire using more recent seismic hazard curves reduces the SFP failure frequency across the population of plants assessed. This analysis uses the same assumptions described in NUREG-1353 and the SFP seismic failure methodology described in NUREG/CR-5176. Substituting the LLNL 1993 and EPRI 1989 seismic hazard results in the NUREG-1353 analysis yields the following results. Using the LLNL 1993 results, the annual probability of a zircaloy cladding fire is estimated to have a mean value of 5.6 x 10^{-7} per reactor year for either the PWR or the BWR spent fuel pool. Using the EPRI 1989 results, the annual probability of a zircaloy cladding fire is estimated to have a mean value of 1.8 x 10^{-7} per reactor year for either the PWR or the BWR spent fuel pool. On average, use of these updated seismic hazard curves results in a reduction in the SFP failure frequency across the population of plants by a factor of 8 when using 1993 LLNL seismic hazard results and about 70 when using 1989 EPRI seismic hazard results.

The results of this analysis satisfy the probabilistic acceptance criteria for exclusion under Standard Review Plan (SRP) 2.2.3, “Evaluation of Potential Accidents,” Revision 2, 1981. SRP

2.2.3 provides a basis for inclusion or exclusion of potential accidents into the plant design basis. In general, emergency planning is required to ensure the continued protection of the public health and safety in areas around the nuclear facility in the event of a radiological emergency. Application of the SRP 2.2.3 acceptance criteria provides a basis for elimination of the requirements for off-site emergency planning at decommissioning NPPs.

3.2 Introduction

The overall objective of this effort is to provide risk-informed evaluation whether inclusion of “beyond design basis accidents,” particularly a zircaloy oxidation reaction [fire] accident as the basis for Decommissioning Emergency Planning is warranted. This issue was satisfactorily resolved for all plants by NUREG-1353 in 1989. The conclusions remain valid today, because the decommissioning state does not adversely affect the results on which the conclusions were based. Since the publication of NUREG-1353, significant improvements have been made in the seismic hazard results on which the previous conclusions were based. In particular, recent work by both the regulator and the industry has reduced the calculated seismic hazard, which is the dominant contributor to the overall spent fuel pool release frequency.

The purpose of this document is to describe the methodology and results of the seismic technical analysis used to demonstrate the above conclusions are valid. NUREG-1353 “Regulatory Analysis for the Resolution of Generic Issue 82 Beyond Design Basis Accidents in Spent Fuel Pools,” dated April 1989 is considered a valid framework for this analysis. Given the NUREG-1353 framework, the Spent Fuel Pool failure frequencies due to seismic was updated using more current seismic hazard results.

Table 4.7.1 of NUREG-1353 summarizes the frequency of spent fuel damage resulting from accident sequences which can result in the loss of water from the Spent Fuel Pool (SFP) either through drainage or boiling as a result of loss of cooling. As described in Reference 1, the seismic event contributes over 90% of the PWR spent fuel damage probability, and nearly 95% for the BWR. However, since publication of NUREG-1353, revisions have been made to the published seismic hazard results at those sites previously evaluated for SFP failure frequency. In particular, revisions to the Lawrence Livermore National Laboratory (LLNL) seismic hazard results at 69 Eastern United States (EUS) sites was published in 1993. In addition, Electric Power Research Institute (EPRI) hazard results are also available at 61 EUS sites.

NUREG-1353 is considered a valid framework to calculate release frequencies at these sites. SFP accident frequencies for other scenarios (Missiles, Aircraft crashes, etc.) as shown in Table 3-1, which is a verbatim copy of Table 4.7.1 in NUREG-1353, are considered valid for this analysis. Only the SFP failure frequency due to seismic is updated. The SFP failure frequencies due to seismic used in the NUREG-1353 analysis are from NUREG/CR-5176. Updates of the SFP failure frequency will be based on the methodology and inputs described in NUREG/CR-5176. Therefore, this analysis is in essence a NUREG-1353 analysis with new seismic hazard curves used to calculate spent fuel pool failure frequencies.

Using the 1989 and 1993 Lawrence Livermore National Laboratory (LLNL) seismic hazard results at 69 sites east of the Rocky Mountains, and the 1989 EPRI results at 61 sites east of the Rocky Mountains, the SFP failure frequency at each site is calculated. The reduction in SFP failure frequency due to the use of the 1993 LLNL results and the 1989 EPRI results is quantified. Given the NUREG-1353 framework, and the updated SFP failure frequencies, release frequencies are calculated for each of the 69 sites. The mean annual probability of a zircaloy cladding fire, due to loss of water from the spent fuel pool, is also calculated.

Table 3-1
Summary of Accident Sequence Quantification from NUREG-1353

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

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

3.3 Methodology

3.3.1 SFP Failure Frequency Due to Seismic – NUREG/CR-5176

The methodology to calculate SFP failure frequency is described in NUREG/CR-5176. SFP failure frequency due to seismic is calculated by convolving the seismic hazard distribution with the seismic fragility of the SFP. The convolution process to numerically integrate the family of seismic hazard curves with the family of fragility curves is described in Reference 6.

3.3.2 Generation of the Family of Seismic Hazard Curves

The following assumptions and recommendations from Reference 5 were used to generate the family of seismic hazard curves for this analysis:

1. A lognormal distribution was assumed for the distribution of the uncertainty in probability of exceedance at each acceleration value. The parameters of the lognormal distribution (i.e., median and logarithmic standard deviation (β)) were calculated by using the 50^{th} and 95^{th} percentile values.

2. Given the median and the 95th percentile the logarithmic standard deviation (β) is calculated ($\beta = (\ln(x_{95}/x_{50})/1.64)$). β can also be calculated from the natural log of the ratio of the 85th percentile to the median. Given β, the probability of exceedance (X_n) can then be calculated at various percentiles ($X_n = x_{50} * e^{z\beta}$). Z is the standard normal variate.

3. Because it is possible to get probability of exceedance values greater than 1.0, the lognormal distribution is truncated at X_{99} (the 99 percentile). The lognormal distribution was normalized to get a new distribution with cutoff at X_{99}.

4. The range of hazard represented by the truncated (lognormal) distribution at each acceleration was discretized into eleven discrete values of the hazard with subjective probabilities of 0.03, 0.05, 0.07, 0.12, 0.15, 0.16, 0.15, 0.12, 0.07, 0.05, and 0.03.

Figures 3-1, 3-2, and 3.3 show the family of hazard curves generated for the Vermont Yankee site based upon use of the above process. Figure 3-1 is based upon use of the 1989 LLNL results and an estimate of β based on use of the 95^{th} percentile as described in (2) above. Figure 3-2 is based on use of the 1993 LLNL results and an estimate of β based on use of the 85^{th} percentile as described in (2) above. Figure 3-3 is based upon use of the 1989 EPRI results and an estimate of β based on use of the 85^{th} percentile as described in (2) above. In Figures 2.2 and 2.3, β is estimated using the 85^{th} percentile because 95^{th} percentile results are not available. As can be seen, there is a significant reduction in the uncertainty between Figure 3-1 and Figures 2.2 and 2.3. At about 2 g, the uncertainty in Figure 3-1 ranges from about 5×10^{-5} to about 1×10^{-12}, whereas in Figure 3-2 the uncertainty ranges from about 4×10^{-6} to 2×10^{-11}. The Figure 3-3 results (EPRI) behave similar to the Figure 3-2 results.

Figure 2.4 compares estimates of β (logarithmic standard deviation) at each of the 69 sites using the LLNL 1989 results. Figure 2.4 shows that β based upon use of the 95th percentile is equal to or lower than β based on use of the 85th percentile in all cases. Use of β based on the 85th percentile estimates for the Vermont Yankee site would result in a wider uncertainty band than that shown in Figure 2.1. In terms of overall SFP seismic failure frequency, use of β values based on the use of the 85th percentile would result in SFP failure frequencies about a factor of 2 higher than values calculated based on use of the 95th percentile. As described earlier, LLNL 1993 does not contain 95th percentile results. Therefore, LLNL 1993 SFP failure frequencies that use the 85th percentile to estimate β are, in general, about a factor of 2 higher in probability than SFP failure frequencies based on a 95th percentile estimate of β.

3.3.3 Generation of the Family of Fragility Curves

NUREG/CR-5176 evaluated spent fuel pools at Vermont Yankee (BWR) and Robinson (PWR) to develop realistic estimates of the seismic capacity of typical BWR and PWR spent fuel pools. Other SFP failure frequency analyses have used different approaches. For example, Brookhaven National Laboratory (BNL)[7] estimated seismic fragility using available Probabalisitic Risk Assessments. In particular, the Brookhaven study used the seismic fragility developed for the Oyster Creek reactor for the Millstone 1 BWR. Likewise, it used the fragility of the Zion plant auxiliary building shear walls was used for the Ginna spent fuel pool. NUREG/CR-5176, on the other hand, estimated seismic fragility by thorough reviews of structural drawings, the Final Safety Analysis Report (FSAR) and spent fuel pool reports (References 8 and 9). Based on this review, BWR SFP fragility is defined by:

The median fragility (x_{50}) = 1.4g

The random uncertainty β_R = 0.26

The uncertainty in location β_U = 0.39

For PWRs, the SFP fragility is defined by:

The median fragility (x_{50}) = 2.0g

The random uncertainty β_R = 0.28

The uncertainty in location β_U = 0.40

As described in Reference 6, the uncertainty in the median is described by the following equation:

$$\tilde{a}_i = A_m e^{\beta_u \zeta_i} \qquad (1)$$

where:

$\tilde{a}_i$	=	uncertainty in the median,
A_m	=	the "median median" fragility (1.4 for BWR, 2.0 for PWR),
β_U	=	the lognormal standard deviation of the ã distribution, and
ζ_i	=	the standard normal variate.

Five curves, described by ζ_i values ranging from –1.28, -0.58, 0.0, 0.58, and 1.28, are used to define the uncertainty in median SFP fragilities. The basis for five fragility curves is described in Reference 10.

The desired discrete value for a fragility curve is then:

$$a = \tilde{a}_i \, e^{\beta_r Z} \quad (2)$$

where:

a	=	acceleration value at a given failure frequency defined by Z,
$\tilde{a}_i$	=	a median fragility,
β_r	=	the lognormal standard deviation of the random uncertainty about the median, and
Z	=	the standard normal variate.

Equations 1 and 2 can be combined such that the failure frequency at given accelerations (a), usually those which describe the hazard curve, can be readily calculated. The final equation is:

$$F_i(a) = N(\ln(a/A_m \, e^{\beta_u \zeta_i})/\, \beta_r) \quad (3)$$

$$F_i(a) = N(Z) \quad (4)$$

where,

$F_i(a)$	=	the fraction of earthquakes to fail the SFP at acceleration a, and
N(Z)	=	the area under the normal curve up to point Z.

3.3.4 Calculation of the Release Frequency Given SFP Failure

The methodology to calculate the release frequency given SFP failure is described in NUREG-1353. Table 3-1 of this report is a duplicate of Table 4.7.1 of NUREG-1353. As can be seen in Table 3-1, the annual probability of a SFP failure for a PWR is described by the sum (1.5×10^{-7}) of the SFP failure frequencies associated with Structural Failures, Pneumatic Seal Failures, Inadvertent Drainage, and Loss of Cooling Make-up **plus** the Seismic Structural Failure. The annual probability of a release is the product of the annual SFP failure frequency and the conditional probability of zircaloy cladding fire given loss of water. For PWRs, the conditional probability of the zircaloy cladding fire is considered to be 1.0. Values less than 1.0 for a PWR are supported by Table 4.5.1 in NUREG-1353. For a BWR, the process is exactly the same with the exception that the conditional probability of zircaloy cladding fire given loss of water is 0.25. Values less than 0.25 for a BWR are supported by Table 4.5.1 in NUREG-1353.

Using this approach, SFP seismic failure frequencies were calculated at each of the 69 sites using the LLNL results and at 61 sites using the EPRI results.

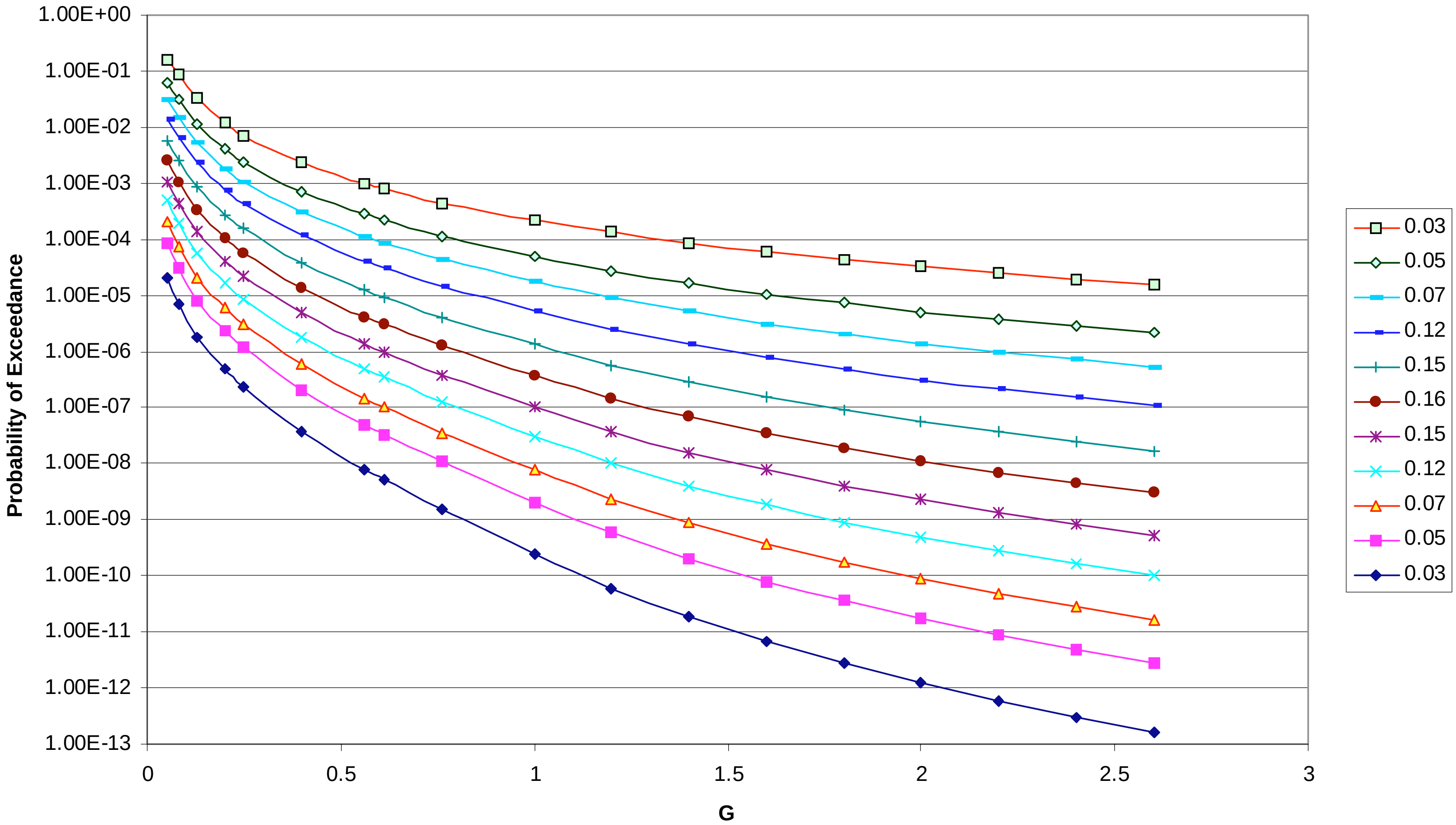

Figure 3-1
Family of Hazard Curves for Vermont Yankee Based on Use of LLNL - 1989

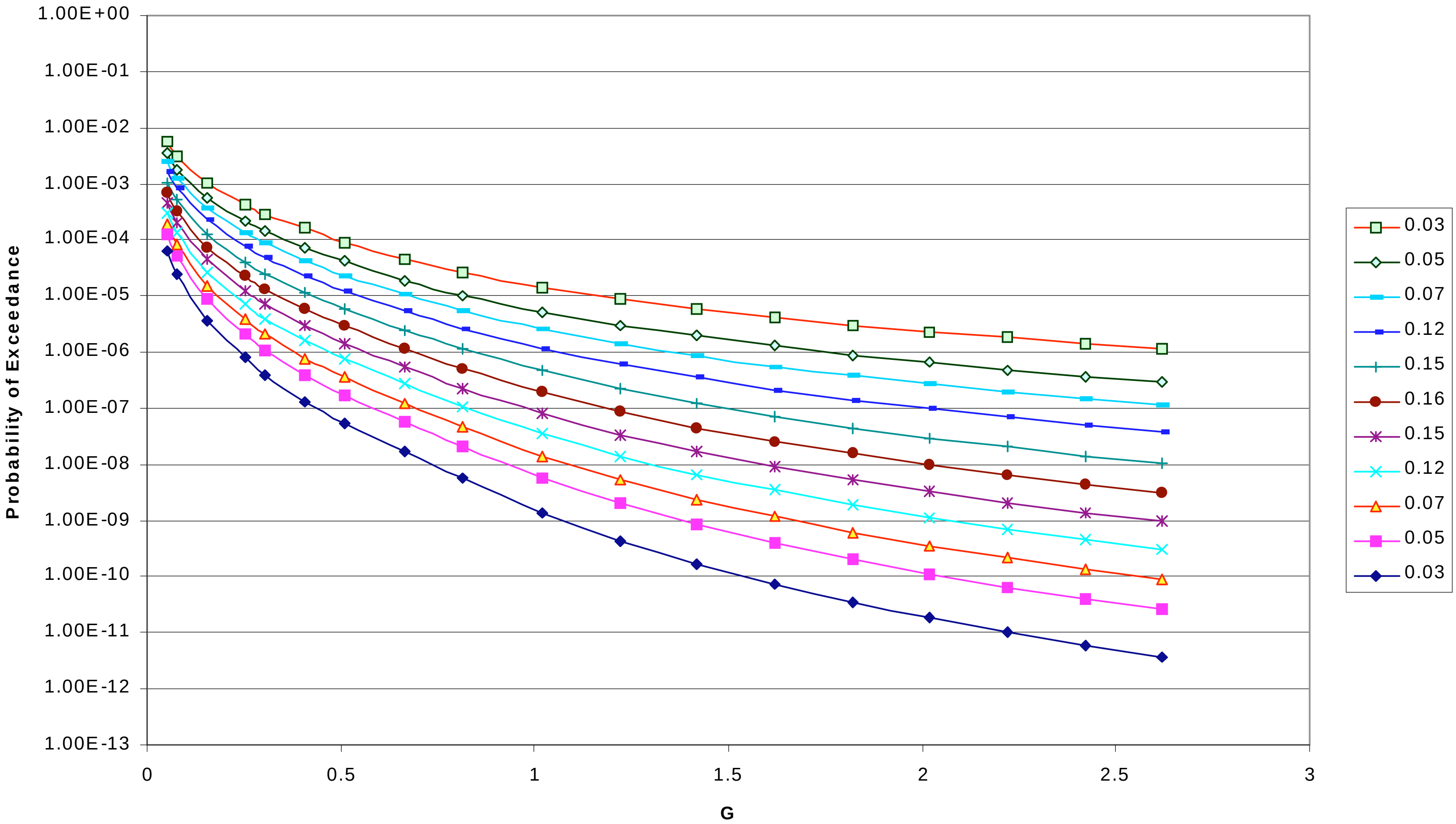

Figure 3-2
Family of Hazard Curves for Vermont Yankee Based on Use of LLNL - 1993

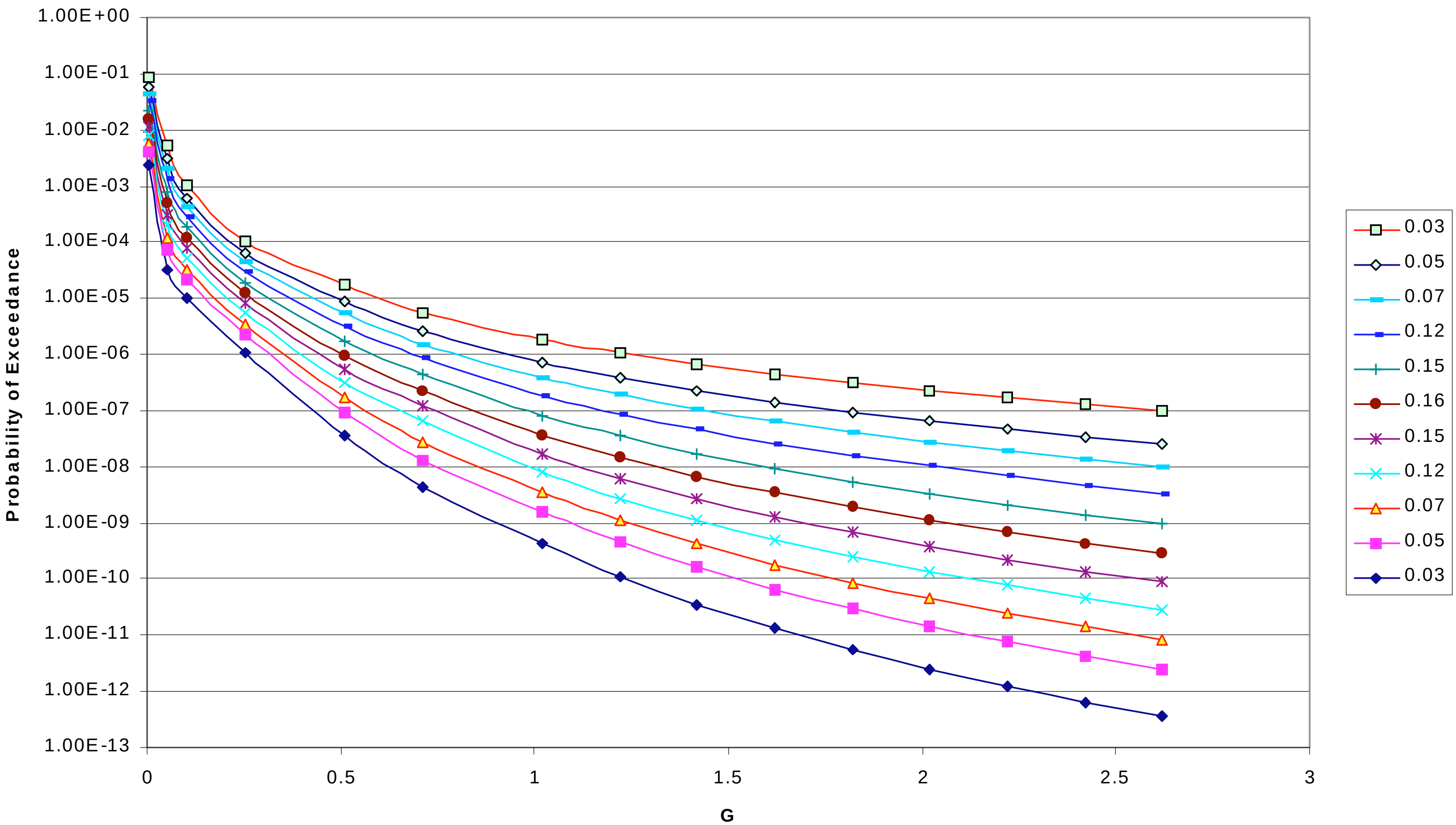

Figure 3-3
Family of Hazard Curves for Vermont Yankee Based on Use of EPRI - 1989

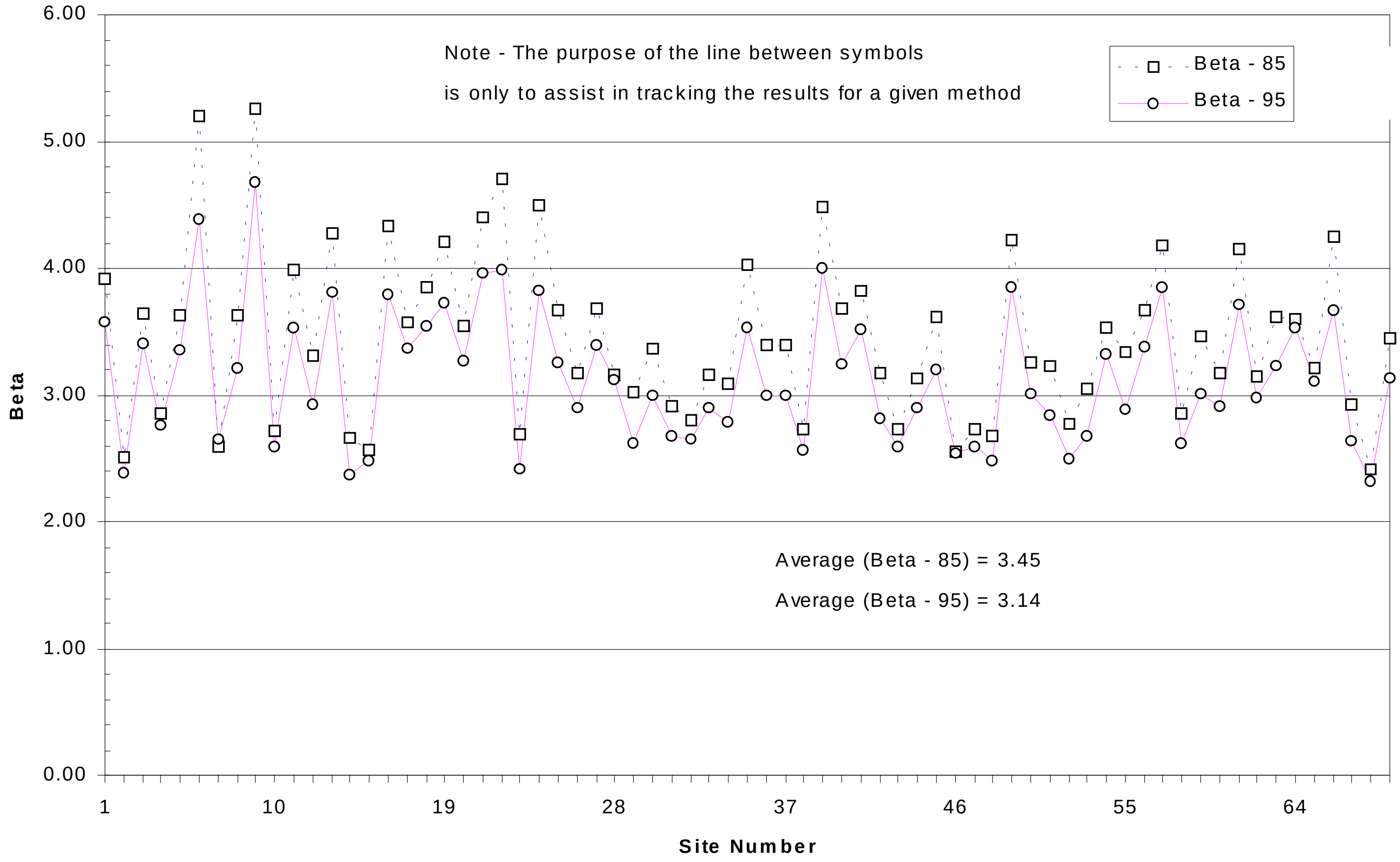

Figure 3-4
Comparison of Alternative Methods to Calculate Beta

3.4 Results

Based on the methodology described in Sections 3.3.2 and 3.3.3, the annual probability of a SFP failure due to seismic was calculated. The SFP failure frequency was calculated based on reproduction of the LLNL 1989 results, and use of the LLNL 1993 results, and the EPRI 1989 results. The annual probability of a release based upon the alternative SFP failure frequencies was also calculated.

Table 3-1 provides the overall results of the analysis. The site numbers in Table 3-1 are ordered the same as the Nuclear Power Plants (NPPs) listed in Table A.1 of NUREG-1353, except that no western US NPPs were included in this analysis. Column 2 contains the code specifying the plant at the site to be either a BWR or a PWR. Column 3 presents the SFP failure frequency based on the LLNL 1989 results and estimates of β based on the natural log of the ratio between the 85th percentile and the 50th percentile. Column 4 presents the SFP failure frequency based on the LLNL 1989 results and estimates of β based on the natural log of the ratio between the 95th percentile and the 50th percentile divided by 1.64. The Column 4 SFP failure frequencies are, on average, about a factor of 2 lower than the Column 3 SFP failure frequencies. Column 5 presents the SFP failure frequency based on the LLNL 1993 results and estimates of β based on the natural log of the ratio between the 85th percentile and the 50th percentile. Column 6 presents the SFP failure frequency based on use of the EPRI 1989 results and estimates of β based on the natural log of the ratio between the 85th percentile and the 50th percentile. Column 7 quantifies the reduction in SFP failure frequency based upon use of the 1993 LLNL results. Column 7 is the ratio of the LLNL 1989 (Column 4) and LLNL 1993 (column 5) SFP failure frequencies. Column 8 quantifies the reduction in SFP failure frequency based upon use of the 1989 EPRI results. Column 8 is the ratio of the LLNL 1989 (Column 4) and EPRI 1993 (Column 5) SFP failure frequencies. Columns 9, 10, and 11 are the overall SFP release frequencies at each site.

As shown in Table 3-1, the LLNL 1993 seismic hazard data reduced the SFP failure frequency by a factor of 8, on average, relative to NUREG-1353 results. Similarly, the EPRI 1989 data reduced the SFP failure frequency by a factor of 70. At some sites, using the LLNL 1993 results caused the SFP failure frequency to increase slightly because the LLNL 1993 results used the 85th percentile to estimate the logarithmic standard deviation, β.

Figure 3-2 is a plot of the annual probability of a release at the population of EUS sites based on the LLNL 1993 results and the EPRI 1989 results. Figure 3-1 shows that the release probabilities for all NPPs are on the order of 10^{-6} or less, based on the LLNL 1993 results, with an overall mean annual probability of 5.6×10^{-7}. Two plants are slightly above the figure of merit (2.0×10^{-6}) presented in NUREG-1353. All NPPs are less than 10^{-6} based on the EPRI 1989 results (overall mean = 1.8×10^{-7}). In general there is excellent agreement between the LLNL and EPRI release frequency results with the exception of those LLNL NPP results that exceed 10^{-6}. All NPPs that exceed 10^{-6} based on LLNL seismic hazard results are soil sites.

Table 3-2
Spent Fuel Pool Analysis

1	2	3	4	5	6	7	8	9	10	11
Site	BWR=1	LLNL - 89	LLNL - 89	LLNL - 93	EPRI - 89	Ratio	Ratio	Release	Release	Release
	PWR=2	Beta=x85/x50	Beta=x95/x50	Beta=x85/x50	Beta=x85/x50	L89/L93	L89/E89	Frequency	Frequency	Frequency
								LLNL-89	LLNL-93	EPRI - 89
1	2	1.10E-05	5.80E-06	3.60E-07	3.70E-08	16.1	156.8	5.95E-06	5.10E-07	1.87E-07
2	2	1.40E-06	1.10E-06	1.10E-06	1.20E-08	1.0	91.7	1.25E-06	1.25E-06	1.62E-07
3	2	1.00E-05	7.20E-06	3.70E-07	8.60E-08	19.5	83.7	7.35E-06	5.20E-07	2.36E-07
4	1	1.60E-06	1.40E-06	9.60E-07	4.40E-09	1.5	318.2	3.88E-07	2.78E-07	3.86E-08
5	2	2.00E-06	1.10E-06	1.20E-07	2.50E-08	9.2	44.0	1.25E-06	2.70E-07	1.75E-07
6	1	9.10E-05	2.00E-05	5.80E-07	8.40E-08	34.5	238.1	5.04E-06	1.83E-07	5.85E-08
7	1	7.30E-06	8.00E-06	3.20E-06	4.80E-07	2.5	16.7	2.04E-06	8.38E-07	1.58E-07
8	2	1.90E-06	8.10E-07	1.80E-07	1.90E-08	4.5	42.6	9.60E-07	3.30E-07	1.69E-07
9	2	1.80E-05	6.50E-06	7.10E-08		91.5		6.65E-06	2.21E-07	
10	2	4.10E-07	3.30E-07	3.90E-07	1.90E-08	0.8	17.4	4.80E-07	5.40E-07	1.69E-07
11	2	5.60E-06	2.30E-06	2.30E-07	1.40E-07	10.0	16.4	2.45E-06	3.80E-07	2.90E-07
12	1	1.30E-05	6.80E-06	2.20E-06	5.80E-08	3.1	117.2	1.74E-06	5.88E-07	5.20E-08
13	2	9.90E-07	3.80E-07	2.80E-08	2.80E-09	13.6	135.7	5.30E-07	1.78E-07	1.53E-07
14	2	1.10E-06	6.50E-07	5.80E-07		1.1		8.00E-07	7.30E-07	
15	1	6.00E-06	5.40E-06	4.90E-06		1.1		1.39E-06	1.26E-06	
16	2	2.20E-06	6.60E-07	8.70E-08	2.80E-09	7.6	235.7	8.10E-07	2.37E-07	1.53E-07
17	2	2.10E-06	1.40E-06	3.50E-07	3.20E-08	4.0	43.8	1.55E-06	5.00E-07	1.82E-07
18	1	7.40E-06	4.30E-06	5.20E-07	8.10E-08	8.3	53.1	1.11E-06	1.68E-07	5.78E-08
19	1	4.10E-06	1.80E-06	1.10E-07		16.4		4.88E-07	6.50E-08	

1	2	3	4	5	6	7	8	9	10	11
Site	BWR=1	LLNL - 89	LLNL - 89	LLNL - 93	EPRI - 89	Ratio	Ratio	Release	Release	Release
	PWR=2	Beta=x85/x50	Beta=x95/x50	Beta=x85/x50	Beta=x85/x50	L89/L93	L89/E89	Frequency	Frequency	Frequency
20	1	3.90E-06	2.50E-06	3.80E-07	1.40E-07	6.6	17.9	6.63E-07	1.33E-07	7.25E-08
21	2	4.40E-06	1.70E-06	9.70E-08	3.70E-09	17.5	459.5	1.85E-06	2.47E-07	1.54E-07
22	1	1.90E-05	5.50E-06	2.20E-07	4.50E-08	25.0	122.2	1.41E-06	9.25E-08	4.88E-08
23	2	1.80E-06	1.10E-06	1.30E-06		0.8		1.25E-06	1.45E-06	
24	2	7.50E-06	1.80E-06	1.30E-07	3.60E-08	13.8	50.0	1.95E-06	2.80E-07	1.86E-07
25	1	1.90E-06	9.90E-07	3.50E-07	6.20E-09	2.8	159.7	2.85E-07	1.25E-07	3.91E-08
26	2	3.10E-06	2.00E-06	3.80E-07	1.90E-07	5.3	10.5	2.15E-06	5.30E-07	3.40E-07
27	2	2.20E-06	1.30E-06	1.20E-07	3.10E-08	10.8	41.9	1.45E-06	2.70E-07	1.81E-07
28	1	2.30E-06	2.20E-06	1.10E-06	1.20E-07	2.0	18.3	5.88E-07	3.13E-07	6.75E-08
29	1	2.80E-06	1.50E-06	1.80E-06	2.50E-07	0.8	6.0	4.13E-07	4.88E-07	1.00E-07
30	2	6.10E-06	2.90E-06	4.90E-07	3.00E-07	5.9	9.7	3.05E-06	6.40E-07	4.50E-07
31	2	6.10E-07	3.90E-07	4.70E-07	9.60E-08	0.8	4.1	5.40E-07	6.20E-07	2.46E-07
32	1			1.30E-06					3.63E-07	
33	1	5.70E-06	4.50E-06	2.50E-06	2.10E-07	1.8	21.4	1.16E-06	6.63E-07	9.00E-08
34	1	9.80E-06	6.40E-06	1.20E-06	4.10E-07	5.3	15.6	1.64E-06	3.38E-07	1.40E-07
35	2	5.50E-06	3.50E-06	4.90E-07	1.10E-07	7.1	31.8	3.65E-06	6.40E-07	2.60E-07
36	2	7.10E-06	2.30E-06	1.50E-07	5.60E-08	15.3	41.1	2.45E-06	3.00E-07	2.06E-07
37	1	1.10E-05	6.20E-06	1.00E-06	5.40E-07	6.2	11.5	1.59E-06	2.88E-07	1.73E-07
38	2	3.90E-06	1.90E-06	3.40E-07	1.60E-07	5.6	11.9	2.05E-06	4.90E-07	3.10E-07
39	1	2.40E-06	1.90E-06	1.40E-06	1.90E-07	1.4	10.0	5.13E-07	3.88E-07	8.50E-08
40	1	1.40E-05	5.60E-06	2.30E-07	4.50E-08	24.3	124.4	1.44E-06	9.50E-08	4.88E-08
41	2	4.50E-06	1.80E-06	2.50E-07	2.00E-07	7.2	9.0	1.95E-06	4.00E-07	3.50E-07

1	2	3	4	5	6	7	8	9	10	11
Site	BWR=1	LLNL - 89	LLNL - 89	LLNL - 93	EPRI - 89	Ratio	Ratio	Release	Release	Release
	PWR=2	Beta=x85/x50	Beta=x95/x50	Beta=x85/x50	Beta=x85/x50	L89/L93	L89/E89	Frequency	Frequency	Frequency
42	2	8.10E-06	4.30E-06	3.00E-07	1.90E-07	14.3	22.6	4.45E-06	4.50E-07	3.40E-07
43	1	2.60E-06	1.50E-06	2.00E-06	2.50E-07	0.8	6.0	4.13E-07	5.38E-07	1.00E-07
44	2	9.20E-07	7.10E-07	4.60E-07	4.50E-08	1.5	15.8	8.60E-07	6.10E-07	1.95E-07
45	1	1.10E-05	7.50E-06	1.20E-06	4.50E-07	6.3	16.7	1.91E-06	3.38E-07	1.50E-07
46	1	4.50E-06	2.40E-06	3.70E-07	1.00E-07	6.5	24.0	6.38E-07	1.30E-07	6.25E-08
47	1	2.20E-05	2.20E-05	1.20E-05	1.10E-06	1.8	20.0	5.54E-06	3.04E-06	3.13E-07
48	2	7.00E-07	5.20E-07	4.10E-07	7.00E-08	1.3	7.4	6.70E-07	5.60E-07	2.20E-07
49	2	7.50E-07	5.40E-07	5.80E-07	2.10E-08	0.9	25.7	6.90E-07	7.30E-07	1.71E-07
50	1	5.50E-06	2.90E-06	2.70E-07	1.00E-07	10.7	29.0	7.63E-07	1.05E-07	6.25E-08
51	2	2.20E-06	1.30E-06	1.60E-06	6.60E-08	0.8	19.7	1.45E-06	1.75E-06	2.16E-07
52	1	1.10E-06	6.60E-07	3.10E-07	6.40E-09	2.1	103.1	2.03E-07	1.15E-07	3.91E-08
53	2	6.70E-07	4.20E-07	6.60E-07	7.60E-08	0.6	5.5	5.70E-07	8.10E-07	2.26E-07
54	2	8.40E-06	4.80E-06	8.40E-07	3.00E-07	5.7	16.0	4.95E-06	9.90E-07	4.50E-07
55	2	1.00E-05	6.90E-06	4.10E-07	3.50E-07	16.8	19.7	7.05E-06	5.60E-07	5.00E-07
56	2	2.10E-07	9.90E-08	2.20E-07	9.60E-10	0.5	103.1	2.49E-07	3.70E-07	1.51E-07
57	2	4.50E-07	2.30E-07	2.70E-07		0.9		3.80E-07	4.20E-07	
58	2	1.00E-05	4.90E-06	3.40E-07	1.00E-07	14.4	49.0	5.05E-06	4.90E-07	2.50E-07
59	2	4.20E-07	2.80E-07	4.60E-07	4.40E-08	0.6	6.4	4.30E-07	6.10E-07	1.94E-07
60	1	9.60E-06	4.60E-06	7.90E-07	9.90E-08	5.8	46.5	1.19E-06	2.35E-07	6.23E-08
61	2	3.90E-06	2.20E-06	3.40E-07	1.10E-07	6.5	20.0	2.35E-06	4.90E-07	2.60E-07
62	2	1.40E-06	5.00E-07	9.80E-08		5.1		6.50E-07	2.48E-07	
63	2	2.20E-06	1.50E-06	1.40E-06	6.00E-08	1.1	25.0	1.65E-06	1.55E-06	2.10E-07

1	2	3	4	5	6	7	8	9	10	11
Site	BWR=1	LLNL - 89	LLNL - 89	LLNL - 93	EPRI - 89	Ratio	Ratio	Release	Release	Release
	PWR=2	Beta=x85/x50	Beta=x95/x50	Beta=x85/x50	Beta=x85/x50	L89/L93	L89/E89	Frequency	Frequency	Frequency
64	1	1.50E-05	7.70E-06	8.40E-07	1.40E-07	9.2	55.0	1.96E-06	2.48E-07	7.25E-08
65	2	6.30E-07	5.10E-07	3.00E-07	5.30E-10	1.7	962.3	6.60E-07	4.50E-07	1.51E-07
66	2	6.60E-06	5.70E-06	4.00E-07	2.00E-07	14.3	28.5	5.85E-06	5.50E-07	3.50E-07
67	2	2.10E-06	6.60E-07	5.30E-08	2.00E-08	12.5	33.0	8.10E-07	2.03E-07	1.70E-07
68	2	4.80E-06	2.90E-06	2.50E-06	6.60E-08	1.2	43.9	3.05E-06	2.65E-06	2.16E-07
69	2	9.10E-07	7.50E-07	1.10E-06	5.00E-08	0.7	15.0	9.00E-07	1.25E-06	2.00E-07
	Average	6.52E-06	3.28E-06	9.07E-07	1.35E-07	8.2	73.9	1.91E-06	5.61E-07	1.82E-07

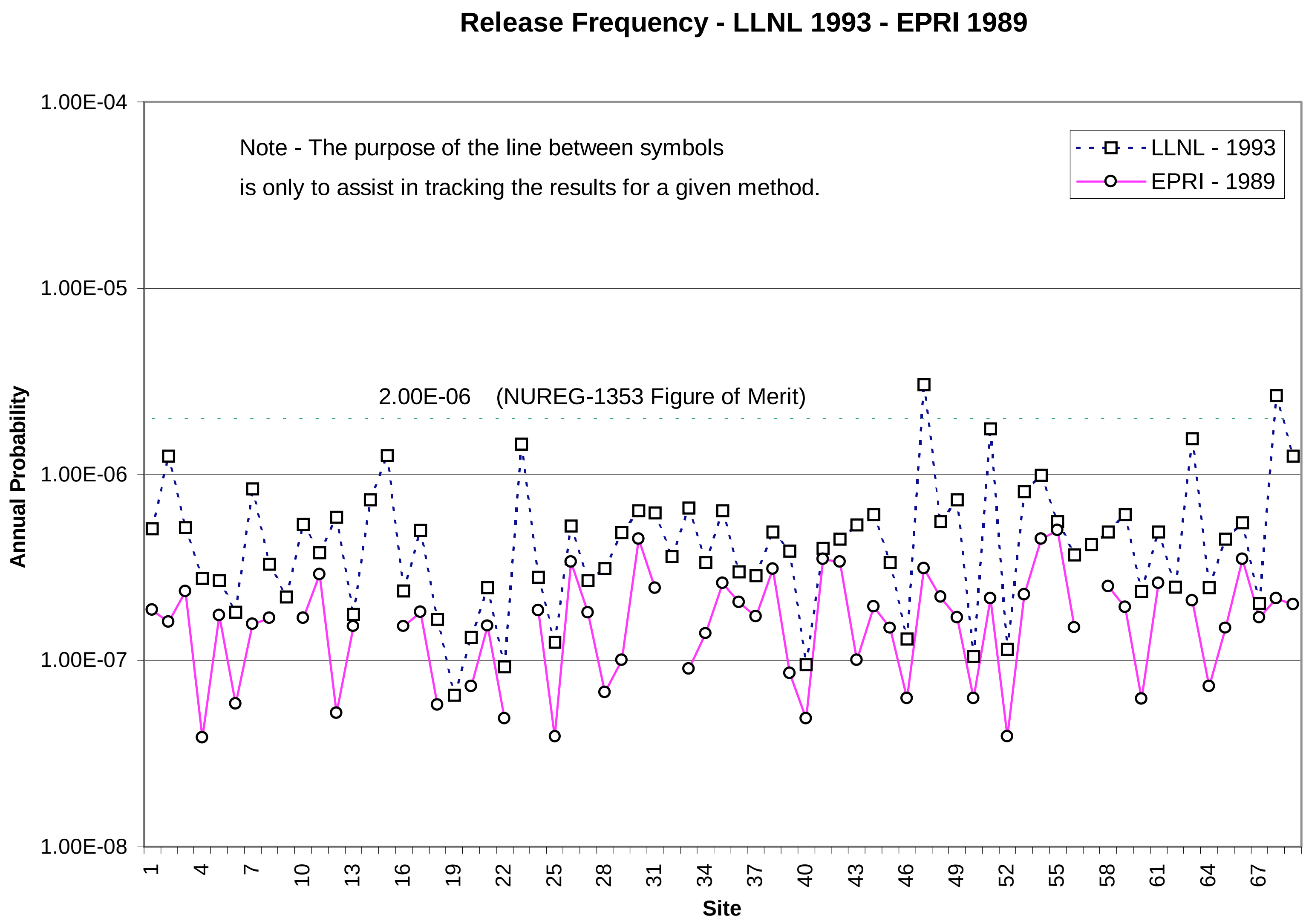

Figure 3-5
Comparison of SFP Release Frequencies with the NUREG-1353 Figure of Merit

3.5 Conclusions

The following key results are derived from NUREG-1353:

- The annual probability of a zircaloy cladding fire, resulting from the loss of water from the spent fuel pool, is estimated to have a mean value of 2 x 10^{-6} per reactor year for either the PWR or the BWR spent fuel pool.
- The seismic event is the dominant contributor to the annual probability of a zircaloy cladding fire resulting from the loss of water from the spent fuel pool.
- The risk due to beyond design basis accidents in spent fuel pools, while not negligible, are sufficiently low such that no further risk reductions were warranted.

The SFP failure frequency due to seismic and used in NUREG-1353 is documented in NUREG/CR-5176. Since publication of NUREG-1353 and NUREG/CR-5176 the LLNL seismic hazard results have been updated. Industry also published seismic hazard results at 61 NPP sites.

Using the methodology to calculate SFP failure frequency due to seismic described in NUREG/CR-5176, along with the NUREG-1353 assumptions, the NUREG-1353 SFP release values have been updated based upon use of the LLNL 1993 and EPRI 1989 seismic hazard results. The average reduction in SFP failure frequency across the population of EUS sites was about a factor of 8 when the LLNL 1993 results were used and over a factor of 70 when the EPRI 1989 results were used relative to the SFP failure frequency using the 1989 LLNL results. Using the LLNL 1993 results, the annual probability of a zircaloy cladding fire, resulting from the loss of water from the spent fuel pool, is estimated to have a mean value of 5.6 x 10^{-7} per reactor year for either the PWR or the BWR spent fuel pool. Using the EPRI 1989 results, the annual probability of a zircaloy cladding fire, resulting from the loss of water from the spent fuel pool, is estimated to have a mean value of 1.8 x 10^{-7} per reactor year for either the PWR or the BWR spent fuel pool. These results indicate that the mean risk due to beyond design basis accidents in spent fuel pools across the population of EUS sites is essentially negligible.

In addition, NUREG-1353 states that the high confidence low probability of failure (HCLPF) value for SFPs is estimated to be in the 0.5 to 0.65 g range, about three times the safe shutdown earthquake (SSE) peak ground acceleration values for typical EUS NPPs. The SFP median capacity is estimated to be in the 1.4 to 2.0 g range. 10 CFR Part 100, Appendix III.(c) defines an SSE as:

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

The results of this analysis also meet the probabilistic criteria of SRP 2.2.3, “Evaluation of Potential Accidents.” This SRP provides a basis for inclusion or exclusion of potential accidents into the plant design basis. For operating NPPs, emergency planning is required to ensure the continued protection of the public health and safety in areas around the nuclear facility in the event of a radiological emergency. Application of the SRP 2.2.3 criteria provides a basis for elimination of the requirements for off-site emergency planning at decommissioning NPPs, as explained below.

The probabilistic acceptance criteria for exclusion of accidents, in SRP 2.2.3, is as follows: “Accordingly, the expected rate of occurrence of potential exposures in excess of 10 CFR Part 100 guidelines of approximately 10^{-6} per year is acceptable if, when combined with reasonable qualitative arguments, the realistic probability can be shown to be lower.” Figure 3-1 illustrates that the LLNL 1993 mean results are on the order of 10^{-6}. The results of this analysis are conservative for the following reasons:

1. Loss of cooling/makeup is less probable for decommissioning plants because of fewer potential challenges to the fuel pool cooling/makeup system, as well as increased simplicity/reliability of the system.

2. Complete loss of SFP water is assumed given the seismic failure in NUREG-1353, although only a partial loss may actually result.

3. The conditional probability of a zircaloy cladding fire, given loss of water, for operating PWRs and BWRs has been assumed to be guaranteed and 0.25 respectively (bounding values). Decommissioning PWRs and BWRs, experience spent fuel decay which immediately and continuously reduces this probability.

4. Following permanent shutdown and defueling of the reactor, the decay heat declines exponentially, rapidly reducing and finally eliminating any potential for a zircaloy fire.

The results of this analysis satisfy the probabilistic acceptance criteria for exclusion of potential accidents that could result in radiological release in excess of the 10 CFR Part 100 guidelines, thereby obviating the need for off-site emergency planning at decommissioning plants.

4 CONCLUSION

Four Decommissioning Standard Review Plans (DSRPs) in the following areas were drafted:

1. Emergency Plan Exemption Request for Permanently Defueled Facilities

2. Request for Exemption of Permanently Defueled Facilities from Certain Requirements of 10CFR73, Physical Protection of Plants and Materials

3. Financial Protection Requirements Limits: Exemption Request for Permanently Defueled Facilities

4. Facility Personnel Training Requirements for Permanently Defueled Facilities

Because of changes in NRC decommissioning management and changes in NRC's approaches to development of decommissioning regulations, the DSRPs were not submitted to NRC for formal review. As a result of the modified direction taken by the NRC in resolving decommissioning concerns with the regulations, the draft DSRPs were not further developed after the first quarter of 1999. Nevertheless, the draft DSRPs are included in this EPRI Technical Report in draft form for the industry's information and use. The draft DSRPs are intended to assist licensees preparing those exemption requests still required.

In the second quarter of 1999, also as a result of the changes in decommissioning management and in the approaches to development of decommissioning regulation at the NRC mentioned above, a series of interactions between the NEI Decommissioning Working Group and the NRC focused on risk informing decommissioning regulation. In particular, the Working Group provided a risk-informed justification to exclude evaluations of beyond design basis accidents for permanently defueled plants. A white paper entitled "Evaluation of Spent Fuel Pool Seismic Failure Frequency in Support of Risk-Informed Decommissioning – Emergency Planning" was developed. In particular, the conclusion of this evaluation supports the position of the draft DSRP "Emergency Plan Exemption Requests for Permanently Defueled Facilities," concerning elimination of the requirements for off-site response due to beyond design basis accidents.

5
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A INDUSTRY DRAFT STANDARD REVIEW PLANS

A.1 Overall Description of Content/Purpose

A.1.1 Background and Purpose

The NRC revised its decommissioning rule, 10 CFR 50.82, in 1996 to make significant changes to the regulatory process for decommissioning plants. However, licensees must still obtain a series of exemptions to modify certain key programs not addressed in the rulemaking. The regulatory review/approval time for these exemptions has been inordinately long. In some cases reviews and approvals of critical exemption requests have taken more than a year. The extended time taken by the NRC reviewers has been attributed to two factors: low priority for support from reviewers outside the decommissioning branch and the extension of the scope of the review to areas beyond those that could have a significant impact on public health or safety. As a result, hundred of thousands of dollars have been spent each month in maintaining these programs using funds that should be spent on decommissioning activities.

The objective of the Decommissioning Standard Review Plans (DSRPs) was to expedite the NRC staff review process. Draft DSRPs were developed for submittal to NRC. The intent was to obtain endorsement by NRC in a Regulatory Guide. Individual NRC Project Managers would have used the DSRPs to approve the exemption requests in a more timely manner. The draft DSRPs would also have been issued to the industry as interim guidance during the NRC management approval period.

A.2 General Format and Content of Decommissioning Standard Review Plans

The following guidelines were used for the general format and content of the Decommissioning Standard Review Plans:

DSRPs Format

Review Responsibilities		Project Manager
I	Areas of Review	What review involves rather than who performs review
II	Acceptance Criteria	Requirements, criteria
III	Review Procedures	Review process for scope and acceptability
IV	Evaluation Findings	Desired evaluation conclusions
V	Implementation	Guidance for use of the DSRP
VI	References	

The Decommissioning Standard Review Plans follow the format of the Standard Review Plans in NUREG-0800.

In general, where there are existing SRPs for operating plants dealing with the same subject matter, e.g., emergency planning, the scope of the related DSRP is no more extensive and generally less than that for the operating plant SRP, commensurate with the reduced scope of activities and safety concerns for the decommissioning plant.

The DSRPs focus on the "Acceptance Criteria" and "Evaluation Findings" sections of the general DSRP format. The "Acceptance Criteria" are first keyed to specific regulatory requirements. Where appropriate, the specific regulatory requirements are amplified/clarified by precedents established by previous successful submittals of decommissioning plants, such as, Maine Yankee, Yankee Nuclear Power Station, and Trojan. Applications which were in preparation at the time of the development of the DSRPs, e.g., Oyster Creek and Zion were also evaluated to identify any potentially beneficial enhancements for the DSRPs. In addition, any potential rulemaking that would affect a DSRP was evaluated for possible inclusion in the DSRP's "Acceptance Criteria."

A.3 Guidance for Current Use

Although positions advocated in the DSRPs are still under discussion and review by the NRC, the DSRPs are being published for industry information and use. The information contained in the DSRPs is intended to be, at a minimum, a tutorial for the format and content of those licensee exemption submittals which are still required. The draft DSRPs provide basis information and are intended to be useful as a distillation of the issues in the topical areas. The DSRPs development was discontinued in March 1999. Given the open issues associated with these DSRPs, it is therefore recommended that the DSRPs be used as a general guideline for the development of submittals, which would be based on the Commission's rules and regulations in effect at that time.

B
DECOMMISSIONING STANDARD REVIEW PLAN: EMERGENCY PLAN EXEMPTION REQUEST FOR PERMANENTLY DEFUELED FACILITIES

Review Responsibilities

Primary - Applicant's Project Manager

Secondary - None

I. AREAS OF REVIEW

A facility which is in a permanently shutdown and defueled condition poses a significantly reduced risk to the public health and safety. After a "Certification of Permanent Cessation of Operation and Removal of Fuel from the Reactor Vessel" has been docketed in accordance with 10 CFR 50.82 and, in view of the reduced risk, it is anticipated that the licensee of such a facility will seek exemption from certain requirements of 10 CFR 50.54(q), 10 CFR 50.54(t), 10 CFR 50.47(b) and (c), and Appendix E to 10 CFR Part 50 which are no longer appropriate. This review is expected to provide the basis for exempting the applicant/licensee from the requirement of 10 CFR 50.54(q) that the licensee "shall follow and maintain in effect emergency plans which meet the standards in 10 CFR 50.47(b) and the requirements in appendix E of this part." [1] The specific elements of 10 CFR 50.47 and Appendix E to 10 CFR Part 50 which are expected to be subject to exemption and those elements which are expected to remain in effect are detailed in Attachment I of this SRP. A licensee has the option of either adopting the content of Attachment I by reference in an exemption request or proposing an alternative plant-specific exemption.

The review is performed by the licensee's Project Manager. The review involves evaluation of the licensee's Defueled Safety Analysis Report (DSAR), and any other relevant information provided in the exemption request against the acceptance criteria of Section II of this SRP. In the defueled condition, there are no longer any credible design basis accidents associated with an operating plant from startup through full power operation. The design basis accidents relative to a defueled facility are a small subset of those considered for an operating facility and are limited to the following:

1. a fuel handling incident,
2. a spent fuel cask drop, and
3. accidents associated with radioactive waste storage or processing.

The only possible addition to this list would be any design basis accident which was included in the licensee's original licensing basis for the operating facility and which is still valid for the facility in the defueled condition. As discussed in detail in Section III of this SRP, REVIEW PROCEDURES, there are no relevant beyond design basis accidents which could require offsite response capabilities and which would therefore need to be evaluated in order to grant the exemption request. This position is supported by the analyses performed in Section 3.0 of this report.

The subject review will be performed consistent with the requirements of the Commission's backfitting rule, 10 CFR 50.109. The policy underlying the rule, i.e., to ensure that new requirements are properly justified from a safety and cost-benefit standpoint, continues to apply during the decommissioning process so long as a facility's 10 CFR Part 50 license remains in effect. Decommissioning facilities are entitled to the same predictability, stability and protection from arbitrary actions as operating facilities.

II. ACCEPTANCE CRITERIA

Pursuant to 10 CFR 50.12(a), the Commission may, upon application by any interested person or upon its own initiative, grant exemptions from the requirements of the regulations which are:

1. authorized by law, will not present an undue risk to public health and safety, and are consistent with the common defense and security, and

2. present special circumstances.

Special circumstances exist when:

1. application of the regulation in the particular circumstance would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule (10 CFR 50.12(a)(2)(ii)); or

2. compliance would result in undue hardship or other costs that are significantly in excess of those contemplated when the regulation was adopted, or that are significantly in excess of those incurred by others similarly situated (10 CFR 50.12(a)(2)(iii)); or

3. there is present other material circumstances not considered when the regulation was adopted for which it would be in the public interest to grant an exemption (10 CFR 50.12(a)(2)(vi)).

The underlying purpose of 10 CFR 50.54(q) is to ensure licensees follow and maintain in effect emergency plans that provide reasonable assurance that adequate protective measures can and will be taken in the event of an emergency at a nuclear reactor. Sections 50.47(b) and (c) outline the planning standards and size of Emergency Planning Zones, respectively, that are to be considered in emergency plans and Appendix E to 10 CFR Part 50 identifies the information that must be included in emergency plans.

In the permanently defueled condition, the risk associated with the plant has been significantly reduced. Under the provisions of 10 CFR 50.54(q), when a change to an emergency plan is made, the change is evaluated against the bases for commitments made in the plan to determine whether there is a decrease in effectiveness. It is not a decrease in effectiveness if the reduction in the commitment is commensurate with a reduction in the bases for that commitment. Therefore, if the licensee satisfactorily demonstrates that the calculated maximum offsite dose for the postulated releases evaluated in the facility's DSAR is less than the U.S. Environmental Protection Agency (EPA) Protective Action Guides (PAGs), it is to be concluded that there has not been a reduction in the bases that require offsite emergency planning. This conclusion, in turn, satisfies the special circumstance criteria of 10 CFR 50.12(a)(2)(ii), i.e., requiring the licensee to comply with the requirements of 10 CFR 50.54(q) would not serve the underlying purpose of the rule.

With respect to the other "special circumstances" noted above, nothing in the documentation of the emergency planning rulemakings discussed the applicability or appropriateness of these regulations for facilities in the permanently shutdown and defueled condition; the subject regulations were established for power operations because such conditions create the potential for an accident with offsite consequences.

Because experience has shown that there are significant costs incurred in complying with the offsite emergency planning requirements of 10 CFR 50.54(q) and because such costs are ultimately borne by the general public as ratepayers, a demonstration by the licensee that the basis for the offsite emergency planning requirement no longer exists will be sufficient grounds for concluding, *prima facie,* that compliance would result in "costs that are significantly in excess of those contemplated when the regulation was adopted," i.e., 10 CFR 50.12(a)(2)(iii) is satisfied; furthermore, with such a demonstration, there is present "other material circumstance not considered when the regulation was adopted for which it would be in the public interest to grant an exemption," i.e., 10 CFR 50.12(a)(2)(vi) is satisfied.

III. REVIEW PROCEDURES

The review consists of the Project Manager's evaluation of the plant specific information submitted by the applicant/licensee using the foregoing Acceptance Criteria. The bulk of this information should be found in the applicant's Defueled Safety Analysis Report. This information may be supplemented by a personal visit to the site by the reviewer and meetings with the applicant.

The design basis accidents requiring evaluation for a defueled facility are a small subset of those considered for an operating facility and are limited to the following:

1. a fuel handling incident

2. a spent fuel cask drop

3. accidents associated with radioactive waste storage or processing

As noted in Section I of this SRP, the only possible addition to this list would be a design basis accident in the licensee's licensing basis for the operating facility which remains valid for the facility in the defueled condition.

If a specific accident analysis, which is contained in the FSAR for the operating plant, has been previously found to be acceptable to the Commission and there has been no meaningful change in the key parameters and assumptions which served as the basis for the original judgment, the original conclusion of acceptability will remain valid.

There are no relevant beyond design basis events which could require offsite response capabilities and which would therefore need to be evaluated in order to grant the exemption request. NUREG-0396 [2] provided recommendations on emergency planning zones and a range of time values in which emergency response officials should be prepared to implement protective actions. The NUREG also presented the chemical and physical characteristics of those radionuclides which contribute most significantly to human exposure. These radionuclides primarily consist of short-lived isotopes in the form of noble gases and volatiles such as iodine.

In a policy statement [3] concerning the planning basis for emergency response, the Commission stated in reference to NUREG-0396: “In endorsing this guidance, the Commission recognizes that it is appropriate and prudent for emergency planning guidance to take into consideration the principal characteristic (such as nuclides released and distance likely to be involved) of a spectrum of design basis and core melt accidents.

Thus, one of the principal considerations which formed the underlying basis of the emergency planning rule was the radionuclide distribution associated with the design basis and core melt (beyond design basis) accidents.

After a permanently shutdown plant has undergone a modest level of decay (60 to 90 days), the nuclide distribution is significantly different than that upon which the emergency planning rule was based. Many of the requirements of the emergency planning rule were based upon a spectrum of accidents which may result in early fatalities and early injury due to the presence of shorter-lived isotopes. The consequences of beyond design basis events for permanently shutdown plants are dominated by long-lived isotopes. Thus, the health consequences are dominated by the risk of latent cancer fatalities due to long term exposures; there are no early fatalities and the risk of early injury is negligible. [4] As such, many of the requirements of the emergency planning rule no longer apply to permanently shutdown plants which have undergone a modest level of decay. These requirements include the ten mile radius emergency planning zone and protective action recommendations.

In addition to the reasons cited above, approval for an exemption from the emergency planning rule requirements is justified based upon the inherently large safety margins associated with the storage of spent fuel. The simplicity and robustness of spent fuel pool and dry cask storage designs make the occurrence of a beyond design basis event of such low probability that they can be eliminated from consideration on the basis of risk alone. These design characteristic include seismic capability, versatile structural capability, passive cooling capability and passive shielding capability.

Therefore, for the reasons cited above, there are no beyond design basis events which need to be considered for approval of an exemption consistent with Attachment I of this Standard Review Plan.

The reviewer must determine whether or not the acceptance criteria identified in Section II of this SRP have been satisfied. Any deficiencies should be clearly identified and should form the basis for a request for additional information to the applicant. If any deficiencies remain at the conclusion of the review, they must be identified in the Safety Evaluation Report and subsequently resolved with participation of higher level NRC management.

It should be recognized that the detailed application of the acceptance criteria will in some instances require the exercise of judgement on the part of the reviewer. The reviewer is expected to achieve a safety finding based on a traditional risk-informed reasonableness threshold which treats the acceptance criteria as akin to an “adequate protection” standard rather than seeking to impose a standard of absolute safety.

The reviewer should confirm that the applicant/licensee has informed the appropriate officials of the State, local government, and the Federal Emergency Management Agency of the exemption request.

IV. EVALUATION FINDINGS

The desired evaluation findings should be substantially equivalent to the following statement:

The Commission has completed its review of the licensee’s request for an exemption from certain requirements of 10 CFR 50.54(q), 10 CFR 50.54(t), 10 CFR 50.47(b) and (c), and Appendix E to 10 CFR Part 50 and concludes that the request is acceptable in view of the greatly reduced offsite radiological consequences from any reasonably conceivable accident which could occur with the plant in a permanently shutdown status. The specific elements of 10 CFR 50.47 and Appendix E to 10 CFR Part 50 which are to be exempted and those which will remain in effect are listed in Attachment I.

The Commission finds that the potential dose to the general public from any reasonably conceivable accident would not exceed EPA PAGs and, for the bounding accident, the length of time available provides confidence that offsite measures for the public could be taken without preplanning.

The Commission, based on its independent evaluation, agrees with the licensee’s analyses and concludes that sufficient bases have been presented for approval of the exemption request. The Commission has determined that pursuant to 10 CFR 50.12(a)(1), this exemption is authorized by law, will not present an undue risk to the public health and safety, and is consistent with the common defense and security. Furthermore, the Commission finds that there are special circumstances presented that satisfy the requirements of 10 CFR 50.12(a)(2)(ii), (iii), and (vi). The Commission hereby grants the requested exemption.

V. IMPLEMENTATION

The following is intended to provide guidance to licensees regarding the Commission’s plan for using this SRP.

Except in those cases in which a licensee proposes an acceptable alternative method, the methods described herein will be used by the Commission in its evaluation of requests by licensees of facilities that are planning to, or are already permanently shutdown and defueled, for exemption from certain offsite emergency planning requirements and reductions in the scope of onsite emergency plans.

It is important to note that those provisions of 10 CFR 50.54 (q) which permit a licensee to make changes to the Emergency Plan without prior Commission approval, remain in effect. The licensee must simply ascertain that the changes do not decrease the effectiveness of the Plan and that the Plan meets the requirements of Appendix E to 10 CFR Part 50 as modified by approval of the exemption request.

VI. REFERENCES

1. 10 CFR 50.54(q).

2. NUREG-0396, “Planning Basis for Development of State and Local Government Radiological Emergency Response Plans In Support of Light Water Nuclear Power Plants,” December 1978.

3. 44 FR 61123, “Planning Basis for Emergency Responses to Nuclear Power Reactor Accidents,” October 23, 1979.

4. NUREG-1353, “Regulatory Analysis for Resolution of Generic Issue 82, Beyond Design Basis Accidents in Spent Fuel Pools,” April 1989.

ATTACHMENT I

Exemptions To 10 CFR 50.54(Q), 10 CFR 50.54(T), 10 CFR 50.47, and Appendix E To Part 50

The licensee seeks exemption to the provisions of 10 CFR 50.54(q) which require the licensee "to follow and maintain in effect emergency plans which meet the standards in §50.47(b) and the requirements in appendix E of this part" in their entirety. Those requirements in 10CFR50.47(b) and (c) and Appendix E to Part 50 which will continue to apply in the defueled plant condition are shown below in the unshaded text; those requirements from which exemption is to be granted are indicated as shaded text. The licensee also seeks exemption to the provisions of 10 CFR 50.54(t).

§50.47 Emergency plans

(b) The onsite and, except as provided in paragraph (d) of this section, offsite emergency response plans for nuclear power reactors must meet the following standards:

1. Primary responsibilities for emergency response by the nuclear facility licensee and by State and local organizations within the Emergency Planning Zones have been assigned, the emergency responsibilities of the various supporting organizations have been specifically established, and each principal response organization has staff to respond and to augment its initial response on a continuous basis.

2. On-shift facility licensee responsibilities for emergency response are unambiguously defined, adequate staffing to provide initial facility accident response in key functional areas is maintained at all times, timely augmentation of response capabilities is available and the interfaces among various onsite response activities and offsite support and response activities are specified.

3. Arrangements for requesting and effectively using assistance resources have been made, arrangements to accommodate State and local staff at the licensee's near-site Emergency Operations Facility have been made, and other organizations capable of augmenting the planned response have been identified.

4. A standard emergency classification and action level scheme, the bases of which include facility system and effluent parameters, is in use by the nuclear facility licensee, and State and local response plans call for reliance on information provided by the facility licensees for determination of minimum initial offsite response measures.

5. Procedures have been established for notification, by the licensee, of State and local response organizations and for notification of emergency personnel by all organizations; the content of initial and follow-up messages to response organizations and the public has been established; and means to provide early notification and clear instruction to the populace within the plume exposure pathway Emergency Planning Zone have been established.

6. Provisions exist for prompt communications among principal response organizations to emergency personnel and to the public.

7. Information is made available to the public on a periodic basis on how they will be notified and what their initial actions should be in an emergency (e.g., listening to a local broadcast station and remaining indoors), the principal points of contact with the news media for dissemination of information during an emergency (including the physical location or locations) are established in advance, and procedures for coordinated dissemination of information to the public are established.

8. Adequate emergency facilties and equipment to support the emergency response are provided and maintained.

9. Adequate methods, systems, and equipment for assessing and montoring actual or potential offsite consequences of a radiological emergency condition are in use.

10. A range of protective actions have been developed for the plume exposure pathway EPZ for emergency workers and the public. Guidelines for the choice of protective actions during an emergency, consistent with Federal guidance, are developed and in place, and protective actions for the ingestion exposure pathway EPZ appropriate to the locale have been developed.

11. Means for controlling radiological exposures, in an emergency, are established for emergency workers. The means for controlling radiological exposures shall include exposure guidelines consistent with EPA Emergency Worker and Lifesaving Activity Protective Action Guides.

12. Arrangements are made for medical services for contaminated injured individuals.

13. General plans for recovery and reentry are developed.

14. Periodic exercises are (will be) conducted to evaluate major portions of emergency response capabilities, periodic drills are (will be) conducted to develop and maintain key skills, and deficiencies identified as a result of exercises or drills are (will be) corrected.

15. Radiological emergency response training is provided to those who may be called on to assist in an emergency.

16. Responsibilities for plan development and review and for distribution of emergency plans are established, and planners are properly trained.

(c)(2) Generally, the plume exposure pathway EPZ for nuclear power plants shall consist of an area about 10 miles (16 km) in radius and the ingestion pathway EPZ shall consist of an area about 50 miles (80 km) in radius. The exact size and configuration of the EPZs surrounding a particular nuclear power reactor shall be determined in relation to local emergency response needs and capabilities as they are affected by such conditions as demography, topography, land characteristics, access routes, and jurisdictional boundaries. The size of the EPZs also may be determined on a case-by-case basis for gas-cooled nuclear reactors and for reactors with an authorized power level less than 250 MW thermal. The plans for the ingestion pathway shall focus on such actions as are appropriate to protect the food ingestion pathway.

Appendix E to Part 50

Emergency Planning and Preparedness for Production and Utilization Facilities

III. The Final Safety Analysis Report

The Final Safety Analysis Report shall contain the plans for coping with emergencies. The plans shall be an expression of the overall concept of operation; they shall describe the essential elements of advance planning that have been considered and the provisions that have been made to cope with emergency situations. The plans shall incorporate information about the emergency response roles of supporting organizations and offsite agencies. That information shall be sufficient to provide assurance of coordination among the supporting groups and with the licensee.

The plans submitted must include a description of the elements set out in Section IV for the Emergency Planning Zones (EPZs) to an extent sufficient to demonstrate that the plans provide reasonable assurance that adequate protective measures can and will be taken in the event of an emergency.

IV. Content of Emergency Plans

The applicant's emergency plans shall contain, but not necessarily be limited to, information needed to demonstrate compliance with the elements set forth below, i.e., organization for coping with radiation emergencies, assessment action, activation of emergency organization, notification procedures, emergency facilities and equipment, training, maintaining emergency preparedness, and recovery. In addition, the emergency response plans submitted by an applicant for a nuclear power reactor operating license shall contain information needed to demonstrate compliance with the standards described in §50.47(b), and they will be evaluated against those standards. The nuclear power reactor operating license applicant shall also provide an analysis of the time required to evacuate and for taking other protective actions for various sectors and distances within the plume exposure pathway EPZ for transient and permanent populations.

A. Organization

The organization for coping with radiological emergencies shall be described, including definition of authorities, responsibilities, and duties of individuals assigned to the licensee's emergency organization and the means for notification of such individuals in the event of an emergency. Specifically, the following shall be included:

1. A description of the normal plant operating organization.

2. A description of the onsite emergency response organization with a detailed discussion of:

a. Authorities, responsibilities, and duties of the individual(s) who will take charge during an emergency;

b. Plant staff emergency assignments;

c. Authorities, responsibilities, and duties on an onsite emergency coordinator who shall be in charge of the exchange of information with offsite authorities responsible for coordinating and implementing offsite emergency measures.

5. A description, by position and function to be performed, of the licensee's headquarters personnel who will be sent to the plant site to augment the onsite emergency organization.

6. Identification, by position and function to be performed, of persons within the licensee organization who will be responsible for making offsite dose projections, and a description of how these projections will be made and the results transmitted to State and local authorities, NRC, and other appropriate governmental agencies.

7. Identification, by position and function to be performed, of other employees of the licensee with special qualifications for coping with emergency conditions that may arise. Other persons with special qualifications, such as consultants, who are not employees of the licensee and who may be called upon for assistance for emergencies shall also be identified. The special qualifications of these persons shall be described.

8. A description of the local offsite services to be provided in support of the licensee's emergency organization.

9. Identification of, and assistance expected from, appropriate State, local, and Federal agencies with responsibilities for coping with emergencies.

10. Identification of the State and/or local officials responsible for planning for, ordering, and controlling appropriate protective actions, including evacuations when necessary.

B. Assessment Actions

The means to be used for determining the magnitude of and for continually assessing the impact of the release of radioactive materials shall be described, including emergency action levels that are to be used as criteria for determining the need for notification and participation of local and State agencies, the Commission, and other Federal agencies, and the emergency action levels that are to be used for determining when and what type of protective measures should be considered within and outside the site boundary to protect health and safety. The emergency action levels shall be based on in-plant conditions and instrumentation in addition to onsite and offsite monitoring. These emergency action levels shall be discussed and agreed on by the applicant and State and local governmental authorities and approved by the NRC. They shall also be reviewed with the State and local governmental authorities on an annual basis.

C. Activation of Emergency Organization

The entire spectrum of emergency conditions that involve the alerting or activating of progressively larger segments of the total emergency organization shall be described. The communication steps to be taken to alert or activate emergency personnel under each class of emergency shall be described. Emergency action levels (based not only on onsite and offsite radiation monitoring information but also on readings from a number of sensors that indicate a potential emergency, such as the pressure in containment and the response of the Emergency Core Cooling System) for notification of offsite agencies shall be described. The existence, but not the details, of a message authentication scheme shall be noted for such agencies. The emergency classes defined shall include: (1) notification of unusual events, (2) alert, (3) site area emergency, and (4) general emergency. These classes are further discussed in NUREG-0654; FEMA-REP-1.

D. Notification Procedures

Administrative and physical means for notifying local, State, and Federal officials and agencies and agreements reached with these officials and agencies for the prompt notification of the public and for public evacuation or other protective measures, should they become necessary, shall be described. This description shall include identification of the appropriate officials, by title and agency, of the State and local government agencies within the EPZs.

Provisions shall be described for yearly dissemination to the public within the plume exposure pathway EPZ of basic emergency planning information, such as the methods and times required for public notification and the protective actions planned if an accident occurs, general information as to the nature and effects of radiation, and a listing of local broadcast stations that will be used for dissemination of information during an emergency. Signs or other measures shall also be used to disseminate to any transient population within the plume exposure pathway EPZ appropriate information that would be helpful if an accident occurs.

A licensee shall have the capability to notify responsible State and local government agencies within 15 minutes after declaring an emergency. The licensee shall demonstrate that the State/local officials have the capability to make a public notification decision promptly on being informed by the licensee of an emergency condition. By February 1, 1982, each nuclear power reactor licensee shall demonstrate that administrative and physical means have been established for alerting and providing prompt instructions to the public within the plume exposure pathway EPZ. The four-month period in 10 CFR 50.54(s)(2) for the correction of emergency plan deficiencies shall not apply to the initial installation of this public notification system that is required by February 1, 1982. The four-month period will apply to the correction of deficiencies identified during the initial installation and testing of the prompt public notification systems as well as those deficiencies discovered thereafter. The design objective of the prompt public notification system shall be to have the capability to essentially complete the initial notification of the public within the plume exposure pathway EPZ within about 15 minutes. The use of this notification capability will range from immediate notification of the public (within 15 minutes of the time that State and local officials are notified that a situation exists requiring urgent action) to the more likely events where there is substantial time available for the State and local governmental officials to make a judgment whether or not to activate the public notification system. Where there is a decision to activate the notification system, the State and local officials will determine whether to activate the entire notification system simultaneously or in a graduated or staged manner. The responsibility for activating such a public notification system shall remain with the appropriate governmental authorities.

E. Emergency Facilities and Equipment

Adequate provisions shall be made and described for emergency facilities and equipment, including:

1. Equipment at the site for personnel monitoring;

2. Equipment for determining the magnitude of and for continuously assessing the impact of the release of radioactive materials to the environment;

3. Facilities and supplies at the site for decontamination of onsite individuals;

4. Facilities and medical supplies at the site for appropriate emergency first aid treatment;

5. Arrangements for the services of physicians and other medical personnel qualified to handle radiation emergencies on-site;

6. Arrangements for transportation of contaminated injured individuals from the site to specifically identified treatment facilities outside the site boundary;

7. Arrangements for treatment of individuals injured in the support of licensed activities on the site at treatment facilities outside the site boundary;

8. A licensee onsite technical support center and a licensee near-site emergency operations facility from which effective direction can be given and effective control can be exercised during an emergency;

9. At least one onsite and one offsite communications system; each system shall have a backup power source.

All communication plans shall have arrangements for emergencies, including titles and alternates for those in charge at both ends of the communication links and the primary and backup means of communication. Where consistent with the function of the governmental agency, these arrangements will include:

a. Provisions for communications with contiguous State/local governments within the plume exposure pathway EPZ. Such communications shall be tested monthly.

b. Provision for communications with Federal emergency response organizations. Such communications systems shall be tested annually.

c. Provisions for communications among the nuclear power reactor control room, the onsite technical support center, and the near-site emergency operations facility; and among the nuclear facility, the principal State and local emergency operations centers, and the field assessment teams. Such communications systems shall be tested annually.

d. Provisions for communications by the licensee with NRC Headquarters and the appropriate NRC Regional Office Operations Center from the nuclear power reactor control room, the onsite technical support center, and the near-site emergency operations facility. Such communications shall be tested monthly.

F. Training

1. The program to provide for: (a) The training of employees and exercising, by periodic drills, of radiation emergency plans to ensure that employees of the licensee are familiar with their specific emergency response duties, and (b) The participation in the training and drills by other persons whose assistance may be needed in the event of a radiation emergency shall be described. This shall include a description of specialized initial training and periodic retraining programs to be provided to each of the following categories of emergency personnel:

    i. Directors and/or coordinators of the plant emergency organization;

    ii. Personnel responsible for accident assessment, including control room shift personnel;

    iii. Radiological monitoring teams;

    iv. Fire control teams (fire brigades);

    v. Repair and damage control teams;

    vi. First aid and rescue teams;

    vii. Medical support personnel;

    viii. Licensee’s headquarters support personnel;

    ix. Security personnel.

    In addition, a radiological orientation training program shall be made available to local services personnel; e.g., local emergency services/Civil Defense, local law enforcement personnel, local news media persons.

11. The plan shall describe provisions for the conduct of emergency preparedness exercises as follows: Exercises shall test the adequacy of timing and content of implementing procedures and methods, test emergency equipment and communications networks, test the public notification system, and ensure that emergency organization personnel are familiar with their duties.

b. Each licensee at each site shall conduct an exercise of its onsite emergency plan every 2 years. The exercise may be included in the full participation biennial exercise required by paragraph 2.c. of this section. In addition, the licensee shall take actions necessary to ensure that adequate emergency response capabilities are maintained during the interval between biennial exercises by conducting drills, including at least one drill involving a combination of some of the principal functional areas of the licensee's onsite emergency response capabilities. The principal functional areas of emergency response include activities such as management and coordination of emergency response, accident assessment, protective action decision-making, and plant system repair and corrective actions. During these drills, activation of all of the licensee's emergency response facilities (Technical Support Center (TSC), Operations Support Center (OSC), and the Emergency Operations Facility (EOF)) would not be necessary, licensees would have the opportunity to consider accident management strategies, supervised instruction would be permitted, operating staff would have the opportunity to resolve problems (success paths) rather than have controllers intervene, and the drills could focus on onsite training objectives.

c. Offsite plans for each site shall be exercised biennially with full participation by each offsite authority having a role under the plan. Where the offsite authority has a role under a radiological response plan for more than one site, it shall fully participate in one exercise every two years and shall, at least, partially participate in other offsite plan exercises in this period.

d. A State should fully participate in the ingestion pathway portion of exercises at least once every six years. In States with more than one site, the State should rotate this participation from site to site.

e. Licensees shall enable any State or local Government located within the plume exposure pathway EPZ to participate in the licensee's drills when requested by such State or local Government.

f. Remedial exercises will be required if the emergency plan is not satisfactorily tested during the biennial exercise, such that NRC, in consultation with FEMA, cannot find reasonable assurance that adequate protective measures can be taken in the event of a radiological emergency. The extent of State and local participation in remedial exercises must be sufficient to show that appropriate corrective measures have been taken regarding the elements of the plan not properly tested in the previous exercise.

g. All training, including exercises, shall be provided for formal critiques in order to identify weak or deficient area that need correction. Any weaknesses or deficiencies that are identified shall be corrected.

h. The participation of State and local governments in an emergency exercise is not required to the extent that the applicant has identified those governments as refusing to participate further in emergency planning activities, pursuant to 10

CFR 50.47(c)(1). In such cases, an exercise shall be held with the applicant or licensee and such governmental entities as elect to participate in the emergency planning process.

G. Maintaining Emergency Preparedness

Provisions to be employed to ensure that the emergency plan, its implementing procedures, and emergency equipment and supplies are maintained up to date shall be described.

H. Recovery

Criteria to be used to determine when, following an accident, reentry of the facility would be appropriate or when operation could be resumed shall be described.

V. Implementing Procedures

No less than 180 days prior to the scheduled issuance of an operating license for a nuclear power reactor or a license to possess nuclear material the applicant's detailed implementing procedures for its emergency plan shall be submitted to the Commission as specified in §50.4. Licensees who are authorized to operate a nuclear power facility shall submit any changes to the emergency plan or procedures to the Commission, as specified in §50.4, within 30 days of such changes.

C
DECOMMISSIONING STANDARD REVIEW PLAN: REQUEST FOR EXEMPTION OF PERMANENTLY DEFUELED FACILITIES FROM CERTAIN REQUIREMENTS OF 10 CFR 73, PHYSICAL PROTECTION OF PLANTS AND MATERIALS

(NOTE: See "General Guidance" which follows this draft DSRP.)

Review Responsibilities

Primary – Applicant's Project Manager

Secondary – None

I. AREAS OF REVIEW

A facility which is in a permanently shutdown and defueled condition poses a significantly reduced risk to the public health and safety. After a "Certification of Permanent Cessation of Operation and Removal of Fuel from the Reactor Vessel" has been docketed in accordance with the provisions of 10 CFR 50.82, and in view of the reduced risk, it is anticipated that the licensee of such a facility will seek exemption from certain requirements of 10 CFR 73 which are no longer appropriate. This review is expected to provide the basis for exempting the applicant/licensee from the specific requirements of 10 CFR 73 listed in Attachment I of this Standard Review Plan (SRP). A licensee has the option of adopting all of Attachment I by reference in an exemption request, adopting a subset of specific exemption requests from Attachment I, and/or proposing alternative plant-specific exemptions.

The review is performed by the licensee's Project Manager. The review involves evaluation of relevant portions of the licensee's Defueled Security Plan (DSP), the licensee's Defueled Safety Analysis Report (DSAR), and any other relevant information provided in the exemption request against the acceptance criteria of Section II of this SRP.

II. ACCEPTANCE CRITERIA

It is stated in 10 CFR 73.55(a) that "The licensee shall establish and maintain an onsite physical protection system and security organization which will have as its objective to provide high assurance that activities involving special nuclear material are not inimical to the common defense and security and do not constitute an unreasonable risk to the public health and safety."

Pursuant to 10 CFR 73.5, "Specific exemptions," the Commission may, upon application of any interested person or upon its own initiative, grant such exemptions from the requirements of the regulations in this part as it determines are authorized by law and will not endanger life or property or the common defense and security, and are otherwise in the public interest. The Code of Federal Regulations at 10 CFR 73.55(a) allows the Commission to authorize a licensee to provide alternative measures for protection against radiological sabotage, provided the licensee demonstrates that the proposed measures meet the general performance requirements of the regulation and that the overall level of system performance provides protection against radiological sabotage equivalent to that provided by the regulation.

The underlying purpose of 10 CFR 73.55 is to provide reasonable assurance that adequate security measures can be taken in the event of an act of radiological sabotage. In the permanently shutdown and defueled condition, the radiological risk from the licensee's unit will be significantly less than the risk from an operating unit, i.e., the potential source term associated with the remaining design-basis accidents and radiological sabotage will have decreased significantly. A demonstration by the licensee of a defueled facility that the calculated maximum offsite doses associated with the remaining design basis accidents do not exceed the U.S. Environmental Protection Agency (EPA) Protective Action Guides is considered to be appropriate justification to exempt the licensee from the Commission's requirements for offsite emergency planning. [1] Likewise, if it can be reasonably shown that an exemption to Part 73 would not diminish a facility's security to the point that a credible act of radiological sabotage could give rise to offsite doses which would exceed the EPA PAGs, then it is reasonable to conclude that full compliance with Part 73 is not essential to satisfy the underlying purpose of the rule.

The costs associated with security activities at nuclear facilities should be commensurate with the risk. Because, as previously discussed, the offsite radiological risk associated with the shutdown and defueled facility has been significantly reduced, requiring full compliance with the applicable regulations would result in costs that do not provide any additional benefit. Full compliance with certain requirements of 10 CFR 73 that are clearly intended for operating reactor facilities will result in undue financial burdens for licensees and their ratepayers. Granting exemptions in such circumstances is clearly in the public interest.

As noted in Section I, "AREAS OF REVIEW," of this SRP, examples of specific Part 73 exemption requests for permanently shutdown and defueled facilities are provided in Attachment I. The bases for granting such exemptions are also noted.

III. REVIEW PROCEDURES

The review consists of the Project Manager's evaluation of the plant specific information submitted by the applicant/licensee using the foregoing Acceptance Criteria. The bulk of this information should be found in the applicant's Defueled Security Plan and Defueled Safety Analysis Report. This information may be supplemented by a personal visit to the site by the reviewer and meetings with the applicant.

The reviewer must determine whether or not the acceptance criteria identified in Section II of this SRP have been satisfied. Any deficiencies should be clearly identified and should form the basis for a request for additional information to the applicant. If any deficiencies remain at the conclusion of the review, they must be identified in the Safety Evaluation Report and subsequently resolved with participation of higher level NRC management.

It should be recognized that the detailed application of the acceptance criteria will in some instances require the exercise of judgement on the part of the reviewer. The reviewer is expected to achieve a finding based on a risk-informed reasonableness threshold which treats the acceptance criteria as akin to an “adequate protection” standard rather than seeking to impose a standard of absolute protection.

IV. EVALUATION FINDINGS

The desired evaluation findings should be substantially equivalent to the following statement:

The Commission has completed its review of the licensee’s request for an exemption from certain requirements of 10 CFR 73. Pursuant to 10 CFR 73.5, “Specific exemptions,” the Commission may, upon application of any interested person or upon its own initiative, grant such exemptions from the requirements of the regulations in this part as it determines are authorized by law and will not endanger life or property or the common defense and security, and are otherwise in the public interest. The Code of Federal Regulations at 10 CFR 73.55 allows the Commission to authorize a licensee to provide alternative measures for protection against radiological sabotage, provided the licensee demonstrates that the proposed measures meet the general performance requirements of the regulation and that the overall level of system performance provides protection against radiological sabotage equivalent to that provided by the regulation.

The underlying purpose of 10 CFR 73.55 is to provide reasonable assurance that adequate security measures can be taken in the event of an act of radiological sabotage. In the permanently shutdown and defueled condition, the radiological risk from the licensee’s unit will be significantly less than the risk from an operating unit, i.e., the potential source term associated with the remaining design-basis accidents and radiological sabotage will have decreased significantly.

For the foregoing reasons, the Commission has determined that the proposed alternative measures for protection against sabotage meet the same assurance objective and the general performance requirements of 10 CFR 73.55 associated with the reduced risk of radiological sabotage for a permanently shutdown reactor site that has all of the fuel in the spent fuel pool. In addition, the staff has determined that the overall level of the proposed system’s performance, as limited by this exemption, would not result in a reduction in the physical protection capabilities for the protection of special nuclear material or of the unit. Specifically, a limited exemption is being granted for the following specific requirements: [List of specific exemptions approved.]

Accordingly, the Commission has determined pursuant to 10 CFR 73.5, this exemption is authorized by law, will not endanger life or property or the common defense and security, and is otherwise in the public interest. Therefore, the Commission hereby grants the licensee a limited exemption as described above from those requirements of 10 CFR 73.55 at [Unit Name] in its permanently defuel condition. Furthermore, pursuant to 10 CFR 51.32, the Commission has determined that this exemption will not have a significant effect on the quality of the human environment.

V. IMPLEMENTATION

The following is intended to provide guidance to licensees regarding the Commission's plan for using this SRP.

Except in those cases in which a licensee proposes an acceptable alternative method, the methods described herein will be used by the Commission in its evaluation of requests by licensees of facilities that are planning to, or are already permanently shutdown and defueled, for exemption from certain requirements of 10 CFR 73.

VI. REFERENCES

1. Standard Review Plan – Exemption Request For Permanently Defueled Facilities

ATTACHMENT I

Exemption 1

An exemption is requested from the requirement of 10 CFR 73.55(a) that suspension of "any safeguards measures pursuant to § 73.55 in an emergency when this action is immediately needed to protect the public health and safety and no action consistent with license conditions and technical specification that can provide adequate or equivalent protection is immediately apparent" "must be approved as a minimum by a licensed senior operator prior to taking the action."

Basis for Granting Exemption 1

The presumption is that with the licensee's certifications of permanent cessation of operations and permanent removal of fuel from the reactor vessel pursuant to 10 CFR 50.82, all of the facility's senior reactor operator licenses will be surrendered and this function will be replaced with certified fuel handlers. The certified fuel handler will be the individual on shift with knowledge about protecting the fuel from situations that may occur at the site. Therefore, the exemption is necessary to permit reassignment of the authority to suspend safeguards measures to the certified fuel handlers. Although the certified fuel handler is not an individual licensed by the NRC as is the licensed senior reactor operator, this individual's responsibilities are part of the licensee's certification process and training program. (It is assumed as a prerequisite that the NRC has reviewed and approved the licensee's certification process and training program for fuel handlers.)

Exemption 2

An exemption is requested from the requirement of 10 CFR 73.55(e)(1) that "Onsite secondary power supply systems for alarm annunciator equipment and non-portable communications equipment as required in paragraph (f) of this section [i.e., 10 CFR 73.55(f)] must be located within vital areas."

Basis for Granting Exemption 2

The regulations require that the secondary power supply be located in vital areas. Because vital areas, by definition, no longer exist at defueled sites, this requirement becomes moot. Also, because of the reduced size of the protected area, it may not be possible to relocate the power supply into the protected area. In addition, hardened alarm stations may be located outside the protected area, which would require at least a portion of the equipment to be installed outside of the protected area. For the reasons stated above, it is believed that the presence of the equipment/function is the more significant issue, not its location.

Exemption 3

An exemption is requested from the requirement of 10 CFR 73.55(d)(1) that "The individual responsible for the last access control function (controlling admission to the protected area) must be isolated within a bullet-resisting structure as described in paragraph (c)(6) of this section [i.e., 10 CFR 73.55(c)(6)] to assure his or her ability to respond or to summon assistance."

Basis for Granting Exemption 3

It is expected that the entrance to the spent fuel building protected area at a defueled site will be controlled by a guard (an armed individual) who will have communication capability with the central alarm station. This person will be responsible for access control, search functions, and authorizations into the protected area for all personnel, and he (she) can summon assistance in an emergency. When personnel access and work-related functions are not required within the spent fuel building, the building must be locked and alarmed at all access points and monitored by the central alarm station.

Exemption 4

An exemption is requested from the requirement of 10 CFR 73.55(h)(3) that "The total number of guards, and armed, trained personnel immediately available at the facility to fulfill these response requirements shall nominally be ten (10), unless specifically required otherwise on a case by case basis by the Commission; however, this number may not be reduced to less than five (5) guards."

Basis for Granting Exemption 4

With the transition from an operating reactor site to a defueled facility, the size of the protected area is reduced to a single, smaller area that needs to be monitored and protected. It is expected that the licensee will propose a security program for the defueled site that provides both security related equipment and a security force, some of whom will be armed, to protect the spent fuel from acts of radiological sabotage. The armed security force members on site will be trained and qualified, and will be able to react to different scenarios based on preplanned contingency events. In addition, it is anticipated that the licensee will have coordinated with local law enforcement agencies to respond to threats against the site.

Exemption 5

An exemption is requested from the requirement of 10 CFR 73.55(e)(1) that, in addition to a central alarm station located within the protected area, there must be "at least one other continuously manned station not necessarily onsite , so that a single act cannot remove the capability of calling for assistance or otherwise responding to an alarm."

Basis for Granting Exemption 5

With the reduction of the protected area at a defueled site to incorporate only the spent fuel pool and the surrounding building structure, the secondary alarm station will no longer be required because the central alarm station or another security station, also located outside the protected area, will be required to be bullet resistant. Accordingly, a redundant secondary alarm station is not necessary to guarantee emergency offsite communications to local law enforcement agencies.

Exemption 6

An exemption is requested from the requirement of 10 CFR 73.55(f)(4) that "Non-portable communications equipment controlled by the licensee and required by this section [i.e., 10 CFR 73.55(f)] shall remain operable from independent power sources in the event of the loss of normal power."

Basis for Granting Exemption 6

Assuming the non-portable communication equipment is located in the facility's central alarm station, and is not on the backup power supply, the exemption is to be granted if the licensee commits to providing an appropriate alternative communication system (e.g., portable radio equipment) to contact the local law enforcement authorities during an emergency.

Standard Review Plan for Security Exemption Request

The following general comments are offered concerning requests for exemptions from the provisions of 10 CFR 73 for decommissioning facilities and the proposed SRP:

- As noted in Maine Yankee's request for exemption from certain provisions of 10 CFR 73, the NRC has acknowledged that the provisions of the current regulations do not provide clear guidance relative to the reduction of security requirements for permanently shut down plants. NUREG-1497, "Interim Licensing Criteria for Physical Protection of Certain Storage of Spent Fuel," issued 11/94, and Proposed Rule Making to 10 CFR Parts 60, 72, 73, and 75 (60 FR 42079, published 8/15/95) both contain discussions relative to the lack of clear regulatory guidance provided for the security requirements for permanently shut down power reactors. The proposed SRP attempts to address this concern as follows:

    1. A general acceptance criterion for an exemption request is proposed that, in a manner analogous to criteria justifying the exemption from off site emergency planning, if it can be shown that there is no creditable sabotage event that would result in doses at the site boundary in excess of the EPA Protective Action Guidelines as a result of exempting a licensee from a specific provision of 10 CFR 73, there is adequate justification for granting the exemption. (Although this is an attempt to provide a quasi-objective basis for granting an exemption, it is clear that any supporting analysis/justification still must rely heavily on subjective judgements.)

    2. Attachment I of the SRP offers a list of specific exemptions previously granted by the Staff along with the basis cited by the Staff in the associated SER. Clearly such a listing could be conceived as a living document which could capture additional specific exemptions to 10 CFR 73 exemptions as they are approved by the Staff.

- In the case of requests for exemption from the requirements of off site emergency planning, it is believed that there is no need for the licensee to provide the Decommissioned Emergency Plan to support such a request. However, in the case of requests for exemption from certain provisions of 10 CFR 73, it seems clear that the licensee would have to provide the Decommissioned Security Plan, or appropriate elements thereof, in order to provide the Staff with an appropriate basis/justification for granting an exemption.

- Because of the requirements for confidentiality generally associated with the security plan for a specific unit, it can be extremely difficult (if not impossible) for a third party to understand the precise basis for the Staff's granting of a security-related exemption. This, in turn, would seem to argue for a reasonable amount of preliminary interactions with the Staff prior to filing the formal exemption request.

D
DECOMMISSIONING STANDARD REVIEW PLAN: FINANCIAL PROTECTION REQUIREMENTS LIMITS: EXEMPTION REQUEST FOR PERMANENTLY DEFUELED FACILITIES

Review Responsibilities

Primary - Applicant - Project Manager

Secondary - None

I. AREAS OF REVIEW

A facility in a permanently shutdown and defueled condition poses a significantly reduced risk to the public health and safety. After “Certifications of Permanent Cessation of Operation and Removal of Fuel from the Reactor Vessel” have been docketed in accordance with 10 CFR 50.82(a)(1) and, in view of the reduced risk, it is anticipated that the licensee of such a facility will seek exemption from certain requirements of 10 CFR 50.54(w) and 10 CFR 140.11 which are no longer appropriate. This review is expected to provide the basis for exempting the applicant/licensee from the requirement of Section 50.54(w) of 10 CFR Part 50 which requires power reactors to maintain onsite property insurance coverage in the amount of $1.06 billion[1] and Section 140.11(a)(4) of 10 CFR Part 140 which requires that a reactor with a rated capacity of 100,000 electrical kilowatts or more to maintain liability insurance of $200 million and to participate in a secondary insurance pool.[2] (Attachment I provides excerpts from 10 CFR 50.54 and 10 CFR 140 pertinent to this exemption request.)

The review is performed by the licensee’s Project Manager. The review involves evaluation of the licensee’s Defueled Safety Analysis Report (DSAR), and any other relevant information provided in the exemption request against the acceptance criteria of Section II of this SRP. In the defueled condition, there are no longer any credible design basis accidents associated with an

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[1] 10 CFR 50.54(w) is applicable to a production or utilization facility as described in 10 CFR 50.22. 10 CFR 50.22 defines Class 103 licenses as commercial and industrial facilities stating "such facility is deemed to be for industrial or commercial purposes if the facility is to be used so that more than 50 percent of the annual cost of owning and operating the facility is devoted to the production of materials, products, or energy for sale or commercial distribution..." (See Appendix I)

[2] 10 CFR 140.11 (a)(4) "for each nuclear reactor which is licensed to operate..." (See Appendix I)

operating plant from startup through full power operation. The design basis accidents relative to a defueled facility are a small subset of those considered for an operating facility and are limited to the following:

1. a fuel handling incident,

2. a spent fuel cask drop, and

3. accidents associated with radioactive waste storage or processing.

The only possible addition to this list would be any design basis accident which was included in the licensee's original licensing basis for the operating facility and which is still valid for the facility in the defueled condition. As discussed in detail in Section III of this SRP, REVIEW PROCEDURES, there are no relevant beyond design basis accidents which could result in significant on-site or off-site consequences and which would therefore need to be evaluated in order to grant the exemption request.

The subject review will be performed consistent with the requirements of the Commission's backfitting rule, 10 CFR 50.109. The policy underlying the rule, i.e., to ensure that new requirements are properly justified from a safety and cost-benefit standpoint, continues to apply during the decommissioning process so long as a facility's 10 CFR Part 50 license remains in effect. Decommissioning facilities are entitled to the same predictability, stability and protection from arbitrary actions as operating facilities.

II. ACCEPTANCE CRITERIA

Pursuant to 10 CFR 50.12(a), the Commission may, upon application by any interested person or upon its own initiative, grant exemptions from the requirements of the regulations which are:

1. authorized by law, will not present an undue risk to public health and safety, and are consistent with the commaon defense and security, and

12. present special circumstances.

Special circumstances exist when:

1. application of the regulation in the particular circumstance would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule (10 CFR 50.12(a)(2)(ii)); or

13. compliance would result in undue hardship; or other costs that are significantly in excess of those contemplated when the regulation was adopted, or that are significantly in excess of those incurred by others similarly situated (10 CFR 50.12(a)(2)(iii)); or

14. there is present other material circumstances not considered when the regulation was adopted for which it would be in the public interest to grant an exemption (10 CFR 50.12(a)(2)(vi)).[3]

In addition to the general provisions of 10 CFR 50.12(a), provisions are provided for exemptions to 10 CFR 140 within the rule itself (§140.8):

> "The Commission may, upon application of any interested person or upon its own initiative, grant such exemptions from the requirements of the regulations in this part as it determines are authorized by law and are otherwise in the public interest."

The underlying purpose of Section 50.54(w) is to provide sufficient property damage insurance coverage to ensure funding for onsite post-accident recovery stabilization and decontamination costs in the unlikely event of an accident at a nuclear power plant. The underlying purpose of Section 140.11 is to provided sufficient liability insurance to ensure funding for claims resulting from a nuclear incident or precautionary evacuation.

The financial protection limits of 10 CFR 50.54(w) and 10 CFR 140.11 were established to require a licensee to maintain sufficient insurance to cover the costs of a nuclear accident a an operating reactor. Those costs were derived from the consequences of release of radioactive material from the reactor. In the permanently defueled condition, both the risk and consequences associated with the plant have been significantly reduced.

In an operating plant, the high temperature and pressure of the reactor coolant system, as well as inventory of relatively short-lived radionuclides, contribute to both the risks and consequences of an accident. In a permanently shutdown and defueled reactor facility, the reactor will never be operated which eliminates the possibility of reactor accidents. A further reduction in risk occurs because decay heat from the spent fuel decreases exponentially upon reactor shutdown, which commensurately reduces the amount of cooling required to prevent the spent fuel from heating up to a temperatures that could compromise the ability of the fuel cladding to retain fission products.

Along with the reduction in risk, the consequences of a release decline after a reactor permanently shuts down and defuels. The short-lived radionuclides contained in the spent fuel, particularly volatile components such as iodine and noble gases, decay away, thereby reducing the inventory of radioactive materials that are readily dispersible and transportable in air.

[3] Nothing in the documentation of the rulemakings for onsite or offsite insurance coverage discusses the applicability or appropriateness of these regulations for facilities in the permanently shutdown and defueled condition. The subject regulations were established for power operations because such conditions create the potential for an accident with potentially significant on and offsite consequences. Because experience has shown that there are significant costs incurred in complying with the requirements of requirements of 10 CFR 50.54(w) and 10 CFR 140.11 and because such costs are ultimately borne by the general public as ratepayers, a demonstration by the licensee that the basis for the full insurance requirements no longer exists will be sufficient grounds for concluding, *prima facie*, that compliance would result in "costs that are significantly in excess of those contemplated when the regulation was adopted," i.e., 10 CFR 50.12(a)(2)(iii), and there is present "other material circumstance not considered when the regulation was adopted for which it would be in the public interest to grant an exemption," i.e., 10 CFR 50.12(a)(2)(iv).

Therefore, the consequences of beyond design basis events for permanently shutdown and defueled plants are dominated by long-lived isotopes which predominantly remain trapped within both the fuel cladding and fuel matrix itself and are neither readily dispersible nor transportable thus significantly limiting any potential for off-site consequences.

For the purpose of modifying the amount of insurance coverage maintained for a permanently shutdown and defueled facility, the cost of recovery from potential accident scenarios must be evaluated. These accident scenarios are limited. The most significant would be those potential radiological consequences that could be associated with the onsite storage of the spent fuel in the spent fuel pool (SFP). In addition, a site may contain a radioactive inventory of liquid waste, activated reactor components, and contaminated structural materials.

Insurance Requirements

On-site insurance coverage and the off-site financial protection on a generic basis can be significantly reduced for permanently shutdown (PSD) reactors. Adjusted coverage levels consider credible accidents involving the spent fuel pool and the amount of liquid radioactive waste stored onsite in post-shutdown modes. The insurance coverage requirements are based on the estimated costs resulting from two configurations as described below and address consideration of on-site costs of recovery (i.e., stabilization and decontamination).

Reactor Configuration 1: Spent Fuel in Pool

In reactor Configuration 1, the reactor is defueled and permanently shutdown, all spent fuel is in the spent fuel pool and the fuel is presumed to have undergone a modest level of decay (60-90 days). During reactor Configuration 1, licensees would be required to maintain off-site insurance coverage in the amount of $10 million, however, the secondary financial protection layer would no longer be required. This coverage is conservative in that there are no credible events that pose offsite consequences once a plant is permanently shutdown.

On-site insurance requirements are based on the assumption of a spill of slightly contaminated liquid from the largest on-site storage tank, an assumed 100,000 gallons.

Onsite insurance would be required in the amount of $25 million.

Reactor Configuration 2: Spent Fuel Offsite or in Onsite ISFSI

In reactor Configuration 2, the reactor is defueled and permanently shutdown, and spent fuel is in no longer in the spent fuel pool having been transferred offsite or to an onsite ISFSI. Offsite insurance coverage could be reduced to zero.

The only plausible event of any significance for this configuration continues to be a postulated spill of slightly contaminated liquid. Onsite insurance requirements would be continue to be required in the amount of $25 million until the site has less than 1000 gallons of liquid material onsite, at which time the onsite coverage may be reduced to zero.

The licensee will be permitted further exemptions from the insurance requirements specified above provided there is sufficient justification. This justification could consider the following plant specific considerations:

- Population density
- Barriers to radionuclide release from the site
- Quantity, radionuclide composition and concentration, and storage container integrity, and other barriers to release for slightly contaminated liquids stored on site.

The specific accidents considered would be limited to those indicated above, namely, a fuel handling incident, a spent fuel cask drop, and accidents associated with radioactive waste storage or processing; or any design basis accident which was included in the licensee's original licensing basis for the operating facility and which is still valid for the facility in the defueled condition.

The licensee should be allowed to reduce insurance coverage consistent with the reduction in consequences and risk presented by the existing plant configuration. The configurations describe above with their associated reductions in required insurance coverage can be used as guidance in the evaluation of plant specific reductions. Further reductions in requirements could result from plant specific analysis of the risk and consequence posed by mobile sources of radioactivity on site.

III. REVIEW PROCEDURES

The review consists of the Project Manager's evaluation of the information submitted by the applicant/licensee using the foregoing Acceptance Criteria. The bulk of this information should be found in the applicant's the Defueled Safety Analysis Report and the licensee exemption request. This information may be supplemented by a personal visit to the site by the reviewer and meetings with the applicant.

The design basis accidents requiring evaluation for a defueled facility are a small subset of those considered for an operating facility and are limited to: a fuel handling incident, a spent fuel cask drop, accidents associated with radioactive waste storage or processing, and any design basis accident which was included in the licensee's original licensing basis for the operating facility and which is still valid for the facility in the defueled condition.

There are no other design basis or reasonably credible beyond design basis events which would create significant on-site or off-site consequences and which would therefore need to be evaluated in order to grant the exemption request.

If a specific accident analysis has been previously found to be acceptable to the Commission and there has been no material change in the key parameters and assumptions which served as the basis for the original judgment, the original conclusion of acceptability will remain valid.

The reviewer must determine whether or not the acceptance criteria identified in Section II have been satisfied. Any deficiencies should be clearly identified and should form the basis for a request for additional information to the applicant. If any deficiencies remain at the conclusion of the review, they must be identified in the Safety Evaluation Report and subsequently resolved with participation of higher level NRC management.

It should be recognized that the detailed application of the acceptance criteria will in some instances require the exercise of judgement on the part of the reviewer. The reviewer is expected to achieve a safety finding based on a traditional risk-informed reasonableness threshold which treats the acceptance criteria as akin to an "adequate protection" standard rather than seeking to impose a standard of absolute safety.

IV. EVALUATION FINDINGS

The desired evaluation findings should be substantially equivalent to the following statement:

Reactor Configuration 1

The staff has completed its review of your request to reduce your insurance coverage, and approves the reduction to $25 million for onsite property coverage and $10 million for offsite liability coverage, and withdrawal from participation in the secondary financial protection layer provided under the Price-Anderson Act.

These reductions in required levels of insurance protection are based on the reduction in potential risk to public health due to the permanently shutdown and defueled status of the plant and the limited risk from rupture of a large (i.e., greater than 1000 gallons) slightly-contaminated storage tank.

The onsite insurance coverage may be reduced to zero when the site has less than 1000 gallons of slightly-contaminated liquid stored on site.

Reactor Configuration 2

The staff has completed its review of your request to reduce your insurance coverage, and approves the reduction to $25 million for onsite property coverage until such time that the site has less that 1000 gallons of liquid material on site at which time the onsite insurance coverage may be reduced to zero.

Offsite coverage may be immediately reduced to zero along withdrawal from participation in the secondary financial protection layer provided under the Price-Anderson Act.

These reductions in required levels of insurance protection are based on the reduction in potential risk to public health and due to the permanently shutdown and defueled status of the plant, the relocation of the spent fuel offsite or onsite to a dry storage ISFSI and the limited risk from rupture of a large slightly-contaminated storage tank.

Conclusion

The Commission has completed its review of the licensee's request for an exemption from certain requirements of certain requirements of 10 CFR 50.54(w) and 10 CFR 140.11 and concludes that the request is acceptable in view of the greatly reduced onsite and offsite radiological consequences from any reasonably conceivable accident which could occur with the plant in its present status.

The Commission, based on its independent evaluation, agrees with the licensee's analyses and concludes that sufficient bases have been presented for approval of the exemption request. The Commission has determined that pursuant to 10 CFR 50.12(a)(1), this exemption is authorized by law, will not present an undue risk to the public health and safety, and is consistent with the common defense and security. Furthermore, the Commission finds that there are special circumstances presented that satisfy the requirements of 10 CFR 50.12(a)(2)(ii), (iii), and (vi). The Commission hereby grants the requested exemption.

V. IMPLEMENTATION

The following is intended to provide guidance to licensees regarding the Commission's plan for using this SRP.

Except in those cases in which a licensee proposes an acceptable alternative method, the methods described herein will be used by the Commission in its evaluation of requests by licensees of facilities that are permanently shutdown and defueled for exemption from certain requirements for onsite property coverage and offsite liability coverage, and withdrawal from participation in the secondary financial protection layer provided under the Price-Anderson Act.

APPENDIX I

SELECTED EXCERPTS FROM THE REGULATIONS PERTAINING TO ON-SITE AND OFF-SITE INSURANCE PROTECTION REQUIREMENTS

§50.54 Condition of Licenses

(w) Each power reactor licensee under this part for a production or utilization facility of the type described in §50.21(b) or 50.22 shall take reasonable steps to obtain insurance available at reasonable costs and on reasonable terms from private sources or to demonstrate to the satisfaction of the NRC that it possesses an equivalent amount of protection covering the licensee's obligation, in the event of an accident at the licensee's reactor, to stabilize and decontaminate the reactor and the reactor station site at which the reactor experiencing the accident is located, provided that:

(1) The insurance required by paragraph (w) of this section must have a minimum coverage limit for each reactor station site of either $1.06 billion or whatever amount of insurance is generally available from private sources, whichever is less. The required insurance must clearly state that, as and to the extent provided in paragraph (w)(4) of this section, any proceeds must be payable first for stabilization of the reactor and next for decontamination of the reactor and the reactor station site. If a licensee's coverage falls below the required minimum, the licensee shall within 60 days take all reasonable steps to restore its coverage to the required minimum. The required insurance may, at the option of the licensee, be included within policies that also provide coverage for other risks, including, but not limited to, the risk of direct physical damage.

(2)(i) With respect to policies issued or annually renewed on or after April 2, 1991, the proceeds of such required insurance must be dedicated, as and to the extent provided in this paragraph, to reimbursement or payment on behalf of the insured of reasonable expenses incurred or estimated to be incurred by the licensee in taking action to fulfill the licensee's obligation, in the event of an accident at the licensee's reactor, to ensure that the reactor is in, or is returned to, and maintained in, a safe and stable condition and that radioactive contamination is removed or controlled such that personnel exposures are consistent with the occupational exposure limits in 10 CFR part 20. These actions must be consistent with any other obligation the licensee may have under this chapter and must be subject to paragraph (w)(4) of this section. As used in this section, an "accident" means an event that involves the release of radioactive material from its intended place of confinement within the reactor or on the reactor station site such that there is a present danger of release off site in amounts that would pose a threat to the public health and safety.

(ii) The stabilization and decontamination requirements set forth in paragraph (w)(4) of this section must apply uniformly to all insurance policies required under paragraph (w) of this section.

(3) The licensee shall report to the NRC on April 1 of each year the current levels of this insurance or financial security it maintains and the sources of this insurance or financial security.

(4)(i) In the event of an accident at the licensee's reactor, whenever the estimated costs of stabilizing the licensed reactor and of decontaminating the reactor and the reactor station site exceed $100 million, the proceeds of the insurance required by paragraph (w) of this section must be dedicated to and used, first, to ensure that the licensed reactor is in, or is returned to, and can be maintained in, a safe and stable condition so as to prevent any significant risk to the public health and safety and, second, to decontaminate the reactor and the reactor station site in accordance with the licensee's cleanup plan as approved by order of the Director of the Office of Nuclear Reactor Regulation. This priority on insurance proceeds must remain in effect for 60 days or, upon order of the Director, for such longer periods, in increments not to exceed 60 days except as provided for activities under the cleanup plan required in paragraphs (w)(4)(iii) and (w)(4)(iv) of this section, as the Director may find necessary to protect the public health and safety. Actions needed to bring the reactor to and maintain the reactor in a safe and stable condition may include one or more of the following, as appropriate: (A) Shutdown of the reactor; (B) Establishment and maintenance of long-term cooling with stable decay heat removal; Maintenance of sub-criticality; (D) Control of radioactive releases; and (E) Securing of structures, systems, or components to minimize radiation exposure to onsite personnel or to the offsite public or to facilitate later decontamination or both.

(ii) The licensee shall inform the Director of the Office of Nuclear Reactor Regulation in writing when the reactor is and can be maintained in a safe and stable condition so as to prevent any significant risk to the public health and safety. Within 30 days after the licensee informs the Director that the reactor is in this condition, or at such earlier time as the licensee may elect or the Director may for good cause direct, the licensee shall prepare and submit a cleanup plan for the Director's approval. The cleanup plan must identify and contain an estimate of the cost of each cleanup operation that will be required to decontaminate the reactor sufficiently to permit the licensee either to resume operation of the reactor or to apply to the Commission under §50.82 for authority to decommission the reactor and to surrender the license voluntarily. Cleanup operations may include one or more of the following, as appropriate: (A) Processing any contaminated water generated by the accident and by decontamination operations to remove radioactive materials; (B) Decontamination of surfaces inside the auxiliary and fuel-handling buildings and the reactor building to levels consistent with the Commission's occupational exposure limits in 10 CFR part 20, and decontamination or disposal of equipment; Decontamination or removal and disposal of internal parts and damaged fuel from the reactor vessel; and (D) Cleanup of the reactor coolant system.

(iii) Following review of the licensee's cleanup plan, the Director will order the licensee to complete all operations that the Director finds are necessary to decontaminate the reactor sufficiently to permit the licensee either to resume operation of the reactor or to apply to the Commission under 150.82 for authority to decommission the reactor and to surrender the license voluntarily. The Director shall approve or disapprove, in whole or in part for stated reasons, the licensee's estimate of cleanup costs for such operations. Such order may not be effective for more than 1 year, at which time it may be renewed. Each subsequent renewal order, if imposed, may be effective for not more than 6 months.

(iv) Of the balance of the proceeds of the required insurance not already expended to place the reactor in a safe and stable condition pursuant to paragraph (w)(2)(I) of this section, an amount sufficient to cover the expenses of completion of those decontamination operations that are the subject of the Director's order shall be dedicated to such use, provided that, upon certification to the Director of the amounts expended previously and from time to time for stabilization and decontamination and upon further certification to the Director as to the sufficiency of the dedicated amount remaining, policies of insurance may provide for payment to the licensee or other loss payees of amounts not so dedicated, and the licensee may proceed to use in parallel (and not in preference thereto) any insurance proceeds not so dedicated for other purposes.

§140.7 Fees

(a) Each reactor licensee shall pay a fee to the Commission based on the following schedule:

(1) For indemnification from $500 million to $400 million inclusive, a fee of $30 per year per thousand kilowatts of thermal capacity authorized in the license;

(2) For indemnification from $399 million to $300 million inclusive, a fee of $24 per year per thousand kilowatts of thermal capacity authorized in the license;

(3) For indemnification from $299 million to $200 million inclusive, a fee of $18 per year per thousand kilowatts of thermal capacity authorized in the license;

(4) For indemnification from $199 million to $100 million inclusive, a fee of $12 per year per thousand kilowatts of thermal capacity authorized in the license; and

(5) For indemnification from $99 million to $1 million inclusive, a fee of $6 per year per thousand kilowatts of thermal capacity authorized in the license.

Provided, *however*, That no fee shall be less than $100 per annum for any nuclear reactor. This fee is for the period beginning with the date on which the applicable indemnity agreement is effective. The various levels of indemnity fees are set forth in the schedule in this paragraph. The amount of indemnification for determining indemnity fees will be computed by subtracting from the statutory limit of liability the amount of financial protection required of the licensee. In the case of licensees subject to the provision of §140.11(a)(4), this total amount will be the amount, as determined by the Commission, of the financial protection available to licensees at the close of the calendar year preceding the one in which the fee becomes due. For those instances in which a certified financial statement is provided as a guarantee of payment of deferred premiums in accordance with §140.21(e), a fee of $1,000 or the indemnity fee, whichever is greater, is required.

§140.11 Amounts of financial protection for certain reactors

(a) Each licensee is required to have and maintain financial protection:

(1) In the amount of $1,000,000 for each nuclear reactor he is authorized to operate at a thermal power level not exceeding ten kilowatts;

(2) In the amount of $1,500,000 for each nuclear reactor he is authorized to operate at a thermal power level in excess of ten kilowatts but not in excess of one megawatt;

(3) In the amount of $2,500,000 for each nuclear reactor other than a testing reactor or a reactor licensed under section 104b of the Act which he is authorized to operate at a thermal power level exceeding one megawatt but not in excess of ten megawatts; and

(4) In an amount equal to the sum of $200,000,000 and the amount available as secondary financial protection (in the form of private liability insurance available under an industry retrospective rating plan providing for deferred premium charges equal to the pro rata share of the aggregate public liability claims and costs, excluding costs payment of which is not authorized by subsection 170o.(1)(D) of the Act, in excess of that covered by primary financial protection) for each nuclear reactor which is licensed to operate and which is designed for the production of electrical energy and has a rated capacity of 100,000 electrical kilowatts or more: Provided, however, that under such a plan for deferred premium charges for each nuclear reactor which is licensed to operate, no more than $83,900,000 with respect to any nuclear incident (plus any surcharge assessed under subsection 170o.(1)(E) of the Act) and no more than $10,000,000 per incident within one calendar year shall be charged.

(b) In any case where a person is authorized pursuant to part 50 of this chapter to operate two or more nuclear reactors at the same location, the total primary financial protection required of the licensee for all such reactors is the highest amount which would otherwise be required for any one of those reactors: *Provided*, That such primary financial protection covers all reactors at the location.

[25 FR 2944, Apr. 7, 1960, as amended at 34 FR 706, Jan. 17, 1969; 37 FR 3423, Feb. 16, 1972; 39 FR 5310, Feb. 12, 1974; 40 FR 7082, Feb. 19, 1975; 42 FR 49, Jan. 3, 1977; 42 FR 20140, Apr. 18, 1977; 44 FR 20632, Apr. 6, 1979; 54 FR 24158, June 6, 1989; 58 FR 42852, Aug. 12, 1993]

§140.12 Amount of financial protection required for other reactors

(a) Each licensee is required to have and maintain financial protection for each nuclear reactor for which the amount of financial protection is not determined in §140.11, in an amount determined pursuant to the formula and other provisions of this section: *Provided*, That in no event shall the amount of financial protection required for any nuclear reactor under this section be less than $4,500,000 or more than $74,000,000.

(b)(1) The formula is:

x=B times P.

(2) In the formula:

x=Amount of financial protection in dollars.

B=Base amount of financial protection.

P=Population factor.

(3) The base amount of financial protection is equal to $185 times the maximum power level, expressed in thermal kilowatts, as authorized by the applicable license.

(4) The population factor (P) shall be determined as follows:

(I) *Step 1*. The area to be considered includes all minor civil divisions (as shown in the 1950 Census of Population, Bureau of the Census, or later data available from the Bureau) which are wholly or partly within a circle with the facility at its center and having a radius in miles equal to the square root of the maximum authorized power level in thermal megawatts.

(ii) *Step 2*. Identify all minor civil divisions according to the same census which are in whole or in part within the circle determined in Step 1. Determine the population of each such minor civil division (according to the same census or later data available from the Bureau of the Census). For each minor civil division, divide its population by the square of the estimated distance to the nearest mile from the reactor to the geographic center of the minor civil division: *Provided*, That no such distance shall be deemed to be less than one mile. If the sum of the quotients thus obtained for all minor civil divisions wholly or partly within the circle is 1,000 or less, the population factor is 1. If the sum of these quotients is more than 1,000 but not more than 3,000, the population factor is 1.2. If the sum of these quotients is more than 3,000 but not more than 5,000, the population factor is 1.4. If the sum of these quotients is more than 5,000 but not more than 7,000, the population factor is 1.6. If the sum of these quotients is more than 7,000 but not more than 9,000, the population factor is 1.8. If the sum of these quotients is more than 9,000 the population factor is 2.0.

(c) In any case where a person is authorized pursuant to part 50 of this chapter to operate two or more nuclear reactors at the same location, the total financial protection required of the licensee for all such reactors is the highest amount which would otherwise be required for any one of those reactors: *Provided*, That such financial protection covers all reactors at the location.

(d) Except in cases where the amount of financial protection calculated under this section is a multiple of $100,000, amounts determined pursuant to this section shall be adjusted to the next highest multiple of $100,000.

[25 FR 2944, Apr. 7, 1960, as amended at 26 FR 1397, Feb. 17, 1961; 32 FR 8125, June 7, 1967]

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DECOMMISSIONING STANDARD REVIEW PLAN: FACILITY PERSONNEL TRAINING REQUIREMENTS FOR PERMANENTLY DEFUELED FACILITIES

Review Responsibilities

Primary – Applicant's Project Manager

Secondary - None

I. AREAS OF REVIEW

A facility in a permanently shutdown and defueled condition poses a significantly reduced risk to the public health and safety. Furthermore, there are a relatively small number of complex activities required to be conducted at a permanently defueled nuclear power plant as compared to those required at a plant license for power operations. After "Certifications of Permanent Cessation of Operation and Removal of Fuel from the Reactor Vessel" have been docketed in accordance with 10 CFR 50.82(a)(1) and, in view of the reduced risk, it is anticipated that the licensee of such a facility will seek revision to plant technical specifications which are no longer appropriate and may also seek exemption from certain requirements of the regulations, principally 10 CFR 50.54 and 10 CFR 50.120.[1]

This review is expected to provide the basis for evaluating proposed changes to the technical specifications that pertain to the requirements for Licensed Operators, operations staffing levels, and training requirements. In addition, this review provides the basis for evaluating exemptions (including need for exemption) from certain requirements of Section 50.54 which restricts the conduct of certain activities to Licensed Operators or Senior Licensed Operators, and from Section 50.120 which requires the maintenance of a Systems Approach to Training for the training and qualification of nuclear power plant personnel as further described in 10 CFR 55.4.

The review is performed by the licensee's Project Manager. The review involves evaluation of the licensee's Defueled Safety Analysis Report (DSAR), proposed revisions to the technical specifications, the Certified Fuel Handler and Training and Retraining Program, and any other

[1] The subject regulations were established for power operations.

relevant information provided in the exemption request. Acceptance criteria are provided in this SRP.[2]

The subject review will be performed consistent with the requirements of the Commission's backfitting rule, 10 CFR 50.109. The policy underlying the rule, i.e., to ensure that new requirements are properly justified from a safety and cost-benefit standpoint, continues to apply during the decommissioning process so long as a facility's 10 CFR Part 50 license remains in effect. Decommissioning facilities are entitled to the same predictability, stability and protection from arbitrary actions as operating facilities.

II. ACCEPTANCE CRITERIA

Exemption Requests under 10 CFR 50.12(a)

Pursuant to 10 CFR 50.12(a), the Commission may, upon application by any interested person or upon its own initiative, grant exemptions from the requirements of the regulations which are:

1. authorized by law,

2. will not present an undue risk to public health and safety, and

3. are consistent with the common defense and security, and

4. present special circumstances.

Special circumstances exist when:

1. application of the regulation in the particular circumstance would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule (10 CFR 50.12(a)(2)(ii)); or

2. compliance would result in undue hardship; or other costs that are significantly in excess of those contemplated when the regulation was adopted, or that are significantly in excess of those incurred by others similarly situated (10 CFR 50.12(a)(2)(iii)); or

3. there is present other material circumstances not considered when the regulation was adopted for which it would be in the public interest to grant an exemption (10 CFR 50.12(a)(2)(vi)).

Underlying Purpose of 10 CFR 50.54 and 10 CFR 120

The underlying purpose of 10 CFR 50.54 is to specify the conditions the NRC attaches to facility operating licenses. Certain of these conditions restrict the conduct of specific activities to Licensed Operators or Senior Licensed Operators. These conditions are described below:

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[2] If certain requirements are retained in the technical specifications and training/retraining programs there is no need for an exemption from the requirements of 10 CFR 50.54 and 10 CFR 50.120.

1. 10 CFR 50.54(i) prohibits manipulation of the controls of a nuclear power plant by anyone who is not a Licensed Operator. Controls are defined in 10 CFR 55.4 as apparatus and mechanisms the manipulation of which directly affects the reactivity or power level of the reactor.

2. 10 CFR 50.54(j) requires that manipulation of apparatus of mechanisms other than controls, which may affect reactivity or power level be done with the knowledge and consent of a Licensed or Senior Operator who is present at the controls.

3. 10 CFR 50.54(k) requires a Licensed Operator or Senior Licensed Operator to be present at the controls at all time during operation of the facility.

4. 10 CFR 50.54(l) requires licensees to designate individuals to be responsible for directing the licensed activities of licensed operators and that these individuals shall be Senior Licensed Operators.

5. 10 CFR 50.54(m)(1) requires a Senior Licensed Operator to be present at the facility or readily available on call at all times during its operation. It also requires a Senior Licensed Operator to be present during certain evolutions or as otherwise prescribed in the facility license.

6. 10 CFR 50.54 (m)(2) specifies the minimum Licensed Operator staffing levels for nuclear power plant licenses.

The underlying purpose of 10 CFR 50.120 is to assure other key personnel are qualified to operate and maintain the nuclear facility in a safe manner in all modes of operation by requiring a Systematic Approach to Training, as defined in 10 CFR 55.4.[3] Sufficient records must be maintained by the licensee to maintain program integrity and be kept available for NRC inspection to verify the adequacy of the program, but NRC approval of the training program is not required. Nine categories of nuclear power personnel are to be included in these training programs:

1. Non-licensed Operator

2. Shift Supervisor

3. Shift Technical Advisor

4. Instrument and Control Technician

5. Electrical Maintenance Personnel

6. Mechanical Maintenance Personnel

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[3] The five key elements for the Systems Approach to Training include (1) analysis of job performance requirements and training needs, (2) derivation of learning objectives based upon this analysis, (3) design and implementation of the training program based upon learning objectives, (4) trainee evaluation, and (5) program evaluation and revision.

7. Radiological Protection Personnel

8. Chemistry Technician

9. Engineering Support Personnel

On July 29, 1996, a final rule amending the regulations on decommissioning procedures was published in the Federal Register (61 FR 39278). The final rule amended 10 CFR Part 2, "Rules of Practice for Domestic Licensing Proceedings and Issuance of Orders," Part 50, "Domestic Production and Utilization Facilities," and Part 51, "Environmental Protection for Regulations for domestic Licensing and Related Regulatory Functions." This rule clarified the regulation for decommissioning nuclear power facilities. The final rule became effective on August 28, 1996.

The amendments to 10 CFR Parts 2 and 51 pertain to the activities that the licensee undertakes to terminate the license. The changes to 10 CFR 50 serve to explicitly or implicitly modify several of the above cited requirements of 10 CFR 50.54 and 10 CFR 50.120 for a permanently shutdown and defueled facility. The following are explicit modifications.

Certified Fuel Handler

An addition is made to the list of definitions in 10 CFR 50.2, that of a Certified Fuel Handler. The Certified Fuel Handler is defined as a non-licensed operator who has qualified in accordance with a fuel handler training program approved by the Commission. This classification of non-licensed operator was previously undefined.

While the requirements for licensing of a Reactor Operator and Senior Reactor Operator are detailed in 10 CFR 55 "Operators Licenses," the required training for the Certified Fuel Handler is not explicitly described in the regulations. However, the fuel handler training program must be reviewed and approved by the Commission just as for the Reactor Operator and Senior Reactor Operator.

10 CFR 50.54

10 CFR 50.54(y) is modified to include the Certified Fuel Handler and now reads "Licensee action permitted by paragraph (x) of this section[4] shall be approved , as a minimum, by a licensed senior operator, or, at a nuclear power facility for which the certifications required under Section 50.82(a)(1) have been submitted, by either a licensed senior operator or a certified fuel handler, prior to taking the action."

The effect of these explicit modifications to the rules is that, upon:

1. submittal of certifications required under Section 50.82(a)(1), and

2. NRC approval of the certified fuel handler training program, and

[4] 10 CFR 50.54(x) permits deviation from the license conditions or technical specifications under exigent circumstances.

3. appropriate revisions to the Technical Specifications to define the role of the Certified Fuel Handler (CFH),

the CFH in a permanently shutdown and defueled plant is granted the responsibility and authority equivalent to that of a Senior Reactor Operator in an operating power plant.

Therefore, the licensee that has complied with the specified requirements 1, 2, and 3, will remain in compliance with 10 CFR 50.54.

10 CFR 120

While there were numerous changes to decommissioning related rules in 1996, no explicit changes were made to 10 CFR 120. However, current licensing practice is to reduce the burden of training required for plant personnel in a permanently shutdown and defueled facility in recognition of the associated reduction in number of complex activities and tasks.

For a permanently shutdown and defueled facility, the qualifications and training program for the Certified Fuel Handler and for each personnel category included in 10 CFR 120 should be substantially equivalent to the following:

<u>Certified Fuel Handler</u>

The Certified Fuel Handler training and Retraining Program must be consistent with current licensing practice and must provide adequate confidence that appropriate SAT-based training of personnel who perform certified fuel handler duties is conducted to ensure the facility is maintained in a safe and stable condition.

<u>Non-Licensed Operator</u>

A Non-Licensed Operator, or Non-Certified Operator (NCO), refers to personnel other than a Certified Fuel Handler. The NCOs may be qualified to stand watch in the Control Room. Minimum qualifications are established in Regulatory Guide 1.9 - September 1975. Continuing training for NCO•s should address facility changes, procedure changes and improvement areas as necessary based upon the assigned duties.

<u>Shift Supervisor</u>

Shift Supervisors should be CFH qualified and should participate in the CFH Retraining Program. In addition it is desirable that supervisory skills training is provided as part of the position qualification process.

<u>Shift Technical Advisor (STA)</u>

This position may be eliminated along with the associated training program and associated qualification and continuing training requirements upon NRC approved modification of the Technical Specifications.

Instrument and Controls (I&C) Technician, Electrical Maintenance Personnel, Mechanical Maintenance Personnel, Radiological Protection (RP) Technician and Chemistry Technician

The minimum qualifications for Technicians, including I&C Technicians, Electrical Maintenance Personnel, Mechanical Maintenance Personnel, RP Technician and Chemistry Technicians is (1) two years experience applicable to their specialty, or (2) two years of related academic training in engineering or science and one year of applicable experience to their specialty. The training program for each of these disciplines should include specialty areas and on-the-job training commensurate with the tasks to be performed. The continuing training program should include facility changes, procedure revisions, and improvement areas. Provision for additional individualized continued training should be provided as a the need arises.

Engineering Support Personnel

Each member of the engineering support personnel staff should meet or exceed the qualifications stated in Regulatory Guide 1.8 - September 1975 for comparable positions. The training program should provide for continuing training on an as-needed basis for the appropriate engineering functions.

Facility Staff Qualifications

Each member of the facility support staff should meet or exceed the qualifications stated in Regulatory Guide 1.8 - September 1975 for comparable positions. The training program should provide for continuing training on an as-needed basis.

All of the above training programs should meet the intent of 10 CFR 55.4 and should consider: (1) analysis of job performance requirements and training needs, (2) derivation of learning objectives based upon this analysis, (3) design and implementation of the training program based upon learning objectives, (4) trainee evaluation, and (5) program evaluation and revision. However, the scope and depth of the training programs should be commensurate with the significant reduction in risk and the reduction in the number of complex activities associated with a permanently shutdown and defueled facility.

III. REVIEW PROCEDURES

The review consists of the Project Manager’s evaluation of the information submitted by the applicant/licensee using the foregoing Acceptance Criteria. The bulk of this information should be found in the applicant’s the Defueled Safety Analysis Report, proposed revisions to the technical specifications, the Certified Fuel Handler and Training and Retraining Program, and any other relevant information provided in the licensee request. Acceptance criteria are provided in Section II of this SRP. This information may be supplemented by a personal visit to the site by the reviewer and meetings with the applicant.

The reviewer must determine whether or not the acceptance criteria identified in Section II have been satisfied. Any deficiencies should be clearly identified and should form the basis for a request for additional information to the applicant. If any deficiencies remain at the conclusion of the review, they must be identified in the Safety Evaluation Report and subsequently resolved with participation of higher level NRC management.

It should be recognized that the detailed application of the acceptance criteria will in some instances require the exercise of judgement on the part of the reviewer. The reviewer is expected to achieve a safety finding based on a traditional risk-informed reasonableness threshold which treats the acceptance criteria as akin to an “adequate protection” standard rather than seeking to impose a standard of absolute safety.

IV. EVALUATION FINDINGS

The desired evaluation findings should address both the proposed amendment and the associated exemption request (if any) and, should be substantially equivalent to the following:

The proposed amendment is designed to eliminate the requirements for licensed operators and a licensed operator retraining program and to replace those positions and programs with certified fuel handlers. A certified fuel handler training and retraining program has been established and is, hereby, accepted by the Commission. This ensures that the qualifications of the operations personnel are commensurate with the tasks to be performed and the conditions requiring a response.

Section 50.120 of Title 10 of the Code of Federal Regulations, Training and Qualification of Nuclear Power Plant Personnel, requires training programs to be established, implemented, maintained, and derived using a systems approach to training (SAT) as defined in 10 CFR 55.4. In its submittal the licensee has described the systematic approach taken to develop, implement, and maintain the necessary training and retraining programs. The approach as describe by the licensee contains five key elements from 10 CFR 55.4 and is intended to provide a training system that will ensure successful performance on the job by trained an qualified personnel. The five key elements are (1) analysis of job performance requirements and training needs, (2) derivation of learning objectives based upon this analysis, (3) design and implementation of the training program based upon learning objectives, (4) trainee evaluation, and (5) program evaluation and revision.

The proposed amendment to the Technical Specifications, the Certified Fuel Handler Training and Retraining Program, and the training program for each personnel category in 10 CFR 120 are consistent with current licensing practice and provide adequate confidence that appropriate SAT-based training of personnel who perform facility duties is conducted to in a manner to assure maintenance of a safe and stable condition. Accordingly, the proposed amendment to the Technical Specifications, the Certified Fuel Handler Training and Retraining Program, and the training program for other facility personnel are found to be acceptable.

Based upon our evaluation and approval of the licensee proposed amendment to the Technical Specifications and our evaluation and approval of the Certified Fuel Handler Training and Retraining Program, and the consideration of the permanently shutdown and defueled status of the plant, the licensee’s programs remain in compliance with the requirements of 10 CFR 50.54 and 10 CFR 50.120 and exemptions are not required to these regulations.

Conclusion

The Commission has completed its review of the licensee's request for modifications to its license requirements as a result of the plant being place in a permanently shutdown and defueled condition. Based upon our evaluation and approval of the licensee proposed amendment to the Technical Specifications and our evaluation and approval of the Certified Fuel Handler Training and Retraining Program, and the consideration of the permanently shutdown and defueled status of the plant, the licensee's programs remain in compliance with the requirements of 10 CFR 50.54 and 10 CFR 50.120 and exemptions are not required to these regulations.

This is based upon the conclusion that the proposed amendment to the Technical Specifications and the Certified Fuel Handler Training and Retraining Program are consistent with current licensing practice and provide adequate confidence that appropriate SAT-based training of personnel who perform certified fuel handler duties is conducted to ensure the facility is maintained in a safe and stable condition.

Furthermore, the Commission has concluded, based upon the considerations discussed above that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commissions's regulations, and) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.