UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

May 28, 1999

ORGANIZATION Nuclear Energy Institute

SUBJECT: SUMMARY OF MEETING WITH THE NUCLEAR ENERGY INSTITUTE AND PUBLIC STAKEHOLDERS

On May 5, 1999, the NRC staff met with representatives of the Nuclear Energy Institute (NEI) and public stakeholders to discuss the status of the staff's efforts to develop a risk-informed approach to certain regulations affecting reactor decommissioning. The agenda and attendance list are provided as Enclosures 1 and 2, respectively.

Bill Huffman, of the NRC staff, began the meeting by summarizing NRC conceptual plans for a comprehensive review of all decommissioning regulations with the goal of consolidating as many regulations as possible within a single location in Title 10 of the *Code of Federal Regulations*. The staff believes that such activities will help clarify the applicability of Part 50 and other regulations to permanently shut down power reactors. In addition, the regulatory consolidation effort should help identify those decommissioning regulations where risk information could be used to reduce any unnecessary regulatory burden. Any rule changes undertaken by the NRC in these areas will be accomplished through the normal rulemaking process and will include ample opportunity for public and industry comments. Slides used by Mr. Huffman in his presentation are provided in Enclosure 3.

Responses from NEI representatives and members of the public were supportive of the NRC initiative described by Mr. Huffman.

Next, Gary Holahan, the Director of the Division of Systems Safety and Analysis, summarized the status of ongoing NRC efforts to establish a risk-informed technical basis for reviewing exemption requests and initiating rulemaking related to emergency preparedness, safeguards, and insurance at permanently shut down nuclear power plants. Slides used by Mr. Holahan, including an outline of the areas and issues being evaluated by the technical working group, are provided in Enclosure 4.

NEI representatives commented that the review undertaken by the NRC addressed too many issues and could not be accomplished within the scheduled time frame. NEI also commented that the effort was still largely of a deterministic nature and appeared to hold decommissioning to a "zero-risk" basis regarding spent fuel pool accidents. NEI stated that the NRC should instead focus on reducing the uncertainties of beyond-design-basis seismic events and their possible contribution to risk associated with spent fuel pool accidents.

Mike Meisner, speaking for the Maine Yankee Atomic Power Company, provided a handout with detailed technical information regarding the Maine Yankee plant (Enclosure 5) in response to the NRC's April 13, 1999, solicitation for additional information on spent fuel pool risk at permanently shutdown plants.

Peter James Atherton stated that although he agreed with the work effort outlined by Mr. Holahan, he was still concerned that the NRC might not maintain a defense-in-depth approach to decommissioning reactor spent fuel pool accidents. He also stated his concern with aging of spent fuel pools and equipment beyond the original 40-year licensed lifetime of the plants.

Ray Shadis (Friends of the Coast) was concerned about inadvertent draining of spent fuel pools through existing piping connections. In response to this concern it was noted that spent fuel pool piping systems are typically designed so that all connections to the pool are made at elevations considerably higher than the top of the fuel. In such cases, operator error can drain the pool only to the level of the connection; many feet of water would remain above the top of the fuel. Mr. Shadis also stated that he was concerned with potential safeguards vulnerabilities at decommissioning reactors and that the NRC should ensure that safeguards and security issues were thoroughly considered by the ongoing effort.

ORIGINAL SIGNED BY:
Richard F. Dudley, Senior Project Manager
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Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Project No. 689

Enclosures:
1. Agenda
2. List of Attendees
3. Set of Slides used by W. Huffman
4. Set of Slides used by G. Holahan
5. Solicitation for Additional Information

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Risk-Informing Decommissioning Rules
NRC/NEI/Public Meeting
May 5, 1999

AGENDA

1. Decommissioning Regulatory Improvement
   Bill Huffman, DLPM

2. Status of Technical Working Group Efforts to Establish Decommissioning Technical Basis
   Gary Holahan, DSSA

3. NEI Presentation

4. Public Questions and Comments

Enclosure 1

ATTENDEES
Decommissioning Public Meeting
May 5, 1999

NAME	ORGANIZATION
Chet Poslusny	NRR/NMSS
Richard Dudley	NRR/DLPM
Bill Huffman	NRR/DLPM
Stuart Richards	NRR/DLPM
Anthony Markley	NRR/DRIP
Suzanne Black	NRR/DLPM
John A. Zwolinski	NRR/DLPM
Gary M. Holahan	NRR/DSSA
Sam Nalluswami	NRR/DLPM
Stewart Brown	NMSS/DWM
Phillip Ray	NRR/DLPM
Michael Webb	NRR/DLPM
Tim Johnson	NMSS/DWM
Vonna Ordaz	NRR/DSSA
George Hubbard	NRR/DSSA
Duke Wheeler	NRR/DLPM
Thomas Fredrichs	NRR/DLPM
Mike Meisner	MYAPC
Lynnette Hendricks	NEI
James Curry	GPU Nuclear
Alan Nelson	NEI
Raymond Shadis	Friends of the Coast
Mike Laggart	GPU Nuclear
P. J. Atherton	Self

Enclosure 2

United States
Nuclear Regulatory Commission

DECOMMISSIONING REGULATORY IMPROVEMENT

PRESENTATION DURING MEETING WITH NEI

May 5, 1999

William Huffman
Decommissioning Section
Project Directorate IV
Division of Licensing and Program Management
Office of Nuclear Reactor Regulation

Enclosure 3

TASKS UNDERWAY BY NRC STAFF

- **Spent Fuel Pool Risk Assessment**
  - **Technical Staff is assessing spent fuel pool risks and developing criteria to evaluate regulatory relief for decommissioning nuclear power plants**

- **Decommissioning Regulatory Improvement**
  - **Projects Staff will apply the criteria developed by the technical staff to improve and risk-inform decommissioning rulemakings currently in progress**
  - **Projects Staff is evaluating other options which will improve the regulation of nuclear power plants undergoing decommissioning**

DECOMMISSIONING REGULATORY IMPROVEMENT

- **Staff is considering consolidating the decommissioning regulations of 10 CFR Part 50 to a dedicated location**
  - **Will only contain rules applicable to nuclear power plants that have permanently ceased operation**
  - **Will eliminate uncertainties as to rules that no longer apply to decommissioning**
  - **Will permit changes to decommissioning requirements without impacting operating licensees**
  - **Will help identify additional decommissioning regulations where risk-informing might reduce unnecessary regulatory burden**
- **Any changes will be accomplished through the normal rulemaking process which permits ample opportunity for public and industry comments**

GOALS

- **To enhance the clarity, efficiency, and effectiveness of decommissioning regulations while maintaining safety**
- **To improve public confidence in the regulatory process of decommissioning nuclear power reactors**
- **Staff encourages comments and questions from the industry and public stakeholders**

Technical Working Group Mission Statement

The Technical Working Group will review and evaluate available technical information and methods to use as an interim risk-informed, technical basis for reviewing exemption requests and rulemaking related to EP, safeguards, indemnification, and other areas. This activity may also identify the need for follow up research or activities to address areas of large uncertainty.

Working Group
Technical Basis Outline

I. Introduction

II. Spent Fuel Pool (SFP) Accident Scenarios
   A. Identification of initiating events that could lead to spent fuel uncovery (Including qualitative screening of events that are not risk significant)
      1. Internal events (e.g., LOSP, loss of UHS, loss of CCW/SW, loss of coolant flow, fire, etc.)
      2. External events (e.g., seismic, tornado/high winds, aircraft impact)
      3. Errors of commission (e.g., heavy load drop, maintenance errors leading to draining of pool, etc.)
   B. Identification of available systems for the mitigation of the initiating event (plant configuration, system alignment, backup systems available, etc.)
   C. Identification of potential operator recovery actions (availability of alarms, instrumentation, procedures, staffing, etc.)
   D. Formulation of accident sequences
      1. Success criteria (timing, system flow rates, etc.)
      2. Accident sequence progression using event trees
      3. System modeling and recovery actions using fault trees
   E. Description of the initiating events under Section II.A.

III. Quantification of Accident Frequency
   A. Estimate frequency of initiating events that could lead to spent fuel uncovery (For each event identified, but not qualitatively screen out it item II.A.)
      1. Existing data (e.g., for LOSP)
      2. Literature search (e.g., site specific hazard curves, load drops, aircraft impact, tornados)
      3. Seismic hazard curves for Susquehanna & Pilgrim in III.A.2.
      4. Fault tree analysis for loss of support system initiating events
      5. HRA for errors of commission
   B. Estimate equipment failure probability for active and passive components/systems. Estimate availability of backup systems.
      1. Information from plant walkdowns
      2. AEOD data
      3. Information from literature search
   C. Perform a human reliability analysis to estimate error probabilities for recovery actions.
   D. Quantify fault trees and event trees using best estimate data. Discuss quantification uncertainty in a qualitative sense.

IV. Consequences of SFP accident scenarios
   A. Inventory discussion on reduction of consequences over time
   B. Evaluation of release fraction due to a zircaloy fire.
   C. Evaluation of inventories of each radionuclide.
   D. Dose assessments for time-dependent offsite consequences for a zircaloy fire [based on Millstone 1, and a fire that covers 3 cores of spent fuel].

1. 30 days with offsite EP and without offsite EP
2. 90 days with offsite EP and without offsite EP
3. One year with offsite EP and without offsite EP

E. Identification of consequences (e.g., early fatalities, cancer fatalities, total population dose)
F. Consequences of other SFP accident scenarios (e.g., loss of cooling)
G. Evaluation of existing accident dose assessments to determine if they represent current operating and storage practices and if they are applicable to decommissioned plants.

V. Overall Risk of SFP accidents at Decommissioned Plants
 A. Risk at 30 days with offsite EP and without offsite EP
 B. Risk at 90 days with offsite EP and without offsite EP
 C. Risk at one year with offsite EP and without offsite EP

VI. Spent Fuel Pool Heatup Analysis Following Loss of Water
 A. Evaluation of the phenomena of a zircaloy fire
  1. Literature search
   a. NRC documentation on zirc fires
   b. UM library for zirc & similar metal fire data
   c. NIST FIREDOC database for zirc & similar metal fire data
   d. Contact DOE for data & experience w/fuel cladding fires
   e. Contact foreign entities for experience/research w/zirc fires
  2. Evaluation of whether to model the zircaloy fire (e.g., fire/yr)
 B. Fuel Failure Criteria
  1. Evaluation of 565 degrees C as an appropriate acceptance criterion for analysis and/or,
  2. Recommendation on an appropriate temperature
 C. Evaluation of existing spent fuel heat up analyses
  1. Evaluation of GSI-82, SHARP Code, and NUREG-6451
  2. Determine if they represent current operating and storage practices, and if they are applicable to decommissioned plants
 D. Heatup Calculation Uncertainties and Sensitivities
  1. Evaluation of existing computer codes (e.g., SHARP, etc.)
  2. Determine if they could be used to analyze the heat up of the SFP
 E. Critical Decay Times for Reaching a Zirc Fire
  1. Perform a 2 year/4 year decay time simulation of a generic BWR using the Fluent Code
  2. Evaluation of the generic decay times associated with SFP configurations
 F. Evaluation of potential fire protection mitigating controls (e.g., high expansion foam, unattended nozzle, etc.)

VII. Structural integrity of the SFP structure
 A. Current NRC studies
 B. Hazards to consider (e.g., seismic, heavy load drops, tornado missiles, safegds)
 C. Risk Ranking of hazards

D. Structure failure modes
E. Deterministic considerations
F. Risk-Informed Performance Goal

VIII. Potential for criticality
A. Evaluation of the potential for criticality from accidents
B. Evaluation of the potential for criticality from personnel actions in response to an accident
C. Evaluation of the worst case criticality scenario (i.e., no boral)
D. Evaluation of potentia! for criticality at older plants

IX. Effects of other Programs
A. Maintenance Rule
1. Identification of maintenance rule concepts at decommissioned plants
2. Identification of potential systems, equipment, functions at decommissioned plants
3. Evaluation of what maintenance rule means to decommissioned plant oversight.
B. Quality Assurance (QA) Programs
1. Identification of QA concepts at decommissioned plants
2. .Jentification of potential QA programs at decommissioned plants
3. Evaluation of how QA applies to decommissioned plant oversight.

X. Comparison of design considerations for Wet-Basin ISFSIs
A. Defense-in-depth
B. Minimum decay time
C. Design events
D. Controls

XI. Technical basis for reviewing SFP accidents for exemption requests that can be applied to emergency preparedness, safeguards, and insurance indemnity at decommissioned plants.
A. Identify risk-informed approach and guidelines
B. Recommend any administrative or other controls (e.g., enhanced TSs for level, temperature, etc.), if necessary

XII. Identify follow up research or other technical support which needs to be performed to address any large uncertainties in the available information.
A. NRC work (NRR, NMSS, RES or contractors, such as INEL, PNNL, etc.)
B. External to the NRC (i.e., NEI, Owner's Groups, etc.)

Solicitation for Additional Information

- **Identification of initiating events and accident sequences:** What are the correct accidents to be evaluating? Why and/or when can an accident be eliminated as a concern?

As discussed in the Maine Yankee Defueled Safety Analysis Report (DSAR), the consequences of all of the assumed possible accident analyses are quite low. Specifically, due to the greater than two year spent fuel decay time, accidents associated with fuel criticality and handling are insignificant compared to the 10 CFR 100 limits and the EPA Protective Action Guidelines.
Accidents for Decommissioning Plants (DBA's and BDBA's)

Low Level Waste Release Incident (Liquid, Gas, Resin)
Fuel Handling Accidents (fuel drop, insufficient shielding)
Fuel Criticality Accidents (fuel misplacement, boron dilution)
Loss of Spent Fuel Pool Inventory
Loss of Spent Fuel Pool Cooling
Spent Fuel Cask Drop

Low Level Waste Release Incidents

Radioactive Waste Gas System Leaks and Failures

The inventory of the waste gas decay tanks can be eliminated shortly after permanent plant shutdown. Therefore this accident may be eliminated from scope.

Radioactive Liquid Waste System Leaks and Failures

Potential releases to the Atmosphere – The inventory of liquid waste may be eliminated shortly after the conclusion of chemical decontamination activities. Even with the limiting event of a rupture of the primary drain tank the consequences are below the EPA - PAG's

Resin Spills or Fires

This is the bounding offsite release dose consequence for design basis accidents at Maine Yankee(using a hypothetical resin cask with an inventory of 20,000 curies). Even with conservative assumptions, the consequences are below the EPA - PAG's. Depending on the type of spent fuel pool purification method, some small amount of contaminated resin will be produced periodically.

Fuel Handling Accidents

Spent Fuel Assembly Drop

The largest contributor to offsite release dose is from the short-lived iodines. Following a modest level of decay (3-4 months), offsite release dose is negligible. Site boundary exposures are reduced to at least two orders of magnitude below EAP PAGs.

Insufficient Shielding

This scenario is coupled with either a lost of inventory in the fuel pool or inadvertent raising of a fuel bundle to near the surface. The dose at the top of the pool as a function of water depth and decay time is presented in Fig. 5.5-3 in the Maine Yankee DSAR. A loss of 16 feet of water inventory, to a level of six feet above the active fuel, would result in a dose at top of pool of about 1 rad/hr. This dose is low enough to allow for operator action to restore pool level. As presented in Figure 5.5-4 of the Maine Yankee DSAR, the projected skyshine Radiation Dose at 610 meters (Exclusion Area Boundary) from the pool would be about 4.7E-7 rad/hr for the same scenario. Raising of a fuel bundle to cause a high radiation dose at the surface of the fuel pool, assuming normal or near normal water level, is precluded by the physical limitations of the fuel crane.

Fuel Criticality Accidents

At Maine Yankee, the Boral fuel racks are designed to preclude criticality assuming no boration of the Fuel Pool water. Misplaced/dropped assemblies are precluded from criticality, assuming the worst case conditions, by the borated Fuel Pool water. Fuel Pool water is maintained borated to about 30% more than required by analysis. This analysis assumes new fuel assemblies. Maine Yankee no longer has any new fuel assemblies. Unless a fuel bundle is misplaced or otherwise not in its assigned location, boron dilution to pure water cannot cause criticality due to the rack design.

Loss of Spent Fuel Pool Inventory

The active portion of the fuel is stored at or below outside grade level (el.21'0"). Drain down of the fuel pool below this grade elevation by leakage or breach of the pool is highly unlikely. The suction and discharge piping of the fuel pool cooling system is protected by passive syphon breakers which would prohibit drain down by siphon. At Maine Yankee the only penetration into the pool below the level of the fuel is the fuel transfer tube into the Containment building. This tube is sealed on the pool side by a valve and on the containment side by a blank flange. Any human error related to a leak path through the fuel transfer tube would be impossible. This would require the deliberate act of two individuals. Only incredible events (eg. meteor, massive earthquakes) would be precursor events to a loss of inventory rapid enough to preclude operator action.

Loss of SFP Inventory due to loss of cooling

The Maine Yankee DSAR documents the parametric studies prepared to demonstrate the time available for operation action to recover from a loss of SFP cooling or loss of SFP inventory event. With the spent fuel cooled for over two years and the syphon breaks installed in the SFP cooling supply and return lines, it was calculated that it would take approximately 64.6 hours to reach bulk boiling (assuming an initial bulk temperature of 100°F and an initial pool level of 40 Ft.). Given these parameters, the calculated boil-off rate would be 9.22 gpm or a loss of SFP level of 1.16 feet/day. These estimates are quite conservative since they assume considerably more latent heat than as actually exists (i.e. The SFP heat-up test conducted in 1997 determined that the ANS decay heat values used in the analyses were high by approximately 60%.) And since all convective and evaporative heat losses are conservatively neglected. Given these inputs the consequence of loss of SFP cooling results in dose levels in the fuel building of less than 2 mR per hour.

Although the ruggedness of the new Maine Yankee SFP cooling system is not credited in the loss of SFP cooling evaluation, it should be pointed out that the new Decay Heat Removal (DHR) system was designed and installed to resist the 0.18g NUREG/CR-0098 response spectrum which resulted from the 1987 Maine Yankee Seismic Margin Review (NUREG/CR-4826). Additionally, the system contains a built-in 100% capacity spare pump and it can be powered by the site's dedicated Security diesel generator. As discussed below, numerous sources of make-up water are available to credit operator actions to adequately assure make-up capability to the SFP.

Gross Seismic Failure of the SFP:

The Maine Yankee SFP is a reinforced concrete structure comprised of 6-foot thick walls and base slab. The pool is supported directly on bedrock and is embedded up to approximately Elevation 21' (or to approximately the top of the spent fuel assemblies). During the NRC-sponsored Maine Yankee Seismic Margin Review (NUREG/CR-4826), the structure was screened out as a seismic initiator having a High-Confidence-of-a-Low-Probability-of-Failure (HCLPF) capacity of > 0.30g (which was the Margins program's screening threshold). NUREG/CR-4334 estimates that similar reinforced concrete structures can be screened out up to HCLPF values of 0.5g ZPA. Similarly, NUREG/CR-5176 calculated a median seismic capacity for a "representative" PWR as 2.0g and the resulting HCLPF as 0.65g. The probability of such a large magnitude earthquake along the coast of Maine can be gleaned from either the EPRI or LLNL seismic hazard curves.

Other Initiators:

A review of "Other" potential initiators, such as tornadoes, fires or operator errors has yielded no credible initiators. The massive, passive SFP is robust enough to resist the most damaging tornado missiles, relies on no operating systems or immediate operator actions, thereby greatly reducing the chance of an operator error or the failure of necessary equipment and is not susceptible to a fire-induced failure.

Loss of inventory through sabotage related initiators are discussed and evaluated in NRC correspondence. [1] The consequences of these inventory related scenarios are minimized due to the fact that the active portion of the fuel is stored at or below outside grade level (el.21'0").

In conclusion, other than a seismically-induced gross failure of the SFP, any other accidents result in minor off-site dose effects and/or days of response time for operators to take remedial actions to restore cooling or to provide make-up water to compensate for loss of cooling or minor leakage losses.

Loss of Spent Fuel Pool Cooling

The Maine Yankee Spent Fuel Pool has been analyzed to 212 °F. The present boil off rate (700 days after shutdown) is less than 10 GPM which results in a boil off loss of less than 1.3 feet per day. This scenario is not a credible accident condition because of the very long time available for operator action.

Spent Fuel Pool Cask Drop

At present Maine Yankee is precluded by license condition from lifting a cask over the fuel pool. Although not part of the present license basis, analyses have been performed which demonstrate a cask drop will result in leakage from the pool but not catastrophic failure. Since the active storage portion of the pool is below grade, leakage from the pool is also limited by the geology of the rock below the pool structure.

Maine Yankee currently prohibits bringing a spent fuel shipping cask into the spent fuel pool (SFP), however, this prohibition will have to be eliminated in order to proceed with plans to create an onsite, dry cask ISFSI. The ISFSI project scope includes plans to replace/modify the existing yard crane (CR-3)

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1 Maine Yankee Letter to NRC dated March 5, 1998 (MN98-14) and July 1, 1998 (MN98-52) Safeguards Information

with a single-failure-proof crane which meets all of the requirements of NUREG/CR-0612.

Besides upgrading the existing crane, Maine Yankee has completed a number of consequence analyses to demonstrate the "worst-case" scenarios associated with a postulated spent fuel shipping cask drop accident. The potential damage to spent fuel, resulting from the postulated drop of an assumed 125 ton shipping cask, was shown to be not a return to criticality concern and that off-site dose level would be well less than the PAGs. An upper bound SFP leak rate of 5 gpm was calculated for the assumed cask drop. This make-up value is less than the calculated evaporation/boil-off value calculated for the loss-of-SFP cooling accident analyses and can easily be compensated for by the various available make-up sources. Available make-up sources include: SFP make-up via pump P-SFP2 and the 160,000 gallon primary water storage tank, the 3,000,000 gallon on-site fire pond and either the electric (P-4) or diesel powered (P-5) fire pumps, the Town of Wiscassett site water line or other actions which could be initiated within the days available (Refer to the loss of SFP cooling discussion for more details on available time for operator actions).

Other Heavy Loads

Current plans call for a prohibition of heavy loads in the vicinity of the containment side of the fuel transfer tube. Greater-than-Class-C (GTCC) wastes resulting from segmentation of the reactor internals will be packaged in the containment with no plans to re-open the fuel transfer tube, thereby eliminating the possibility of introducing any operator errors that could accidentally draining the SFP. With the fuel transfer tube isolation flange installed on the containment side of the transfer tube and the tube isolation valve (FP-21) closed and administratively controlled on the fuel pool side of the tube, there is no credible way of accidentally draining the SFP.

- **Probability of initiating events and accident sequences:** Existing information based on operating reactors and had large uncertainties associated with the estimate? How can these be improved? What else, such as human error, needs to be examined?

Seismic Events

Recent work by EPRI and LLNL regarding the probability of seismic events of a given magnitude should be the basis for determining precursor events. Cask drop events, for those designs susceptible to severe damage from such events should be considered.

- **Methods or criteria to assess scenarios and consequences:** This is a very large, fundamental question – What type or types of analysis should be used? What criteria should be used? Can generic parameters be defined?

- **Mitigative actions or features:** Is there equipment or personnel actions that can be given credit for a given accidents(s)?

The capability of the fuel pool to retain some water due to its relationship to grade elevation should be considered. The active portion of the fuel is stored at or below outside grade level (el.21'0"). Drain down of the fuel pool below this grade elevation by leakage or breech of the pool is highly unlikely.

The availability of make-up water and the ease of initiating make-up should be considered. Available make-up sources include: SFP make-up via pump P-SFP2 and the 160,000 gallon primary water storage tank, the 3,000,000 gallon on-site fire pond and either the electric (P-4) or diesel powered (P-5) fire pumps, the Town of Wiscassett site water line or other actions which could be initiated within the days available

The long time available for operator detection and mitigating action in loss of fuel pool inventory and loss of cooling scenarios should be considered. With the spent fuel cooled for over two years and the syphon breaks installed in the SFP cooling supply and return lines, it was calculated that it would take approximately 64.6 hours to reach bulk boiling (assuming an initial bulk temperature of 100°F and an initial pool level of 40 Ft. - minimum siphon elevation) Given these parameters, the calculated boil-off rate would be 9.22 gpm or a loss of SFP level of 1.16 feet/day. At this rate it would take over 12 days for the spent fuel pool level to reach 4 feet above the fuel racks assuming no operator action. At this level, the radiation dose rate at the exclusion area boundary is less than 2.6×10^{-6} Rad/hr.

The location of the plant in relation to the probability and severity of seismic events should be considered as well as the seismic ruggedness of the pool design.

- **Characteristics of Zircaloy fire:** How does it behave? How energetic is the release? How much is released? When is propagation a concern?

- **Dose from fire after 30 days post-shutdown and beyond:** Previous studies evaluated dose from fire at 30 days; limited fire at 90 days but it appears that they did not evaluate the consequences of a fire when the fuel is older than 90 days – is there a point in time that event does not have offsite consequences?

According to NUREG 0654 and NUREG 0396 the bounds of the parameters for which planning was recommended were identified based upon a knowledge of the potential consequences, timing, and release characteristics of a spectrum of accidents. As described in NUREG-1353 "Regulatory Analysis for the Resolution of Generic Issue 82, Beyond Design Basis Accidents in Spent Fuel Pools, "the source term for the spent fuel pool accident is not the same as the source term associated with core damage accidents. The consequences of a spent fuel pool accident which results in the complete loss of water are dominated by the long lived isotopes, such as cesium and strontium. " "A direct comparison of the consequences of a severe accident in a spent fuel storage pool to the consequences of a severe core accident can be misleading. For the spent fuel pool accident there are no 'early' fatalities and the risk of early injury is negligible. For a severe core damage accident, early fatalities and early injury are part of the risk due to the presence of the shorter lived isotopes." From this discussion, it is clear that the Zirc fire scenario would not have been one of those accident sequences which contributed to the basis for the emergency plan requirements since the accident consequences and release characteristics fall outside the bounding parameters that necessitate offsite response capability.

DSAR Table 5.3.1 (attached) provides the bounding fuel rod gap radiological inventories assumed in the fuel handling accident analysis assuming one year of decay.

Table 5.3.1

BOUNDING SPENT FUEL INVENTORIES FOR RADIOLOGICAL ANALYSES[2]

NUCLIDE	ASSEMBLY INVENTORY (Ci)[1]
I-129	1.85 E-02
I-131	6.60 E-09
Kr-81	3.85 E-07
Kr-85	4.04 E+03
Xe-129m	5.97 E-14
Xe-131m	5.86 E-06
Xe-133	8.06 E-16

(1) Inventories include a 5% uncertainty factor.

(2) On a per assembly basis.