GROUP D

FOIA/PA NO: 2013-0240

RECORDS BEING RELEASED IN THEIR ENTIRETY

Cahill, Christopher

From: Cahill, Christopher
Sent: Wednesday, October 03, 2012 4:29 PM
To: Ziedonis, Adam
Cc: Hansell, Samuel
Subject: Spent Fuel Pool HRA

Adam,

I'm working with the folks from HQ on the final touches for the HRA evaluation associated with the spent fuel pool scoping study and was wondering if you could answer a few questions for us.

1) The portable pumps that the site uses for a loss of pond event (located on the first floor by the cafeteria), how many are there and what is their flow/head?
2) Refueling level indication – Does it get power during a LOOP and SBO?

I appreciate the help. If it's too much of a distraction from you current assignments please let me know and I'll try another avenue.

Thanks again,
Chris

Cahill, Christopher

From: Cahill, Christopher
Sent: Wednesday, October 05, 2011 8:39 AM
To: Zigh, Ghani
Subject: RES Seminar - Investigations of Zirconium Fires During Spent Fuel Pool Loss of Coolant Accidents - 10/05/11

I'm setting up the VTC in RI to attend the training. Are there any slides available that I can share with the staff?

Thanks,
Chris

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Cahill, Christopher

From: Cahill, Christopher
Sent: Thursday, October 04, 2012 7:30 AM
To: Ziedonis, Adam
Subject: RE: Spent Fuel Pool HRA

Thanks again!

From: Ziedonis, Adam
Sent: Thursday, October 04, 2012 6:48 AM
To: Cahill, Christopher
Cc: Hansell, Samuel
Subject: RE: Spent Fuel Pool HRA

Chris – I can definitely get you #2 today....I know who to ask for #1, so that will just be a matter of how quickly that person can get me the info....shouldn't be more than a day or two I would imagine

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From: Cahill, Christopher
Sent: Wednesday, October 03, 2012 4:29 PM
To: Ziedonis, Adam
Cc: Hansell, Samuel
Subject: Spent Fuel Pool HRA

Adam,

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I appreciate the help. If it's too much of a distraction from you current assignments please let me know and I'll try another avenue.

Thanks again,
Chris

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Cahill, Christopher

From: Cahill, Christopher
Sent: Monday, July 09, 2012 9:59 AM
To: R1DRSWORKFLOW RESOURCE
Subject: FW: PB SFP HRA
Attachments: PB SFP HRA

Please add to the DRS calendar. I am the only DRS person attending.

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Cahill, Christopher

Subject: PB SFP HRA

Start: Thu 7/12/2012 12:00 AM
End: Fri 7/13/2012 12:00 AM
Show Time As: Free

Recurrence: (none)

Organizer: Cahill, Christopher

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Cahill, Christopher

Subject:	Peach Bottom SFP Scoping Stydy HRA
Location:	RES Church St
Start:	Wed 8/1/2012 12:00 AM
End:	Thu 8/2/2012 12:00 AM
Show Time As:	Tentative
Recurrence:	(none)
Meeting Status:	Not yet responded
Organizer:	Cahill, Christopher
Required Attendees:	R1DRSCAL RESOURCE; Schmidt, Wayne; Cook, William

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Cahill, Christopher

From: Cahill, Christopher
Sent: Wednesday, October 05, 2011 8:30 AM
To: R1-DL-DRS; R1-DL-DRP; R1-DL-DNMS
Subject: VTC RES Seminar - Investigations of Zirconium Fires During Spent Fuel Pool Loss of Coolant Accidents - 10/05/11

Event: RES Seminar - Investigations of Zirconium Fires During Spent Fuel Pool Loss of Coolant Accidents - 10/05/11

Title: Investigations of Zirconium Fires During Spent Fuel Pool Loss of Coolant Accidents

Date: October 5, 2011

Time: 10:00 A.M. to 11:00 a.m.

Place: RI First Floor VTC Room

Presenters: Dr. Ghani Zigh, RES/DSA

Cahill, Christopher

From: Cahill, Christopher
Sent: Wednesday, June 13, 2012 11:31 AM
To: Mitman, Jeffrey; Zoulis, Antonios; Chang, James
Subject: FW: GoToMeeting Invitation - Seismic SFP HRA study

-----Original Appointment-----
From: Cahill, Christopher
Sent: Wednesday, June 13, 2012 11:21 AM
To: DO, HA
Subject: Tentative: GoToMeeting Invitation - Seismic SFP HRA study
When: Wednesday, June 13, 2012 2:00 PM-3:00 PM (GMT-05:00) Eastern Time (US & Canada).
Where:

I have Saphire 8 Training to support starting at 1:00 and likely will last until 2:30.

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Cahill, Christopher

From: Cahill, Christopher
Sent: Monday, June 25, 2012 3:47 PM
To: Mitman, Jeffrey
Subject: Accepted: JTM & AMZ - At Peach Bottom for SFPSS Mitigating Strategy walkdowns

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Cahill, Christopher

From: Cahill, Christopher
Sent: Thursday, July 19, 2012 6:09 AM
To: Mitman, Jeffrey
Subject: Accepted: SFPSS HRA Meeting

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Cahill, Christopher

From: Cahill, Christopher
Sent: Tuesday, October 02, 2012 7:30 AM
To: Mitman, Jeffrey
Subject: Accepted: GoToMeeting Invitation - SFP HRA

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Cahill, Christopher

From: Cahill, Christopher
Sent: Monday, April 08, 2013 9:07 AM
To: Algama, Don
Subject: Accepted: FW: SFPSS: ACRS Sub-Committee Meeting (HRA, 2.1 and 2.2) -r2

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Cahill, Christopher

From: Cahill, Christopher
Sent: Wednesday, January 23, 2013 7:17 AM
To: Miller, Chris
Subject: RE: SFP Scoping Study

I was expecting this. The PM contacted me last week, so I asked that they send a request through you.

From: Miller, Chris
Sent: Tuesday, January 22, 2013 5:32 PM
To: Cahill, Christopher
Cc: Schmidt, Wayne; Clifford, James; Roberts, Darrell; Wilson, Peter
Subject: SFP Scoping Study

Chris,
Kathy Gibson called and indicated that RES is beginning the process to get concurrence and then issue the study. She asked if Region I could be a collection point for the comments from the Regions (helps to consolidate similar comments or make sense of them, not necessarily to hammer out an agreed upon wording for a final regional input) as was done for SOARCA process. I told her we could do that, unless it got to be too onerous or time consuming an affair and then we would re-negotiate. Let me know if you have any concerns, since you will be our point person and will likely be getting the input from other regions.
Thx
chris

Christopher Miller
USNRC Region I
Director
Division of Reactor Safety
610-337-5128

13

Cahill, Christopher

From: Cahill, Christopher
Sent: Tuesday, April 02, 2013 3:26 PM
To: Algama, Don
Subject: SFP Inputs

Don,

I haven't seen any inputs to consolidate from the other regions. Any idea who has the lead in them?

Chris

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Cahill, Christopher

From: Cahill, Christopher
Sent: Friday, April 12, 2013 8:58 AM
To: Gray, Mel
Subject: RE: Peach Bottom Spent Fuel Pool Scoping Study

The report is in draft, so I wouldn't 't comment. I'm just passing along the info so you are up to speed.

From: Gray, Mel
Sent: Thursday, April 11, 2013 4:01 PM
To: Cahill, Christopher
Cc: Hansell, Samuel
Subject: RE: Peach Bottom Spent Fuel Pool Scoping Study

We have an annual assessment open house next Thursday. Are there current messages we should have on this in process study?

From: Cahill, Christopher
Sent: Thursday, April 11, 2013 10:15 AM
To: Gray, Mel
Cc: Schmidt, Wayne; Cook, William
Subject: FW: Peach Bottom Spent Fuel Pool Scoping Study

Mel,

Here is an update on the study. I expect it will high visibility issue in the near future

Chris

From: Cahill, Christopher
Sent: Thursday, April 11, 2013 10:07 AM
To: McNamara, Nancy; Tifft, Doug
Subject: Peach Bottom Spent Fuel Pool Scoping Study

Nancy/Doug

As I mentioned this morning this study is moving along and I expect we will have a lot of stake holder interaction. I expect, as with other very low probability/high consequence events, communications on this to be challenging. The attached is a summary presentation of the report. Here are some of the highlights:

- No early fatalities
- Low individual latent cancer fatality risk (1E-4 per year)
- Doses are dominated by long-term exposure to lightly contaminated areas
- Significant Cs-137 release can cause a considerable amount of land interdiction and relocation of individuals.

Slide 17 discusses land contamination, which will likely generate a lot of interest. Slide 20 gives a comparison to previous reports.

Here is the schedule going forward.

1. ACRS Meeting

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a. SC meeting on May 8th

i. Currently RES is the lead on this. Expect that NRR/JLD will provide a few comments to frame the SFPSS project within the NRR Tier 3 activities

b. FC meeting is likely on June 5th (tentative, will depend on JLD)

i. It is expected that NRR/JLD, after the Comm.SECY is approved by the Commission, will lead this meeting with RES as support for SFPSS

2. Report finalized?

a. Still looking at a October 2013 deliverable date to NRR

Chris

Cahill, Christopher

From: Miller, Chris
Sent: Wednesday, June 20, 2012 11:18 AM
To: Schmidt, Wayne; Wilson, Peter
Cc: Cook, William; Cahill, Christopher; Roberts, Darrell; Clifford, James; Dean, Bill; Lew, David; Bearde, Diane; Krohn, Paul
Subject: Re: Brief on SFP study at PB

There was lack of clarity among those who periodically interface with licensee and Commission on purpose, assumptions,, results, public communication, and plans fwd. RES and we thought it would be good to get folks who may have to answer questions about this project on the same page.
Sent from NRC BlackBerry

From: Schmidt, Wayne
To: Miller, Chris; Wilson, Peter
Cc: Cook, William; Cahill, Christopher; Roberts, Darrell; Clifford, James; Dean, Bill; Lew, David; Bearde, Diane; Krohn, Paul
Sent: Wed Jun 20 11:11:01 2012
Subject: RE: Brief on SFP study at PB

We had a briefing on this at the SRA counterpart meeting and I thought I summarized it earlier.

From: Miller, Chris
Sent: Wednesday, June 20, 2012 11:05 AM
To: Wilson, Peter
Cc: Cook, William; Cahill, Christopher; Schmidt, Wayne; Roberts, Darrell; Clifford, James; Dean, Bill; Lew, David; Bearde, Diane; Krohn, Paul
Subject: Brief on SFP study at PB

Based on our conversation yesterday, Rich Correia is going to have Kathy Gibson work with us to set up a 1 hr brief for us on what is known, preliminary results etc. Kathy is going to get back with Pete or me to set this up. I am assuming we would have enough interest from DRP, DRS, and ORA to have a medium sized conference room.
chris

Christopher Miller
USNRC Region I
Director
Division of Reactor Safety
610-337-5128

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Schroer, Suzanne

From:	Schroer, Suzanne
Sent:	Thursday, July 14, 2011 6:14 PM
To:	VanWert, Christopher
Cc:	Mrowca, Lynn; Thomas, Brian; Frankl, Istvan; Hernandez, Raul; Hart, Michelle
Subject:	Spent Fuel Integrity WG
Attachments:	SFP Integrity.docx

Chris,

Attached is SPRA's input for the July 19th briefing. Please let me know if you have any questions.

Thanks!
Suzanne

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SPRA Input for SFP Integrity Working Group 07/14/11

NUREG-1738, "Technical Study of Spent Fuel Pool Accident Risk at Decommission Nuclear Power Plants" (2001)

Section 3.5, Beyond Design Basis Spent Fuel Pool Accident Scenarios (External Events). A simplified seismic risk analysis was done; however, the only transients considered were failures of the pool and pool cooling (rapid loss of inventory).

Appendix 6, Stakeholders Concerns Rose During the Public Comment Period. A commenter asked about corrosive chemicals being introduced to the fuel pool and if there is a potential for fuel damage if this should occur. The staff's response was, "even if fuel damage did occur, the shielding provided by the large volume of water above the fuel would preclude any significant radiation release."

Resolution of Generic Issue 82: Beyond Design Basis Accidents in Spent Fuel Pools (1989)

Seismic probabilities were considered, but the discussion is limited to the possibility of a seismic event opening the pool and draining it. The final analysis was that "the staff concluded that reducing the risk for spent fuel pools due to events beyond the SSE would still leave a comparable risk due to core damage accident. Because of the large inherent safety margins in the design and construction of spent fuel pools, this issue was RESOLVED and no new requirements were established."

Letter on Resolution of Generic Safety Issue 173A, Spent Fuel Pool Cooling for Operating Plants (2001)

Although not issue-specific, the letter states that the regulatory analysis guidelines for spent fuel pool issues direct that "issues with a frequency of less than 1×10^{-5} per year received no additional action." The guideline, in fact, states issues that have an estimated reduction of CDF that is less than 1×10^{-5} per year received no additional action. The staff evaluated whether this was appropriate and decided to use 1×10^{-6} per year to consider no action. The staff concluded that "the risk and consequences of an SFP accident at operating reactors meet the Commission's QHOs" and "recommend GSI-173A be considered resolved."

NUREG/CR-5176, "Seismic Failure and Cask Drop Analyses of the Spent Fuel Pools at Two Representative Nuclear Power Plants" (1989)

This study was completed in 1988 and only analyzed loss of liner integrity, loss of pool cooling, and damage to fuel racks caused by fuel rack motion. It did not, however, consider the interaction between the clad and the racks.

Conclusions: Although much research has been done with respect to SFP risk, there have been no evaluations of the probabilities of an interaction between the fuel clad and the racks. In addition, the current regulations do not require a SFP PRA, so the staff has not analyzed the condition from a risk perspective.

Palmrose, Donald

From:	Palmrose, Donald
Sent:	Friday, October 28, 2011 3:56 PM
To:	Spencer, Michael
Cc:	Hart, Michelle; Brown, David; Muir, Jessie; Whited, Ryan; Clayton, Brent
Subject:	Info related to SFPs
Attachments:	SFP accidents and environmental review; 73 FR 46204.pdf

Michael,

As you mentioned in our prior phone call, Section 6.4.6 of the GEIS has info related to SFP accidents. From reading the text, it appears that the following documents formed the basis for the conclusion in the GEIS: 55 FR 38474 and NUREG-1092. I cannot download 55 FR 38474 from GAO's Federal Register website since they only go back to Volume 59 (1994). NUREG-1092 is related to ISFSIs so I would not try to pull up that document. If you have a way of getting 55 FR 38474, this is probably the best one to try reading. Also, if you can get a copy of this FR, please send Michelle and me a copy.

Michelle sent me an email that points to 73 FR 46204 on the denial of two petitions for rulemaking related to SFPs. This document is attached.

Finally, there is also an NRC Fact Sheet on "Reducing Hazards from Stored Spent Nuclear Fuel" at http://www.nrc.gov/reading-rm/doc-collections/fact-sheets/reducing-hazards-spent-fuel.html This may only help to put into context past NRC actions.

Hope this helps,
Don

Don Palmrose
Sr. Project Manager
NRO/DSER/RAP3
301-415-3803
T7-F38

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Palmrose, Donald

From: Hart, Michelle
Sent: Friday, October 28, 2011 3:06 PM
To: Palmrose, Donald
Subject: SFP accidents and environmental review

In general, we keep referring back to the GEIS (section 6.4) with respect to any potential for SFP accident consequences. The general discussion you gave about the design preventing accidents is a major portion of the assessment. An updated and good discussion of the NRC position on SFP accidents and the comparison to reactor accidents (i.e., SFP less than reactor accidents) is given in the Federal Register notice denying two rulemaking petitions from August 8, 2008 (73 FR 46204). Sandia looked at SFP accident progression in 2008, using MELCOR. I don't think the analyses really went to consequences, but SOARCA included something about the SFP accidents (I think – I really need to read the report).

I'm leaving a little early – in about 5 minutes, in fact.

Michelle

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Proposed Rules

Federal Register
Vol. 73, No. 154
Friday, August 8, 2008

This section of the FEDERAL REGISTER contains notices to the public of the proposed issuance of rules and regulations. The purpose of these notices is to give interested persons an opportunity to participate in the rule making prior to the adoption of the final rules.

NUCLEAR REGULATORY COMMISSION

10 CFR Part 51

[Docket No. PRM–51–10, NRC–2006–0022 and Docket No. PRM–51–12, NRC–2007–0019]

The Attorney General of Commonwealth of Massachusetts, The Attorney General of California; Denial of Petitions for Rulemaking

AGENCY: Nuclear Regulatory Commission (NRC).

ACTION: Petition for rulemaking; denial.

SUMMARY: The NRC is denying two petitions for rulemaking (PRM), one filed by the Attorney General of the Commonwealth of Massachusetts (Massachusetts AG) and the other filed by the Attorney General for the State of California (California AG), presenting nearly identical issues and requests for rulemaking concerning the environmental impacts of the high-density storage of spent nuclear fuel in large water pools, known as spent fuel pools (SFPs). The Petitioners asserted that "new and significant information" shows that the NRC incorrectly characterized the environmental impacts of high-density spent fuel storage as "insignificant" in its National Environmental Policy Act (NEPA) generic environmental impact statement (EIS) for the renewal of nuclear power plant licenses. Specifically, the Petitioners asserted that spent fuel stored in high-density SFPs is more vulnerable to a zirconium fire than the NRC concluded in its NEPA analysis.

ADDRESSES: You can access publicly available documents related to these petitions for rulemaking using the following methods:

Federal e-Rulemaking Portal: Go to *http://www.regulations.gov* and search for documents filed under Docket ID [NRC–2006–0022] (PRM–51–10), and [NRC–2007–0019] (PRM–51–12).

NRC's Public Document Room (PDR): The public may examine and have copied for a fee publicly available documents at the NRC's PDR, Public File Area O1 F21, One White Flint North, 11555 Rockville Pike, Rockville, Maryland.

NRC's Agencywide Documents Access and Management System (ADAMS): Publicly available documents created or received at the NRC are available electronically at the NRC's electronic Reading Room at *http://www.nrc.gov/reading-rm/adams.html.* From this page, the public can gain entry into ADAMS, which provides text and image files of NRC's public documents. If you do not have access to ADAMS or if there are problems in accessing the documents located in ADAMS, contact the NRC PDR reference staff at 1–899–397–4209, 301–415–4737, or by e-mail to *pdr.resource@nrc.gov.*

FOR FURTHER INFORMATION CONTACT: L. Mark Padovan, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, telephone (301) 415–1423, e-mail *Mark.Padovan@nrc.gov.*

SUPPLEMENTARY INFORMATION:

I. Background

The NRC received two PRMs requesting that Title 10 of the Code of Federal Regulations (10 CFR), Part 51, be amended. The Massachusetts AG filed its petition on August 25, 2006 (docketed by the NRC as PRM–51–10). The NRC published a notice of receipt and request for public comment in the **Federal Register** on November 1, 2006 (71 FR 64169). The California AG filed its petition on March 16, 2007 (docketed by the NRC as PRM–51–12). PRM–51–12 incorporates by reference the facts and legal arguments set forth in PRM–51–10. The NRC published a notice of receipt and request for public comment on PRM–51–12 in the **Federal Register** on May 14, 2007 (72 FR 27068). The California AG filed an amended petition (treated by the NRC as a supplement to PRM 51–12) on September 19, 2007, to clarify its rulemaking request. The NRC published a notice of receipt for the supplemental petition in the **Federal Register** on November 14, 2007 (72 FR 64003). Because of the similarities of PRM–51–10 and PRM–51–12, the NRC evaluated the two petitions together.

The Petitioners asserted the following in their petitions:

1. "New and significant information" shows that the NRC incorrectly characterized the environmental impacts of high-density spent fuel storage as "insignificant" in the NRC's NUREG–1437, *Generic Environmental Impact Statement for License Renewal of Nuclear Plants,* May 1996. Specifically, the Petitioners asserted that an accident or a malicious act, such as a terrorist attack, could result in an SFP being drained, either partially or completely, of its cooling water. The Petitioners further asserted that this drainage would then cause the stored spent fuel assemblies to heat up and then ignite, with the resulting zirconium fire releasing a substantial amount of radioactive material into the environment.

2. The bases of the "new and significant information" are the following:

a. NUREG–1738, *Technical Study of the Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants,* January 2001

b. National Academy of Sciences Committee on the Safety and Security of Commercial Spent Nuclear Fuel Storage, *Safety and Security of Commercial*

Spent Nuclear Fuel Storage (National Academies Press: 2006) (NAS Report)

c. Gordon R. Thompson, "Risks and Risk-Reducing Options Associated with Pool Storage of Spent Nuclear Fuel at the Pilgrim and Vermont Yankee Nuclear Power Plants," May 25, 2006 (Thompson Report)

3. Specifically, the Petitioners asserted that the "new and significant" information shows the following:

a. The fuel will burn if the water level in an SFP drops to the point where the tops of the fuel assemblies are uncovered (complete or partial water loss resulting from SFP drainage being caused by either an accident or terrorist attack).

b. The fuel will burn regardless of its age.

c. The zirconium fire will propagate to other assemblies in the pool.

d. The zirconium fire may be catastrophic.

e. A severe accident caused by an intentional attack on a nuclear power plant SFP is "reasonably foreseeable."

The Petitioners also asserted that new and significant information shows that the radiological risk of a zirconium fire in a high-density SFP at an operating nuclear power plant can be comparable to, or greater than, the risk of a core-degradation event of non-malicious origin (i.e., a "severe accident") at the plant's reactor. Consequently, the Petitioners asserted that SFP fires must be considered within the body of severe accident mitigation alternatives (SAMAs).

II. Petitioners' Requests

PRM–51–10 requested that the NRC take the following actions:

1. Consider new and significant information showing that the NRC's characterization of the environmental impacts of spent fuel storage as insignificant in NUREG–1437 is incorrect.

2. Revoke the regulations which codify that incorrect conclusion and excuse consideration of spent fuel storage impacts in NEPA decision-making documents, namely, 10 CFR 51.53(c)(2), 51.95(c) and Table B–1, "Summary of Findings on NEPA Issues for License Renewal of Nuclear Power Plants," of appendix B to subpart A of 10 CFR Part 51. Further, revoke 10 CFR 51.23(a) and (b), 51.30(b), 51.53, 51.61, and 51.80(b) to the extent that these regulations find, imply, or assume that environmental impacts of high-density pool storage are insignificant, and therefore need not be considered in any plant-specific NEPA analysis.

3. Issue a generic determination that the environmental impacts of high-density pool storage of spent fuel are significant.

4. Require that any NRC licensing decision that approves high-density pool storage of spent fuel at a nuclear power plant, or any other facility, must be accompanied by a plant-specific EIS that addresses the environmental impacts of high-density pool storage of spent fuel at that nuclear plant and a reasonable array of alternatives for avoiding or mitigating those impacts.

5. Amend its regulations to require that SAMAs that must be discussed in utility company environmental reports (ERs) and NRC supplemental EISs for individual plants under 10 CFR 51.53(c)(3)(ii)(L) and Table B–1 of appendix B to subpart A of 10 CFR part 51 ("Postulated Accidents: Severe Accidents") must include alternatives to avoid, or mitigate, the impacts of high-density pool zirconium fires.

PRM–51–12 incorporates by reference PRM–51–10. PRM–51–12 requested that the NRC take the following actions:

1. Rescind all NRC regulations found in 10 CFR part 51 that imply, find, or determine that the potential environmental effects of high-density pool storage of spent nuclear fuel are not significant for purposes of NEPA and NEPA analysis.

2. Adopt, and issue, a generic determination that approval of such storage at a nuclear power plant, or any other facility, does constitute a major federal action that may have a significant effect on the human environment.

3. Require that no NRC licensing decision that approves high-density pool storage of spent nuclear fuel at a nuclear power plant, or other storage facility, may issue without the prior adoption and certification of an EIS that complies with NEPA in all respects, including full identification, analysis, and disclosure of the potential environmental effects of such storage, including the potential for accidental or deliberately caused release of radioactive products to the environment, whether by accident or through acts of terrorism, as well as full and adequate discussion of potential mitigation for such effects, and full discussion of an adequate array of alternatives to the proposed storage project.

III. Public Comments

The NRC's notice of receipt and request for public comment invited interested persons to submit comments. The comment period for PRM 51–10 originally closed on January 16, 2007, but was extended through March 19, 2007. The public comment period for PRM 51–12 closed on July 30, 2007. Accordingly, the NRC considered comments received on both petitions through the end of July 2007. The NRC received 1,676 public comments, with 1,602 of these being nearly identical form e-mail comments supporting the petitions. Sixty-nine other comments also support the petitions. These comments were submitted by States, private organizations, and members of the U.S. Congress. Two letters from the Nuclear Energy Institute (NEI) oppose the petitions, and three nuclear industry comments endorse NEI's comments.

In general, the comments supporting the petitions focused on the following main elements of the petitions:

- NRC should evaluate the environmental impacts (large radioactive releases and contamination of vast areas) of severe accidents and intentional attacks on high-density SFP storage in its licensing decisions (NEPA analysis).
- The 2006 decision of the United States Court of Appeals for the Ninth Circuit, *San Luis Obispo Mothers for Peace* v. *NRC*, 449 F.3d 1016 (9th Cir. 2006), *cert. denied* 127 S. Ct. 1124 (2007), concluded that the NRC must evaluate the environmental impacts of a terrorist attack on SFP storage in its licensing decisions.
- NRC's claim that the likelihood of a SFP zirconium fire is remote is incorrect. Partial loss of water in an SFP could lead to a zirconium fire and release radioactivity to the environment.
- NRC's characterization of the environmental impacts of high-density SFP storage as "insignificant" in NUREG–1437 is incorrect, and the NRC should revoke the regulations which codify this.
- Any licensing decision approving high-density spent fuel storage should have an EIS.

Comments opposing the petitions centered on the following:

- Petitioners failed to show that regulatory relief is needed to address "new and significant" information concerning the potential for spent fuel zirconium fires in connection with high-density SFP storage. None of the documents that the Petitioners cited or referenced satisfy the NRC's standard for new and significant information.
- Petitioners failed to show that the Commission should rescind its Waste Confidence decision codified at 10 CFR 51.23, or change its determination that the environmental impacts of high-density spent fuel storage are insignificant.
- The Commission has recently affirmed its longstanding view that NEPA demands no terrorism inquiry,

and that the NRC therefore need not consider the environmental consequences of hypothetical terrorist attacks on NRC-licensed facilities.

• The Commission's rejection of the Ninth Circuit Court's view is consistent with the U.S. Supreme Court's position that NEPA should not be read to force agencies to consider environmental impacts for which they cannot reasonably be held responsible. Moreover, the NRC has, in fact, examined terrorism under NEPA and found the impacts similar to the impacts of already-analyzed, severe reactor accidents.

The NRC reviewed and considered the comments in its decision to deny both petitions, as discussed in the following sections:

IV. NEPA and NUREG–1437

The NRC's environmental protection regulations in 10 CFR Part 51 identify renewal of a nuclear power plant operating license as a major federal action significantly affecting the quality of the human environment. As such, an EIS is required for a plant license renewal review in accordance with the NEPA. The Petitioners challenge NUREG–1437, which generically assesses the significance of various environmental impacts associated with the renewal of nuclear power plant licenses. NUREG–1437 summarizes the findings of a systematic inquiry into the potential environmental consequences of operating individual nuclear power plants for an additional 20 years. The findings of NUREG–1437 are codified in Table B–1 of appendix B to subpart A of 10 CFR part 51.

The NUREG–1437 analysis identifies the attributes of the nuclear power plants, such as major features and plant systems, and the ways in which the plants can affect the environment. The analysis also identifies the possible refurbishment activities and modifications to maintenance and operating procedures that might be undertaken given the requirements of the safety review as provided for in the NRC's nuclear power plant license renewal regulations at 10 CFR part 54.

NUREG–1437 assigns one of three impact levels (small, moderate, or large) to a given environmental resource (e.g., air, water, or soil). A small impact means that the environmental effects are not detectable, or are so minor that they will neither destabilize, nor noticeably alter, any important attribute of the resource. A moderate impact means that the environmental effects are sufficient to alter noticeably, but not to destabilize, important attributes of the resource. A large impact means that the environmental effects are clearly noticeable, and are sufficient to destabilize important attributes of the resource.

In addition to determining the significance of environmental impacts associated with license renewal, the NRC determined whether the analysis in NUREG–1437 for a given resource can be applied to all plants. Under the NUREG–1437 analysis, impacts will be considered Category 1 or Category 2. A Category 1 determination means that the environmental impacts associated with that resource are generic (i.e., the same) for all plants. A Category 2 determination means that the environmental impacts associated with that resource cannot be generically assessed, and must be assessed on a plant-specific basis.

The NRC regulations at 10 CFR part 51, subpart A, appendix B, Table B–1 and NUREG–1437 set forth three criteria for an issue to be classified as Category 1. The first criterion is that the environmental impacts associated with that resource have been determined to apply to all plants. The second criterion is that a single significance level (i.e., small, moderate, or large) has been assigned to the impacts.[1] The third criterion is that the mitigation of any adverse impacts associated with the resource has been considered in NUREG–1437 and further, it has been determined that additional plant-specific mitigation measures are not likely to be sufficiently beneficial to warrant implementation. For Category 1 issues, the generic analysis may be adopted in each plant-specific license renewal review.

A Category 2 classification means that the NUREG–1437 analysis does not meet the criteria of Category 1. Thus, on that particular environmental issue, additional plant-specific review is required and must be analyzed by the license renewal applicant in its ER.

For each license renewal application, the NRC will prepare a draft supplemental EIS (SEIS) to analyze those plant-specific (Category 2) issues. Neither the SEIS nor the ER is required to cover Category 1 issues. However, both are required to consider any new and significant information for Category 1 or unidentified issues. The draft SEIS is made available for public comment. After considering public comments, the NRC will prepare and issue the final SEIS in accordance with 10 CFR 51.91 and 51.93. The final SEIS and NUREG–1437, together, serve as the requisite NEPA analysis for any given license renewal application.

The NUREG–1437 analysis, as shown in Table B–1 of appendix B to subpart A of 10 CFR part 51, found that the environmental impact of the storage of spent nuclear fuel, including high-density storage, in SFPs, during any plant refurbishment or plant operation through the license renewal term, are of a small significance level and meet all Category 1 criteria. It is this finding that the Petitioners challenge. After reviewing the petitions and the public comments received, the NRC has determined that its findings in NUREG–1437 and in Table B–1 remain valid, both for SFP accidents and for potential terrorist attacks that could result in an SFP zirconium fire.

V. Reasons for Denial—General

A. Spent Fuel Pools

Spent nuclear fuel offloaded from a reactor is stored in a SFP. The SFPs at all nuclear plants in the United States are massive, extremely-robust structures designed to safely contain the spent fuel discharged from a nuclear reactor under a variety of normal, off-normal, and hypothetical accident conditions (e.g., loss of electrical power, floods, earthquakes, or tornadoes). SFPs are made of thick, reinforced, concrete walls and floors lined with welded, stainless-steel plates to form a leak-tight barrier. Racks fitted in the SFPs store the fuel assemblies in a controlled configuration (i.e., so that the fuel is both sub-critical and in a coolable geometry). Redundant monitoring, cooling, and makeup-water systems are provided. The spent fuel assemblies are positioned in racks at the bottom of the pool, and are typically covered by at least 25 feet of water. SFPs are essentially passive systems.

The water in the SFPs provides radiation shielding and spent fuel assembly cooling. It also captures radionuclides in case of fuel rod leaks. The water in the pool is circulated through heat exchangers for cooling. Filters capture any radionuclides and other contaminants that get into the water. Makeup water can also be added to the pool to replace water loss.

SFPs are located at reactor sites, typically within the fuel-handling (pressurized-water reactor) or reactor building (boiling-water reactor). From a structural point of view, nuclear power plants are designed to protect against external events such as tornadoes, hurricanes, fires, and floods. These structural features, complemented by the deployment of effective and visible

[1] A note to Table B–1 states that significance levels have not been assigned "for collective off site radiological impacts from the fuel cycle and from high level waste and spent fuel disposal." 10 CFR part 51, subpart A, app. B, Table B–1, n. 2.

physical security protection measures, are also deterrents to terrorist activities. Additionally, the emergency procedures and SAMA guidelines developed for reactor accidents provide a means for mitigating the potential consequences of terrorist attacks.

B. Physical Security

The Petitioners raise the possibility of a successful terrorist attack as increasing the probability of an SFP zirconium fire. As the NAS Report found, the probability of terrorist attacks on SFPs cannot be reliably assessed, quantitatively or comparatively. The NRC has determined, however, that security and mitigation measures the NRC has imposed upon its licensees since September 11, 2001, and national anti-terrorist measures to prevent, for example, aircraft hijackings, coupled with the robust nature of SFPs, make the probability of a successful terrorist attack, though numerically indeterminable, very low.

The NRC's regulations and security orders require licensees to develop security and training plans for NRC review and approval, implement procedures for these plans, and to periodically demonstrate proficiency through tests and exercises.[2] In addition, reactor physical security systems use a defense-in-depth concept, involving the following:

- Vehicle (external) barriers.
- Fences.
- Intrusion detection, alarm, and assessment systems.
- Internal barriers.
- Armed responders.
- Redundant alarm stations with command, control, and communications systems.
- Local law enforcement authority's response to a site and augmentation of the on-site armed response force.
- Security and emergency-preparedness procedure development and planning efforts with local officials.
- Security personnel training and qualification.

The NRC's regulatory approach for maintaining the safety and security of power reactors, and thus SFPs, is based upon robust designs that are coupled with a strategic triad of preventive/protective systems, mitigative systems, and emergency-preparedness and response. Furthermore, each licensee's security functions are integrated and coordinated with reactor operations and emergency response functions. Licensees develop protective strategies in order to meet the NRC design-basis threat (DBT).[3] In addition, other Federal agencies such as the Federal Aviation Administration, the Federal Bureau of Investigation, and the Department of Homeland Security have taken aggressive steps to prevent terrorist attacks in the United States. Taken as a whole, these systems, personnel, and procedures provide reasonable assurance that public health and safety, the environment, and the common defense and security will be adequately protected.

C. Very Low Risk

Risk is defined as the probability of the occurrence of a given event multiplied by the consequences of that event.[4] Studies conducted over the last three decades have consistently shown that the probability of an accident causing a zirconium fire in an SFP to be lower than that for severe reactor accidents. The risk of beyond design-basis accidents (DBAs) in SFPs was first examined as part of the landmark *Reactor Safety Study: An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants* (WASH–1400, NUREG–75/014, 1975), and was found to be several orders of magnitude below those involving the reactor core. The risk of an SFP accident was re-examined in the 1980's as Generic Issue 82, *Beyond Design Basis Accidents in Spent Fuel Pools*, in light of increased use of high-density storage racks and laboratory studies that indicated the possibility of zirconium fire propagation between assemblies in an air-cooled environment. The risk assessment and cost-benefit analyses developed through this effort, NUREG–1353, *Regulatory Analysis for the Resolution of Generic Issue 82, Beyond Design Basis Accidents in Spent Fuel Pools*, Section 6.2, April 1989, concluded that the risk of a severe accident in the SFP was low and "appear[s] to meet" the objectives of the Commission's "Safety Goals for the Operations of Nuclear Power Plants; Policy Statement," (August 4, 1986; 51 FR 28044), as amended (August 21, 1986; 51 FR 30028), and that no new regulatory requirements were warranted.[5]

SFP accident risk was re-assessed in the late 1990s to support a risk-informed rulemaking for permanently shutdown, or decommissioned, nuclear power plants. The study, NUREG–1738, *Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants*, January 2001, conservatively assumed that if the water level in the SFP dropped below the top of the spent fuel, an SFP zirconium fire involving all of the spent fuel would occur, and thereby bounded those conditions associated with air cooling of the fuel (including partial-draindown scenarios) and fire propagation. Even when all events leading to the spent fuel assemblies becoming partially or completely uncovered were assumed to result in an SFP zirconium fire, the study found the risk of an SFP fire to be low and well within the Commission's Safety Goals.

Furthermore, significant additional analyses have been performed since September 11, 2001, that support the view that the risk of a successful terrorist attack (i.e., one that results in an SFP zirconium fire) is very low. These analyses were conducted by the Sandia National Laboratories and are collectively referred to herein as the "Sandia studies."[6] The Sandia studies

[2] For additional related information, please see the NRC fact sheet "NRC Review of Paper on Reducing Hazards From Stored Spent Nuclear Fuel," which is available on the NRC's public Web site at: *http://www.nrc.gov/reading-rm/doc-collections/fact-sheets/reducing-hazards-spent-fuel.html.*

[3] The DBT represents the largest threat against which a private sector facility can be reasonably expected to defend with high assurance. The NRC's DBT rule was published in the **Federal Register** on March 19, 2007 (72 FR 12705).

[4] The American Society of Mechanical Engineers (ASME) "Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications," ASME RA–S–2002, defines risk as the probability and consequences of an event, as expressed by the risk "triplet" that is the answer to the following three questions: (1) What can go wrong? (2) How likely is it? and (3) What are the consequences if it occurs?

[5] The Commission's Safety Goals identified two quantitative objectives concerning mortality risks: (1) The risk to an average individual in the vicinity of a nuclear power plant of prompt fatalities that might result from reactor accidents should not exceed one-tenth of one percent (0.1 percent) of the sum of prompt fatality risks resulting from other accidents in which members of the U.S. population are generally exposed; and (2) The risk to the population in the area near a nuclear power plant of cancer fatalities that might result from nuclear power plant operation should not exceed one-tenth of one percent (0.1 percent) of the sum of cancer fatality risks resulting from all other causes.

[6] Sandia National Laboratories, "Mitigation of Spent Fuel Pool Loss-of-Coolant Inventory Accidents and Extension of Reference Plant Analyses to Other Spent Fuel Pools," Sandia Letter Report, Revision 2 (November 2006) incorporates and summarizes the Sandia Studies. This document is designated "Official Use Only—Security Related Information." A version of the Sandia Studies, with substantial redactions, was made public as a response to a Freedom of Information Act request. It is available on the NRC's Agencywide Document Access and Management System (ADAMS). The redacted version can be found under ADAMS Accession No. ML062290362. For access to ADAMS, contact the NRC Public Document Room Reference staff at 1–800–397–4209, 301–415–4737, or by e-mail to *pdr.resource@nrc.gov*. For additional related information, please see the NRC fact sheet "NRC Review of Paper on Reducing Hazards From Stored Spent Nuclear Fuel," which is available on the NRC's public Web site at: *http://www.nrc.gov/reading-rm/doc-collections/fact-sheets/reducing-hazards-spent-fuel.html.*

are sensitive security related information and are not available to the public. The Sandia studies considered spent fuel loading patterns and other aspects of a pressurized-water reactor SFP and a boiling-water reactor SFP, including the role that the circulation of air plays in the cooling of spent fuel. The Sandia studies indicated that there may be a significant amount of time between the initiating event (i.e., the event that causes the SFP water level to drop) and the spent fuel assemblies becoming partially or completely uncovered. In addition, the Sandia studies indicated that for those hypothetical conditions where air cooling may not be effective in preventing a zirconium fire (i.e., the partial drain down scenario cited by the Petitioners), there is a significant amount of time between the spent fuel becoming uncovered and the possible onset of such a zirconium fire, thereby providing a substantial opportunity for both operator and system event mitigation.

The Sandia studies, which more fully account for relevant heat transfer and fluid flow mechanisms, also indicated that air-cooling of spent fuel would be sufficient to prevent SFP zirconium fires at a point much earlier following fuel offload from the reactor than previously considered (e.g., in NUREG–1738). Thus, the fuel is more easily cooled, and the likelihood of an SFP fire is therefore reduced.

Additional mitigation strategies implemented subsequent to September 11, 2001, enhance spent fuel coolability and the potential to recover SFP water level and cooling prior to a potential SFP zirconium fire. The Sandia studies also confirmed the effectiveness of additional mitigation strategies to maintain spent fuel cooling in the event the pool is drained and its initial water inventory is reduced or lost entirely. Based on this more recent information, and the implementation of additional strategies following September 11, 2001, the probability, and accordingly, the risk, of a SFP zirconium fire initiation is expected to be less than reported in NUREG–1738 and previous studies.

Given the physical robustness of SFPs, the physical security measures, and SFP mitigation measures, and based upon NRC site evaluations of every SFP in the United States, the NRC has determined that the risk of an SFP zirconium fire, whether caused by an accident or a terrorist attack, is very low. As such, the NRC's generic findings in NUREG–1437, as further reflected in Table B–1 of appendix B to subpart A of 10 CFR part 51, remain valid.

VI. Reasons for Denial—NRC Responses to Petitioners' Assertions

A. New and Significant Information

The Petitioners asserted that new and significant information shows that the NRC incorrectly characterized the environmental impacts of spent fuel storage as "insignificant." The information relied upon by the Petitioners, however, is neither "new" nor "significant," within the NRC's definition of those terms. The NRC defines these terms in its Supplement 1 to NRC Regulatory Guide 4.2, *Preparation of Supplemental Environmental Reports for Applications to Renew Nuclear Power Plant Operating Licenses,* Chapter 5 (September 2000) (RG 4.2S1). "New and significant" information, which would require supplementing NUREG–1437, is defined as follows:

> (1) Information that identifies a significant environmental issue that was not considered in NUREG–1437 and, consequently, not codified in Appendix B to Subpart A of 10 CFR Part 51, or
>
> (2) Information that was not considered in the analyses summarized in NUREG–1437 and that leads to an impact finding different from that codified in 10 CFR Part 51.

The Petitioners' "new and significant" information does not meet the RG 4.2S1 criteria. NUREG–1437 (Sections 6.4.6.1. to 6.4.6.3.), and the analyses cited therein, including the NRC's "Waste Confidence Rule" (September 18, 1990; 55 FR 38474, 38480–81), extensively considered the risk of SFP accidents. Moreover, to the extent any information submitted by the Petitioners was not considered in NUREG–1437, none of the information is "significant," because, as explained further in this document, it would not lead to "an impact finding different from that codified in 10 CFR Part 51," or as set forth in NUREG–1437.

B. Spent Fuel Assemblies Will Burn If Uncovered

The Petitioners asserted that new and significant information, consisting primarily of the Thompson Report, NUREG–1738, and a government-sponsored study, the NAS Report, show that spent fuel will burn if the water level in an SFP drops to the point where the tops of the fuel assemblies are uncovered. Specifically, the Petitioners asserted that the NRC fails to recognize the danger of a partial loss of water in an SFP, which in the Petitioners' view, is more likely to cause an SFP zirconium fire than a complete loss of water, because the remaining water will block the circulating air that would otherwise act to cool the spent fuel assemblies.

The NRC does not agree with the Petitioners' assertions. The NRC has determined that a zirconium cladding fire does not occur when only the tops of the fuel assemblies are uncovered. In reality, a zirconium fire cannot occur unless fuel uncovering is more substantial. Even then, the occurrence of a zirconium fire requires a number of conditions which are extremely unlikely to occur together. The Sandia studies provide a more realistic assessment of the coolability of spent fuel under a range of conditions and a better understanding of the actual safety margins than was indicated in NUREG–1738. The Sandia studies have consistently and conclusively shown that the safety margins are much larger than indicated by previous studies such as NUREG–1738.

1. Heat Transfer Mechanisms

Past NRC studies of spent fuel heatup and zirconium fire initiation conservatively did not consider certain natural heat-transfer mechanisms which would serve to limit heatup of the spent fuel assemblies and prevent a zirconium fire. In particular, these studies, including NUREG–1738, did not consider heat transfer from higher-decay-power assemblies to older, lower-decay-power fuel assemblies in the SFP. This heat transfer would substantially increase the effectiveness of air cooling in the event the SFP is drained, far beyond the effectiveness of air cooling cited in past studies. Both the Sandia studies and the NAS Report confirm the NRC conclusion that such heat transfer mechanisms allow rapid heat transfer away from the higher-powered assemblies. The NAS Report also noted that such heat transfer could air-cool the assemblies to prevent a zirconium fire within a relatively short time after the discharge of assemblies from the reactor to the SFP.[7] Thus, air cooling is an effective, passive mechanism for cooling spent fuel assemblies in the pool.

2. Partial Drain-Down

Air cooling is less effective under the special, limited condition where the water level in the SFP drops to a point where water and steam cooling is not sufficient to prevent the fuel from overheating and initiating a zirconium fire, but the water level is high enough to block the full natural circulation of air flow through the assemblies. This condition has been commonly referred to as a partial draindown, and is cited in the Thompson Report. Under those

[7] NAS Report at 53.

conditions, however, it is important to realistically model the heat transfer between high- and low-powered fuel assemblies. The heat transfer from hot fuel assemblies to cooler assemblies will delay the heat-up of assemblies, and allow plant operators time to take additional measures to restore effective cooling to the assemblies. Further, for very low-powered assemblies, the downward flow of air into the assemblies can also serve to cool the assembly even though the full-circulation flow path is blocked. Also, as discussed further in this document, all nuclear plant SFPs have been assessed to identify additional, existing cooling capability and to provide new supplemental cooling capability which could be used during such rare events. This supplemental cooling capability specifically addresses the cooling needs during partial draindown events, and would reduce the probability of a zirconium fire even during those extreme events.

3. License Amendments

In January 2006, the nuclear industry proposed a combination of internal and external strategies to enhance the spent fuel heat removal capability systems at every operating nuclear power plant. The internal strategy implements a diverse SFP makeup system that can supply the required amount of makeup water and SFP spray to remove decay heat. The external strategy involves using an independently-powered, portable, SFP coolant makeup and spray capability system that enhances spray and rapid coolant makeup to mitigate a wide range of possible scenarios that could reduce SFP water levels. In addition, in cases where SFP water levels can not be maintained, leakage control strategies would be considered along with guidance to maximize spray flows to the SFP. Time lines have been developed that include both dispersed and non-dispersed spent fuel storage. The NRC has approved license amendments and issued safety evaluations to incorporate these strategies into the plant licensing bases of all operating nuclear power plants in the United States.

C. Fuel Will Burn Regardless of Its Age

The NRC disagrees with the Petitioners' assertion that fuel will burn regardless of age. Older fuel (fuel which has been discharged from the reactor for a longer time) is more easily cooled and is less likely to ignite because of its lower decay power. A study relied upon by the Petitioners, NUREG–1738, did conservatively assume that spent fuel stored in an SFP, regardless of age, may be potentially vulnerable to a partial drain down event, and that the possibility of a zirconium fire could not be ruled out on a generic basis. This conclusion, however, was in no sense a statement of certainty and was made in order to reach a conclusion on a generic basis, without relying on any plant-specific analyses.

Furthermore, the SFP zirconium fire frequency in NUREG–1738 was predicated on a bounding, conservative assumption that an SFP fire involving all of the spent fuel would occur if the water level in the SFP dropped below the top of the spent fuel. The NUREG–1738 analysis did not attempt to specifically address a number of issues and actions that would substantially reduce the likelihood of a zirconium fire, potentially rendering the frequency estimate to be remote and speculative. For example, NUREG–1738 did not account for the additional time available following the spent fuel being partially or completely uncovered, but prior to the onset of a zirconium fire, that would allow for plant operator actions, makeup of SFP water levels, and other mitigation measures. In addition, NUREG–1738 did not consider the impact of plant and procedure changes implemented as a result of the events of the September 11, 2001, terrorist attacks. NUREG–1738 did clarify that the likelihood of a zirconium fire under such conditions could be reduced by accident management measures, but it was not the purpose of NUREG–1738 to evaluate such accident management measures.

D. SFP Zirconium Fire Will Propagate

Although it is possible that once a spent fuel assembly ignites, the zirconium fire can propagate to other assemblies in the SFP, the NRC has determined (as explained previously) that the risk of an SFP zirconium fire initiation is very low.

E. SFP Zirconium Fire May Be Catastrophic

1. Not New and Significant Information; Very Low Probability

The Massachusetts AG states that "while such a catastrophic accident is unlikely, its probability falls within the range that NRC considers reasonably foreseeable." Thus, the Petitioners asserted that an SFP zirconium fire qualifies as a DBA and, that the impacts of an SFP fire must be discussed in the ER submitted by the licensee and the NRC's EIS, as well as designed against under NRC safety regulations.

The facts that a SFP contains a potentially large inventory of radionuclides and that a release of that material could have adverse effects are not new. These facts are well known, and were considered in the risk evaluation of spent fuel storage contained in NUREG–1738. Even with the numerous conservatisms in the NUREG–1738 study, as described previously, the NRC was able to conclude that the risk from spent fuel storage is low, and is substantially lower than reactor risk.

A study relied upon by the Petitioners, the Thompson Report, claimed that the probability (frequency) of an SFP zirconium fire would be 2E–5 per year[8] for events excluding acts of malice (e.g., terrorism) and 1E–4 per year[9] for acts of malice. With respect to random events (i.e., excluding acts of malice), the NRC concludes that the Thompson Report estimate is overly conservative. A more complete and mechanistic assessment of the event, as described in section VI.E.2. of this Notice, and associated mitigation measures, leads to considerably lower values. With respect to events initiated by a terrorist attack, the NRC concludes that such probability (frequency) estimates are entirely speculative. The NRC also concludes that the additional mitigation measures for SFP events implemented since September 11, 2001, together with the more realistic assessment of spent fuel cooling, indicates that the likelihood of a zirconium fire, though numerically indeterminable, is very low.

The 2E–5 per year estimate for events excluding acts of malice is based on an unsubstantiated assumption that 50 percent of all severe reactor accidents that result in an early release of substantial amounts of radioactive material will also lead to a consequential SFP zirconium fire. The Thompson Report does not identify the necessary sequence of events by which such scenarios might lead to SFP zirconium fires, or discuss the probability of their occurrence. The NRC analysis in the Shearon Harris ASLBP proceeding (described in section VI.E.2. of this Notice) showed that a more complete and mechanistic assessment of the event and associated mitigation measures leads to considerably lower values. This assessment includes the following:

- Frequency and characteristics of the releases from the containment for each release location;
- Transport of gases and fission products within the reactor building;

[8] Two occurrences in 100.000 reactor years.

[9] One occurrence in 10,000 reactor years.

- Resulting thermal and radiation environments in the reactor building, with emphasis on areas in which SFP cooling and makeup equipment is located, and areas in which operator access may be needed to implement response actions;
- Availability/survivability of SFP cooling and makeup equipment in the sequences of concern; and
- Ability and likelihood of successful operator actions to maintain or restore pool cooling or makeup (including consideration of security enhancements and other mitigation measures implemented in response to the terrorist attacks of September 11, 2001).

2. Shearon Harris Atomic Safety and Licensing Board Panel (ASLBP) Proceeding

In the proceeding regarding the expansion of the SFP at the Shearon Harris nuclear power plant, located near Raleigh, North Carolina, the Shearon Harris intervenor described a scenario similar to that raised by the Petitioners, namely, that a severe accident at the adjacent reactor would result in a SFP zirconium fire.[10] The Shearon Harris proceeding considered the probability of a sequence of the following seven events:

a. A degraded core accident.
b. Containment failure or bypass.
c. Loss of SFP cooling.
d. Extreme radiation levels precluding personnel access.
e. Inability to restart cooling or makeup systems due to extreme radiation doses.
f. Loss of most or all pool water through evaporation.
g. Initiation of a zirconium fire in the SFP.

Based on a detailed probabilistic risk assessment, the licensee calculated the probability of a severe reactor accident that causes an SFP zirconium fire to be 2.78E–8 per year. The NRC staff calculated the probability to be 2.0E–7 per year. The intervenor calculated the probability to be 1.6E–5 per year. The ASLBP concluded that the probability of the postulated sequence of events resulting in an SFP zirconium fire was "conservatively in the range described by the Staff: 2.0E–7 per year (two occurrences in 10 million reactor years) or less."[11] Accordingly, the ASLBP found that the occurrence of a severe reactor accident causing an SFP zirconium fire "falls within the category of remote and speculative matters."[12] The Commission affirmed the ASLBP's decision, and the United States Court of Appeals, District of Columbia Circuit, upheld the Commission decision.[13]

In the Shearon Harris proceeding, the intervenor assumed that, given an early containment failure or bypass, a spent fuel zirconium fire would occur (i.e., a conditional probability of 1.0). In order for a reactor accident to lead to a SFP zirconium fire a number of additional conditions must occur. The reactor accident and containment failure must somehow lead to a loss of SFP cooling and must lead to a condition where extreme radiation levels preclude personnel access to take corrective action. There must be then an inability to restart cooling or makeup systems. There must be a loss of significant pool water inventory through evaporation (which can take substantial time). Finally, the event must also lead to a zirconium fire. In contrast to the intervenor's estimate, the licensee and the NRC staff estimated a conditional probability of about one percent that a severe reactor accident with containment failure would lead to a SFP accident. The NRC staff expects that the conditional probability of a SFP zirconium fire, given a severe reactor accident, would be similar to that established in the Shearon Harris proceeding. As such, the probability of a SFP zirconium fire due to a severe reactor accident and subsequent containment failure would be well below the Petitioners' 2E–5 per year estimate.

The 1E–4 per year estimate in the Thompson Report for events involving acts of malice assumes that there would be one attack on the population of U.S. nuclear power plants per century, and that this attack will be 100 percent successful in producing a SFP zirconium fire (thus, fire frequency = 0.01 attack/year × 1.0 fire/attack × 1/104 total reactors = 1E–4/year). The security-related measures and other mitigation measures implemented since September 11, 2001, however, have significantly reduced the likelihood of a successful terrorist attack on a nuclear power plant and its associated SFP. Such measures include actions that would improve the likelihood of the following:

a. Identifying/thwarting the attack before it is initiated.

b. Mitigating the attack before it results in damage to the plant.

c. Mitigating the impact of the plant damage such that an SFP zirconium fire is avoided.

Given the implementation of additional security enhancements and mitigation strategies, as well as further consideration of the factors identified above, the NRC staff concludes that the frequency of SFP zirconium fires due to acts of malice is substantially lower than assumed by the Petitioners.

3. SFP Zirconium Fire Does Not Qualify As a DBA

Regarding the Petitioners' assertion that a SFP zirconium fire qualifies as a design-basis accident (DBA), the NRC staff has concluded that a realistic probability estimate would be very low, such that these events need not be considered as DBAs or discussed in ERs and EISs. Moreover, the set of accidents that must be addressed as part of the design basis has historically evolved from deterministic rather than probabilistic considerations. These considerations, which include defense-in-depth, redundancy, and diversity, are characterized by the use of the single-failure criterion.[14] The single-failure criterion, as a key design and analysis tool, has the direct objective of promoting reliability through the enforced provision of redundancy in those systems which must perform a safety-related function. The single failure criterion is codified in Appendix A and Appendix K to 10 CFR Part 50 and other portions of the regulations. The SFP and related systems have been designed and approved in accordance with this deterministic approach.

F. Intentional Attack on a SFP is "Reasonably Foreseeable."

The Petitioners asserted that an intentional attack targeting a plant's SFP is "reasonably foreseeable." Specifically, the Petitioners raised both the NAS study and the decision by the United States Court of Appeals for the Ninth Circuit, *San Luis Obispo Mothers for Peace* v. *NRC*, 449 F.3d 1016 (9th Cir. 2006), *cert. denied* 127 S. Ct. 1124 (2007), to support the assertion that the NRC's NEPA analysis of a license renewal action for a given facility must include analysis of the environmental impacts associated with a terrorist attack on that facility. The NRC has

[10] *Carolina Power Light Co.*, LBP–01–9, 53 NRC 239, 244–245 (2001).

[11] *Id.*, 53 NRC at 267.

[12] *Id.*, 53 NRC at 268.

[13] *Carolina Power Light Co.*, Commission Law Issuance (CLI)–01–11, 53 NRC 370 (2001), *pet. for review denied, sub nom, Orange County, NC* v. *NRC*, 47 Fed. Appx. 1, 2002 WL 31098379 (D.C. Cir. 2002).

[14] "A single failure means an occurrence which results in the loss of capability of a component to perform its intended safety functions * * * Fluid and electric systems are considered to be designed against an assumed single failure if neither (1) a single failure of any active component * * * nor (2) a single failure of a passive component * * * results in a loss of the capability of the system to perform its safety functions." 10 CFR Part 50, App. A.

considered both the NAS Report and the Ninth Circuit decision, and remains of the view that an analysis of the environmental impacts of a hypothetical terrorist attack on an NRC-licensed facility is not required under NEPA.[15] But, if an analysis of a hypothetical terrorist attack were required under NEPA, the NRC has determined that the environmental impacts of such a terrorist attack would not be significant, because the probability of a successful terrorist attack (i.e., one that causes an SFP zirconium fire, which results in the release of a large amount of radioactive material into the environment) is very low and therefore, within the category of remote and speculative matters.

1. NAS Report

The Petitioners rely, in part, upon the NAS Report, the public version of which was published in 2006 and is available from NAS.[16] In response to a direction in the Conference Committee's Report accompanying the NRC's FY 2004 appropriation,[17] the NRC contracted with NAS for a study on the safety and security of commercial spent nuclear fuel. The NAS made a number of findings and recommendations, including:

- SFPs are necessary at all operating nuclear power plants to store recently discharged fuel;
- Successful terrorist attacks on SFPs, though difficult, are possible;
- The probability of terrorist attacks on spent fuel storage cannot be assessed quantitatively or comparatively;
- If a successful terrorist attack leads to a propagating zirconium cladding fire, it could result in the release of large amounts of radioactive material; and
- Dry cask storage has inherent security advantages over spent fuel storage, but it can only be used to store older spent fuel.

The NAS Report found, and the NRC agrees, that pool storage is required at all operating commercial nuclear power plants to cool newly discharged spent fuel. Freshly discharged spent fuel generates too much decay heat to be placed in a dry storage cask.

The NRC agrees with the NAS finding that the probability of terrorist attacks on spent fuel storage cannot be assessed quantitatively or comparatively. However, the NRC concludes that the additional mitigation measures for SFP events implemented since September 11, 2001, together with a more realistic assessment of spent fuel cooling, as shown by the Sandia studies, indicates that the likelihood of a zirconium fire, though numerically indeterminate, is very low.

Furthermore, the NAS Report states that "[i]t is important to recognize, however, that an attack that damages a power plant or its spent fuel storage facilities would not necessarily result in the release of *any* radioactivity to the environment. There are potential steps that can be taken to lower the potential consequences of such attacks."[18] The NAS Report observed that a number of security improvements at nuclear power plants have been instituted since September 11, 2001, although the NAS did not evaluate the effectiveness and adequacy of these improvements and has called for an independent review of such measures. Nevertheless, the NAS Report states that "the facilities used to store spent fuel at nuclear power plants are very robust. Thus, only attacks that involve the application of large energy impulses or that allow terrorists to gain interior access have any chance of releasing substantial quantities of radioactive material."[19]

As discussed previously, following the terrorist attacks of September 11, 2001, the NRC has required that nuclear power plant licensees implement additional security measures and enhancements the Commission believes have made the likelihood of a successful terrorist attack on an SFP remote.

2. Ninth Circuit Decision

The Petitioners asserted that the NRC should follow the decision of the United States Court of Appeals for the Ninth Circuit, *San Luis Obispo Mothers for Peace* v. *NRC*, 449 F.3d 1016 (9th Cir. 2006), *cert. denied* 127 S. Ct. 1124 (2007), by considering the environmental impacts of intentional attacks on nuclear power plant fuel storage pools in all licensing decisions. The Ninth Circuit held that the NRC could not, under NEPA, categorically refuse to consider the consequences of a terrorist attack against a spent fuel storage facility on the Diablo Canyon reactor site.

The NRC's longstanding view is that NEPA does not require the NRC to consider the environmental consequences of hypothetical terrorist attacks on NRC-licensed facilities. NEPA requires that there be a "reasonably close causal relationship" between the federal agency action and the environmental consequences.[20] The NRC renewal of a nuclear power plant license would not cause a terrorist attack; a terrorist attack would be caused by the terrorists themselves. Thus, the renewal of a nuclear power plant license would not be the "proximate cause" of a terrorist attack on the facility.

If NEPA required the NRC to consider the impacts of a terrorist attack, however, the NRC findings would remain unchanged. As previously described, the NRC has required, and nuclear power plant licensees have implemented, various security and mitigation measures that, along with the robust nature of SFPs, make the probability of a successful terrorist attack (*i.e.*, one that causes an SFP zirconium fire, which results in the release of a large amount of radioactive material into the environment) very low. As such, a successful terrorist attack is within the category of remote and speculative matters for NEPA considerations; it is not "reasonably foreseeable." Thus, on this basis, the NRC finds that the environmental impacts of renewing a nuclear power plant license, in regard to a terrorist attack on an SFP, are not significant.

The NRC has determined that its findings related to the storage of spent nuclear fuel in pools, as set forth in NUREG–1437 and in Table B–1 of Appendix B to Subpart A of 10 CFR Part 51, remain valid. Thus, the NRC has met and continues to meet its obligations under NEPA.

G. SFP Zirconium Fire Should Be Considered Within the Analysis of SAMAs

The Petitioners asserted that SFP fires should be considered within the analysis of severe accident mitigation alternatives (SAMAs). While a large radiological release is still possible, and

[15] In the wake of the Ninth Circuit's *Mothers for Peace* decision, the Commission decided against applying that holding to all licensing proceedings nationwide. *See, e.g., Amergen Energy Co. LLC* (Oyster Creek Nuclear Generating Station), CLI–07–8, 65 NRC 124, 128–29 (2007), *pet. for judicial review pending*, No. 07–2271 (3d Cir.). The Commission will, of course, adhere to the Ninth Circuit decision when considering licensing actions for facilities subject to the jurisdiction of that Circuit. *See id.* Thus, on remand in the Mothers for Peace case itself, the Commission is currently adjudicating intervenors' claim that the NRC Staff has not adequately assessed the environmental consequences of a terrorist attack on the Diablo Canyon Power Plant's proposed facility for storing spent nuclear fuel in dry casks. *See, Pacific Gas & Elec. Co.*, CLI–07–11, 65 NRC 148 (2007). The Commission's ultimate decision in that case will rest on the record developed in the adjudication.

[16] The NRC response to the NAS Report is available at ADAMS Accession No. ML0502804280.

[17] Conference Committee's Report (H. Rept. 108–357) accompanying the *Energy and Water Development Act, 2004* (Pub. L. 108–137, December 3, 2003).

[18] NAS Report at 6 (emphasis in the original).

[19] NAS Report at 30.

[20] *Department of Transportation* v. *Public Citizen*, 541 U.S. 752, 767 (2004) *citing Metropolitan Edison* v. *People Against Nuclear Energy*, 460 U.S. 766, 774 (1983).

was assessed as part of Generic Issue 82, *Beyond Design Basis Accidents in Spent Fuel Pools*, and later, in NUREG–1738, the NRC considers the likelihood of such an event to be lower than that estimated in Generic Issue 82 and NUREG–1738. Based on the Sandia studies, and on the implementation of additional strategies implemented following September 11, 2001, the probability of a SFP zirconium fire is expected to be less than that reported in NUREG–1738 and previous studies. Thus, the very low probability of an SFP zirconium fire would result in an SFP risk level less than that for a reactor accident.

For example, in NUREG–1738, the SFP fire frequencies were conservatively estimated to be in the range of 5.8E–7 per year to 2.4E–6 per year. NUREG–1738 conservatively assumed that if the water level in the SFP dropped below the top of the spent fuel, an SFP zirconium fire involving all of the spent fuel would occur, and thereby bounded those conditions associated with air cooling of the fuel (including partial-drain down scenarios) and zirconium fire propagation. It did not mechanistically analyze the time between the spent fuel assemblies becoming partially or completely uncovered and the onset of a SFP zirconium fire, and the potential to recover SFP cooling and to restore the SFP water level within this time. NUREG–1738 also did not consider the possibility that air-cooling of the spent fuel alone could be sufficient to prevent SFP zirconium fires.

Furthermore, the Sandia studies indicated that air cooling would be much more effective in cooling the spent fuel assemblies. In those cases where air cooling is not effective, the time before fuel heatup and radiological release would be substantially delayed, thus providing a substantial opportunity for successful event mitigation. The Sandia studies, which more fully account for relevant heat transfer and fluid flow mechanisms, also indicated that air-cooling of spent fuel would be sufficient to prevent SFP zirconium fires much earlier following fuel offload than previously considered (e.g., in NUREG–1738), thereby further reducing the likelihood of an SFP zirconium fire. Additional mitigation strategies implemented subsequent to September 11, 2001, will serve to further enhance spent fuel coolability, and the potential to recover SFP cooling or to restore the SFP water level prior to the initiation of an SFP zirconium fire.

Given that the SFP risk level is less than that for a reactor accident, a SAMA that addresses SFP accidents would not be expected to have a significant impact on total risk for the site. Despite the low level of risk from fuel stored in SFPs, additional SFP mitigative measures have been implemented by licensees since September 11, 2001. These mitigative measures further reduce the risk from SFP zirconium fires, and make it even more unlikely that additional SFP safety enhancements could substantially reduce risk or be cost-beneficial.

VII. Denial of Petitions

Based upon its review of the petitions, the NRC has determined that the studies upon which the Petitioners rely do not constitute new and significant information. The NRC has further determined that its findings related to the storage of spent nuclear fuel in pools, as set forth in NUREG–1437 and in Table B–1, of Appendix B to Subpart A of 10 CFR Part 51, remain valid. Thus, the NRC has met and continues to meet its obligations under NEPA. For the reasons discussed previously, the Commission denies PRM–51–10 and PRM–51–12.

Commissioner Gregory B. Jaczko's Dissenting View on the Commission's Decision To Deny Two Petitions for Rulemaking Concerning the Environmental Impacts of High-Density Storage of Spent Nuclear Fuel in Spent Fuel Pools

I disagree with the decision to deny the petition for rulemaking as included in this **Federal Register** notice. In general, I approve of the decision not to initiate a new rulemaking to resolve the petitioners' concerns, but because information in support of the petition will be considered when the staff undertakes the rulemaking to update the Generic Environmental Impact Statement for license renewal, I believe that the decision should have been to partially grant the petition rather than deny it.

The petitioners requested the agency review additional studies regarding spent fuel pool storage they believe would change the agency's current generic determination that the impacts of high-density pool storage are "small". I believe that the agency could commit to reviewing the information provided by the petitioners, along with any other new information, when the agency updates the Generic Environmental Impact Statement (GEIS) for License Renewal in the near future. Regardless of whether or not the information will change the GEIS' conclusions, at a minimum, the agency should be committing to ensure that this information is part of the analysis performed by the staff upon the next update of the GEIS. While we can not predict the outcome of the significance level that will ultimately be assigned to the spent fuel category in the GEIS, it seems an obvious commitment to ensure that the ultimate designation will be appropriately based upon all information available to the staff at the time. Thus, I believe this decision should be explained as a partial granting of the petition. It may not provide the petitioners with everything they want, but it would more clearly state the obvious—that this information, and any other new information, will be reviewed by the agency and appropriately considered when the staff begins its update of the license renewal GEIS.

This specific issue illustrates a larger concern about how the agency handles petitions for rulemaking in general. I find it unfortunate that the agency appears to limit its responses to petitions based upon the vocabulary that has been established surrounding this program. Currently, when the agency discusses these petitions, we discuss them in the context of "granting" or "denying" the rulemaking petitions. We then appear to be less inclined to "grant" unless we are committing to the precise actions requested in the petition. But these petitions are, by their very definition, requests for rulemakings; which means, even if we do "grant" a petition for rulemaking, we can not guarantee a particular outcome for the final rule. The final rulemaking is the result of staff's technical work regarding the rule, public comments on the rule, and resolution of those comments. Rulemaking petitions are opportunities for our stakeholders to provide us with new ideas and approaches for how we regulate. By limiting our responses, we limit our review of the request, and thus, we risk missing many potential opportunities to improve the way we regulate.

Additional Views of the Commission

The Commission does not share Commissioner Jaczko's dissenting view. We appreciate his statement of concern about the petition for rulemaking (PRM) process, but believe these matters are extraneous to the Commission's analyses of the petitioners' technical bases for this particular rulemaking request and, consequently, they had no bearing on the majority view. Specifically, the Commission does not agree that the petitions should be granted in part on the basis of the agency's plan to update the Generic Environmental Impact Statement (GEIS) for License Renewal and make attendant

rule changes in the future. The Commission's detailed statement of reasons for denial of the petitions is the product of a careful review of the petitioners' assertions and other associated public comments, and is supported by the facts before us. In these circumstances, the Commission does not believe the petitioners' request can fairly, or reasonably, be "granted" in part based on a future undertaking which itself had no genesis in the petitioners' requests.

The Commission's timely and decisive action in response to the two petitions serves the interests of the Commission and other participants in an effective, disciplined, and efficient rulemaking petition process. In this instance, a decision now has particular value since it directly addresses the petitioners' statements of significant concern about certain, generic aspects of ongoing and future license renewal reviews. While the analyses performed to respond to these petitions will also undoubtedly inform NRC staff proposals regarding the next update of the GEIS, the Commission does not yet have such proposals before it. Any final Commission decisions on an updated GEIS would be preceded by proposed changes, solicitation of public comment, and evaluation of all pertinent information and public comments. Furthermore, a partial "granting" of the petition could imply that the Commission endorses the petitioners' requests and will give them greater weight than other points of view during the GEIS rulemaking.

As to the other matter raised in Commissioner Jaczko's dissent—that of agency review and disposition of petitions for rulemaking more generally—while petitions for rulemaking are indeed opportunities for stakeholders to suggest new considerations and approaches for regulation, Commissioner Jaczko's general concerns about the agency's process for handling rulemaking petitions go beyond the subject of the Commission's action on these petitions. However, this subject matter is being considered, as the Commission has instructed NRC staff [SRM dated August 6, 2007] to conduct a review of the agency's PRM process. At such time as staff may recommend, as an outgrowth of this review, specific proposals for Commission action which would strengthen the agency PRM process, the Commission will assess such recommendations and act on them, as appropriate.

Dated at Rockville, Maryland, this 1st day of August 2008.

For the Nuclear Regulatory Commission.

Annette L. Vietti-Cook,

Secretary of the Commission.

[FR Doc. E8–18291 Filed 8–7–08; 8:45 am]

BILLING CODE 7590–01–P

DEPARTMENT OF THE INTERIOR

Office of Surface Mining Reclamation and Enforcement

30 CFR Part 901

[SATS No. AL–074–FOR; Docket No. OSM–2008–0015]

Alabama Regulatory Program

AGENCY: Office of Surface Mining Reclamation and Enforcement, Interior.

ACTION: Proposed rule; public comment period and opportunity for public hearing on proposed amendment.

SUMMARY: We, the Office of Surface Mining Reclamation and Enforcement (OSM), are announcing receipt of a proposed amendment to the Alabama regulatory program (Alabama program) under the Surface Mining Control and Reclamation Act of 1977 (SMCRA or the Act). Alabama proposes revisions to its regulations regarding permit fees and civil penalties. Alabama intends to revise its program to improve operational efficiency.

This document gives the times and locations that the Alabama program and proposed amendment to that program are available for your inspection, the comment period during which you may submit written comments on the amendment, and the procedures that we will follow for the public hearing, if one is requested.

DATES: Comments on the proposed rule must be received on or before 4 p.m., c.t., September 8, 2008, to ensure our consideration. If requested, we will hold a public hearing on the amendment on September 2, 2008. We will accept requests to speak at a hearing until 4 p.m., c.t. on August 25, 2008.

ADDRESSES: You may submit comments by either of the following two methods:

- *Federal eRulemaking Portal: http://www.regulations.gov.* The proposed rule is listed under the agency name "OFFICE OF SURFACE MINING RECLAMATION AND ENFORCEMENT" and has been assigned Docket ID: OSM–2008–0015. If you would like to submit comments through the Federal eRulemaking Portal, go to *www.regulations.gov* and do the following. Click on the "Advanced Docket Search" button on the right side of the screen. Type in the Docket ID OSM–2008–0015 and click the submit button at the bottom of the page. The next screen will display the Docket Search Results for the rulemaking. If you click on OSM–2008–0015, you can view the proposed rule and submit a comment. You can also view supporting material and any comments submitted by others.
- *Mail/Hand Delivery/Courier:* Sherry Wilson, Director, Birmingham Field Office, Office of Surface Mining Reclamation and Enforcement, 135 Gemini Circle, Suite 215, Homewood, Alabama 35209. Please include the Docket ID (OSM–2008–0015) with your comments.

We cannot ensure that comments received after the close of the comment period (see **DATES**) or sent to an address other than the two listed above will be included in the docket for this rulemaking and considered.

For additional information on the rulemaking process and the public availability of comments, see "III. Public Comment Procedures" in the **SUPPLEMENTARY INFORMATION** section of this document.

You may receive one free copy of the amendment by contacting OSM's Birmingham Field Office. See below **FOR FURTHER INFORMATION CONTACT**.

You may review a copy of the amendment during regular business hours at the following locations:

Sherry Wilson, Director, Birmingham Field Office, Office of Surface Mining Reclamation and Enforcement, 135 Gemini Circle, Suite 215, Homewood, Alabama 35209, Telephone: (205) 290–7282, *swilson@osmre.gov.*

Randall C. Johnson, Director, Alabama Surface Mining Commission, 1811 Second Avenue, P.O. Box 2390, Jasper, Alabama 35502–2390, Telephone: (205) 221–4130.

FOR FURTHER INFORMATION CONTACT: Sherry Wilson, Director, Birmingham Field Office. Telephone: (205) 290–7282. E-mail: *swilson@osmre.gov.*

SUPPLEMENTARY INFORMATION:

I. Background on the Alabama Program

Section 503(a) of the Act permits a State to assume primacy for the regulation of surface coal mining and reclamation operations on non-Federal and non-Indian lands within its borders by demonstrating that its program includes, among other things, "* * * a State law which provides for the regulation of surface coal mining and reclamation operations in accordance

Schaperow, Jason

From: Schaperow, Jason
Sent: Tuesday, May 22, 2012 3:11 PM
To: Santiago, Patricia
Subject: RE: SFPSS meeting

My issues are as follows:

- Seismic initiator - The analysis does not include a concurrent reactor accident.
- Arrangement of fuel - Peach Bottom uses a 1x8 arrangement of fuel, not the 1x4 arrangement assumed in the study.
- Pool damage - Fukushima shows that an earthquake would not make a hole in a spent fuel pool.
- Mitigation - Peach Bottom-specific mitigation measures are not credited.
- Mitigation - Makeup and spray are likely, because the spent fuel pool is an open system and there is a long time available until draindown and fuel damage. Also, offsite equipment began arriving at Fukushima within about 8 hours (INPO report of November 2011).
- Mitigation - The operators are likely to make openings in the reactor building to aid in spent fuel pool cooling and to prevent a buildup of hydrogen from a concurrent reactor accident.
- Mitigation - For one of the "mitigated" cases, the analysis assumes makeup when spray is needed (and available) to prevent fuel overheating.
- Mitigation - The "unmitigated" cases include some B.5.b mitigation, namely, arranging the fuel in a favorable pattern for cooling.
- Release from clad-pellet gap - The assumed release of cesium (magnitude of 0.05, chemical form CsOH) is conservative.
- Release from fuel pellet - The modeling was validated using in-pile tests for reactor accidents, which is not prototypical of spent fuel pool accidents which progress more slowly and have lower fuel temperatures.
- Hydrogen combustion - A single node is used for the area between the refueling floor the reactor building roof. Simple parametric modeling is used for determining whether there will be a burn.
- Public evacuation - Assuming that we can evacuate tens and even hundreds of thousands of people but we cannot get a couple of people up to the spent fuel pool with a fire hose seems illogical.
- Public evacuation - NRC recommended a 50-mile evacuation for Fukushima.
- Public evacuation - MELCOR and MACCS analysis was used for developing evacuation and relocation assumptions, instead of RASCAL.
- Results - The consequence/risk results presented in the study assume the probability of mitigation is zero.

-----Original Message-----
From: Santiago, Patricia
Sent: Tuesday, May 22, 2012 1:30 PM
To: Schaperow, Jason
Subject: RE: SFPSS meeting

can you detail your issues for me again quick.
thanks

From: Schaperow, Jason
Sent: Tuesday, May 22, 2012 1:09 PM
To: Santiago, Patricia
Subject: SFPSS meeting

Per your request, I attended the meeting with Jennifer Uhle and Randy Sullivan on SFPSS at 11:00 a.m. today. The other attendees were Katie Wagner, Scott Burnell, and Don Helton. The objective of the meeting was for

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Randy to communicate NSIR's issues on SFPSS at the office level. Randy described the NSIR issues, and Jennifer listened. Jennifer asked Don about the possibility of doing an HRA to respond to the NSIR issues.

I said that you had asked me to help coordinate the NSIR issues. I then mentioned I had some issues of my own. Jennifer asked me to describe my issues, so I did.

No decisions were reached at the meeting. The meeting's objective was for Jennifer to hear directly the NSIR views.

Thanks,
Jason

Mrowca, Lynn

From: Schaperow, Jason
Sent: Wednesday, July 11, 2012 4:32 PM
To: Powell, Eric
Cc: Mrowca, Lynn
Subject: Spent Fuel Pool Scoping Study
Attachments: RE: SFPSS meeting

Hi Eric,

Lynn indicated that you were following the RES effort on the Spent Fuel Pool Scoping Study. I was involved in the study during the months of May and June 2012. Attached is a list of comments I gave to the RES staff working on the study.

Could you give me a call or stop by?

Thanks,
Jason

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May 22, 2012

Spent Fuel Pool Scoping Study

The SFPSS is different from the security assessments for spent fuel pools because it assesses earthquake frequency, equipment and spent fuel pool fragility, source term, public evacuation, and offsite radiological consequences.

Modeling area	**Conservative assumption**
Seismic initiator	The analysis does not include a concurrent reactor accident.
Arrangement of fuel	Peach Bottom uses a 1x8 arrangement of fuel, not the 1x4 arrangement assumed in the study.
Pool damage	Fukushima shows that an earthquake would not make a hole in a spent fuel pool.
Mitigation	Peach Bottom-specific mitigation measures are not credited.
Mitigation	Makeup and spray are likely, because the spent fuel pool is an open system and there is a long time available until draindown and fuel damage. Also, offsite equipment began arriving at Fukushima within about 8 hours (INPO report of November 2011).
Mitigation	The operators are likely to make openings in the reactor building to aid in spent fuel pool cooling and to prevent a buildup of hydrogen from a concurrent reactor accident.
Mitigation	For one of the "mitigated" cases, the analysis assumes makeup when spray is needed (and available) to prevent fuel overheating.
Mitigation	The "unmitigated" cases include some B.5.b mitigation, namely, arranging the fuel in a favorable pattern for cooling.
Release from clad-pellet gap	The assumed release of cesium (magnitude of 0.05, chemical form CsOH) is conservative.
Release from fuel pellet	The modeling was validated using in-pile tests for reactor accidents, which is not prototypical of spent fuel pool accidents which progress more slowly and have lower fuel temperatures.
Hydrogen combustion	A single node is used for the area between the refueling floor the reactor building roof. Simple parametric modeling is used for determining whether there will be a burn.
Public evacuation	Assuming that we can evacuate tens and even hundreds of thousands of people but we cannot get a couple of people up to the spent fuel pool with a fire hose seems illogical.
Public evacuation	NRC recommended a 50-mile evacuation for Fukushima.
Public evacuation	MELCOR and MACCS analysis was used for developing evacuation and relocation assumptions, instead of RASCAL.
Results	The consequence/risk results presented in the study assume the probability of mitigation is zero.

Schaperow, Jason

From: Schaperow, Jason
Sent: Tuesday, September 04, 2012 12:56 PM
To: Mrowca, Lynn
Subject: RE: ACTION??: comments on SOW for SFPSS HRA.doc
Attachments: comments on SOW for SFPSS HRA and RES feedback.doc

I am resending this to correct the date in the attachment.

>>
Hi Lynn,

Today, I had a phone call with James Chang and shared my comments with him verbally. I noted his responses in the attached list of my comments.

Thanks,
Jason

From: Ader, Charles
Sent: Friday, August 31, 2012 11:05 AM
To: Mrowca, Lynn; Hayes(NRO), Michelle
Cc: Schaperow, Jason
Subject: RE: ACTION??: comments on SOW for SFPSS HRA.doc

Go ahead and share as informal comments.

From: Mrowca, Lynn
Sent: Thursday, August 30, 2012 7:28 PM
To: Ader, Charles; Hayes(NRO), Michelle
Cc: Schaperow, Jason
Subject: ACTION??: comments on SOW for SFPSS HRA.doc
Importance: High

Charlie/Michelle,

Jason and I discussed his comments on the SOW for the spent fuel scoping study HRA. Key points include considering realism in the project and including the improvements required by recent Commission Orders. I also chatted with Kevin Coyne tonight. The working group is already well into this work. He said it would be OK to share our comments informally with James Chang.

ACTION: Would you like us to pass these on as "formal" comments from NRO or have Jason talk to James? The request was for comments by 8/31/12 (Friday).

Thanks!

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September 4, 2012

Comments on SOW for SFPSS HRA

Revise the project objective to clarify that the goal is to develop realistic estimates of failure probabilities.
RES feedback: The estimates will not be precise. The estimates will not be conservative.

Revise the project objective to include developing a distribution and central estimate for the failure probabilities.
RES feedback: We are not planning to develop distributions.

Revise Task 1 to identify the SSCs that are likely to be damaged, instead of SSCs that could be damaged.
RES feedback: We are using fragility estimates from the external events PRA for Peach Bottom.

To avoid developing conservative estimates, pick the mid-point of the earthquake size bin (of 0.5 to 1.0g) for identifying the SSCs that are likely to be damaged.
RES feedback: We are using 0.7g.

Reduce the scope of the study by removing the case where the spent fuel pool does not have a hole.
RES feedback: We need to analyze this case for completeness.

Consider the assistance of onsite and offsite organizations (e.g., TSC, EOF, INPO, NRC) to mitigate the accident.
RES feedback: We are excluding offsite assistance.

Consider the range of mitigation equipment that could be used to flood or spray the spent fuel pool. This includes offsite fire trucks and concrete pump trucks.
RES feedback: We are excluding offsite assistance.

Include the improvements being required by the Commission under orders EA-12-049, Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events, and EA-12-051, Order to Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation. Also, include the improvements being made as a result of the Request for Information on Emergency Preparedness (NTTF Recommendation 9.3) dated March 12, 2012.
RES feedback: We are excluding these improvements, because the completion date for the HRA is September 2012.

Employ seismic and structural experts to develop realistic estimates of plant damage caused by the initiating event. Use realistic estimates of the fragilities of SSCs.
RES feedback: We are using fragility estimates from the external events PRA for Peach Bottom.

Consider Peach Bottom plant-specific mitigation measures and procedures.
RES feedback: We went to Peach Bottom in July 2012 to get plant-specific information.

Schaperow, Jason

From: Schaperow, Jason
Sent: Thursday, September 20, 2012 11:49 AM
To: Mrowca, Lynn; Powell, Eric
Subject: RE: HRA for the Spent Fuel Pool Scoping Study

Hi Lynn and Eric,

Today, I participated in a meeting on the HRA for the Spent Fuel Pool Scoping Study. Sean Peters led the meeting, and the other participants were Jeff Mitman, James Chang, Mark Thaggard, Bob Kahler, and Randy Sullivan. Sean wanted to better understand the comments from the program offices (and where the disagreements were) so he could then take them up with senior management in RES. Today's meeting was a follow-on to the meeting discussed in the email below.

The following issues were discussed today:

- No credit for additional personnel onsite during the refueling outage for diagnosing that the pool had a hole. The assumed time to diagnose the leak is too long (conservative).
- No credit for reliable spent fuel pool instrumentation required by Order EA-12-051.
- No credit for offsite resources (e.g., fire engines, concrete pump trucks) being brought onsite.
- No consideration of the mitigation strategy of early positioning of fire hoses and spray nozzles on the refueling floor (prior to water level draining to the top of the fuel) and later connecting to fire trucks or other pumps at ground level.
- Assuming core damage and containment failure (at 3.5 hours) in some cases.
- Only mitigation that is proceduralized is credited.
- The success criteria are conservative. Mitigating actions would continue throughout the event and may stop or truncate a release.
- Only mitigation from the refueling floor is credited.

Sean said that RES would provide to the program offices on September 30 the draft report documenting the HRA.

Jason

From: Schaperow, Jason
Sent: Friday, September 14, 2012 1:20 PM
To: Mrowca, Lynn
Cc: Powell, Eric
Subject: HRA for the Spent Fuel Pool Scoping Study

Hi Lynn,

Today, I participated (by phone) in a meeting on the HRA for the Spent Fuel Pool Scoping Study. Other participants included Robert Kahler (NSIR) and Kevin Witt (NRR/JLD).

James Chang (RES) walked us through the attached slides. The following issues were mentioned/discussed:

- Assuming that 1 foot of water was immediately lost from the pool due to sloshing.
- No credit for additional personnel onsite during the refueling outage for diagnosing that the pool had a hole.
- No credit for reliable spent fuel pool instrumentation required by Order EA-12-051.
- No credit for offsite resources (e.g., fire engines) being brought onsite.

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- No consideration of the mitigation strategy of early positioning of fire hoses and spray nozzles on the refueling floor (prior to water level draining to the top of the fuel) and later connecting to fire trucks or other pumps at ground level.
- Assuming that the water source of the emergency cooling water basin could not be used. (This water source was credited in the SOARCA study.)
- Assuming core damage and containment failure (at 3.5 hours) in some cases.

RES discussed providing the draft report to NSIR and NRR for review/concurrence. RES did not mention NRO review/concurrence.

Jason

Schaperow, Jason

Subject: update on Spent Fuel Pool Scoping Study
Location: Lynn's office

Start: Mon 10/22/2012 10:00 AM
End: Mon 10/22/2012 10:30 AM
Show Time As: Tentative

Recurrence: (none)

Meeting Status: Not yet responded

Organizer: Schaperow, Jason
Required Attendees: Mrowca, Lynn; Powell, Eric

I request to meet to discuss the Spent Fuel Pool Scoping Study.

On October 19, Eric and I attended a briefing by RES that was intended for Division Directors of NRR, NRO, and NSIR. The briefing's objective was to identify the issues that need to be resolved before the HRA can be briefed to the ACRS. The SES managers that participated were Charles Ader, Mark Lombard, and Richard Correia. (Other SES managers may have participated, but they were either on the phone bridge or I didn't recognize them.)

Status
The HRA for the Spent Fuel Pool Scoping Study has now been completed.
Issues
The HRA is conservative to the point that it may not be useful for characterizing results or guiding policy. For example:

- Overestimates of time required to detect the draindown and add water to the pool.
- Underestimates of water spray and injection flow rates into the pool.
- HRA concludes that new spent fuel pool instrumentation required by Order EA-12-051 does not affect the likelihood of mitigation.

RES appears to be expanding the scope of the study to include simultaneous severe reactor accident and spent fuel pool draindown for one of the two operating reactors at Peach Bottom – MELCOR/MACCS calculations and HRA

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Powell, Eric

From: Powell, Eric
Sent: Friday, February 08, 2013 11:28 AM
To: Algama, Don
Cc: Schaperow, Jason
Subject: RE: SFPSS: IOWG Document Review and Comment
Attachments: IOWG_CompiledComments.xlsx

Don,

I wish I had more time, but attached are my comments and I already forwarded you Jason's comments.

-Eric

From: Algama, Don
Sent: Wednesday, February 06, 2013 12:54 PM
To: Powell, Eric
Subject: RE: SFPSS: IOWG Document Review and Comment

Eric:

End of the week is OK. We are in are comment resolution period, but can accommodate your comments.

Thanks,
Don A.

From: Powell, Eric
Sent: Wednesday, February 06, 2013 11:43 AM
To: Algama, Don
Subject: FW: SFPSS: IOWG Document Review and Comment

Don,

Please see the comments below. Unfortunately, I was not able to review the document, because I've been swamped the past several days. If there is a little wiggle room for accepting comments, like the end of the week, I can give it a look and provide comments. Just let me know. If not, I understand.

-Eric

From: Schaperow, Jason
Sent: Wednesday, February 06, 2013 8:38 AM
To: Powell, Eric
Cc: Mrowca, Lynn
Subject: RE: SFPSS: IOWG Document Review and Comment

The following are my comments on the Spent Fuel Pool Scoping Study:

1. The objectives of the study are not clearly stated in the Executive Summary. I recommend listing the objectives in bullet form.

2. This study would benefit from an independent peer review.

Thanks,

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Jason

Inter-Office Working Group Compiled Comments on Frozen SFPSS D (IWOG Comments in black, helpful info in blue)					
Office	**Received**	**Affected Chapter**	**Name**	**Comment**	**SFPPS Ch. Lead**
NSIR	01/22/13	4		(Pg. 59) "Given the estimated width, length and depth for each localized liner tear and their number, it is still necessary to estimate the leakage rate through these tears. Estimation of this flow rate uses the following assumptions (1) the flow rate can be estimated using an equation similar to that used for flow through the concrete cracks and (2) the friction factor for that equation can be calculated on the basis of test results for leakage rates through cracks in pipes. These assumptions are not validated at this time. Therefore, considerable uncertainty exists for the resulting leakage rate estimate." This seems to say the cracks in the SFP caused by the seismic event were not caluated but assumed based on "These assumptions are not validated at this time	Jose P.
NRO	02/09/13	Abstract	Eric Powell	(Pg.ii) The opening sentences says "best-estimate" I do not think that is an accurate description of the study, because many bounding assumptions were made.	
NRO	02/09/13	Executive Summary	Eric Powell	(pg. v) "Similarly, the selection of a site that has a separate SFP for each reactor (as opposed to a shared pool) is also not intended to suggest that these situations are inherently more vulnerable." It was never stated that a site that has a separate SFP for each reactor was choosen. It seems odd to make this statement without declaring that it was choosen first. Same comment in	

Inter-Office Working Group Compiled Comments on Frozen SFPSS D (IWOG Comments in black, helpful info in blue)					
Office	**Received**	**Affected Chapter**	**Name**	**Comment**	**SFPPS Ch. Lead**
NRO	02/09/13	Executive Summary	Eric Powell	(Pg. vii) #15, is this supposed to answer the question on whether operators should expidite transfer of fuel from the SPF to dry cask storage? If so, that should be clearer. Use similar language to what is used in the abstract. If not, there should be another bullet that says the conclusion with regards to that issue. (if I remember correctly, the study didnt find evidence to support expiditing the transfer of fuel)	
NRO	02/09/13	Introduction and Background	Eric Powell	(Pg. 1) For facilities licensed to operate an independent spent fuel storage installation (ISFSI), the fuel assemblies are later loaded into casks and moved to the ISFSI. Adding a when this is, either time or some qualifier about when the fuel has cooled, would be beneficial.	
NRO	02/09/13	Introduction and Background	Eric Powell	(Pg. 1) Now, let us consider some less-obvious considerations. The list below presents considerations from the perspective of the pros and cons associated with postulated transitioning from the existing use of high-density racking in the United States back to the use of low-density storage. The list is subdivided into two parts—those considerations that are covered within this study and those that are not. Should say something about dry cask storage, because that's what we are talking about.	

Inter-Office Working Group Compiled Comments on Frozen SFPSS D (IWOG Comments in black, helpful info in blue)					
Office	**Received**	**Affected Chapter**	**Name**	**Comment**	**SFPPS Ch. Lead**
NRO	02/09/13	Introduction and Background	Eric Powell	(Pg 2-3) The reader may quickly note that the first set of considerations are generally pros associated with expedited fuel movement to casks, while the latter considerations are generally cons. Why focus on the pros for this study? The agency's position—that spent fuel storage in either pools or casks is safe—is based on a number of past studies and regulatory activities that are discussed later in this chapter. This regulatory position is solid, but we are re-examinging this topic due to potential changes in the state-of-knowledge and stakeholder interest. In reassessing this position, we have started by investigating whether any of the "pros" are more compelling than past studies suggest. If they are, then the issue can be addressed more holistically to see if new information challenges the existing regulatory position. Otherwise, there is insufficient motivation to spend the additional agency resources associated with a more holistic study, and these resources are better devoted to other aspects of the agency's mission of protecting people and the environment. **This paragraph stands out as a little too colloquial (e.g. "the reader may quickly..." and "this regulatory position is solid...."). Also, the last sentence should be deleted or reworded. Although it is true it sounds odd to say it in this report.**	

ocument	
Disposition	**Status**
(Note: Comment not based of official IOWG SFPSS Report)	[redacted]

ocument	
Disposition	Status

ocument	
Disposition	**Status**

Schaperow, Jason

From: Schaperow, Jason
Sent: Tuesday, February 19, 2013 5:02 PM
To: Witt, Kevin
Cc: Mrowca, Lynn
Subject: ~~OFFICIAL USE ONLY~~ - Blue Ribbon Commission - spent fuel safety and security
Attachments: BRC 110203.rev1.ppt; BRC 110328_p4 (3) Rev1.ppt

Hi Kevin,

I appreciate your calling me to let me know about the NAS asking about how we responded to their recommendations on security improvements for spent fuel pools.

Here is what I remember:

The NAS study on Spent Fuel Safety and Security was performed in the 2003 – 2004 time frame. I am pretty sure that Phil Brochman wrote the letter that Chairman Diaz sent to Congress on March 14, 2005, responding to the NAS study. Here is a link to Chairman Diaz's letter (without the classified attachment): http://www.nrc.gov/reading-rm/doc-collections/congress-docs/correspondence/2005/domenici-03142005.pdf

On February 3, 2011, we briefed the Blue Ribbon Commission on America's Nuclear Future. Allison Macfarlane was a member of the Blue Ribbon Commission. The briefing was on Spent Fuel Safety and Security. Because it was a security-related briefing, we gave it at the Naval Reactors offices at the Washington Navy Yard. Several NRC staff members gave parts of the briefing, including Dan Dorman, Eric Bowman, Phil Brochman, and me. My briefing material is attached – I briefed them on severe accident mitigation.

One of the members of the Blue Ribbon Commission (Lee Hamilton) was unable to attend the February 3, 2011, briefing. So, we went back to the Navy Yard on March 28, 2011, and briefed him. This time Catherine Haney gave the entire briefing. She was accompanied by other staff members including me. The severe accident mitigation part of the briefing was included in her briefing material, which is attached.

I hope this is helpful.

Jason

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McBride, Mark

From: Schaaf, Robert
Sent: Wednesday, March 20, 2013 10:24 AM
To: McBride, Mark
Subject: RE: Waste Confidence Rule EIS: Copy of Appendix E for Review

Mark,

Sorry, all that was shared with us was the pdf. Note that it's likely that Appendix E has undergone substantial changes at this point based on the earlier meeting with the directorate, so I wouldn't suggest putting too much effort into the version you have on hand. Not sure whether we'll see another preliminary draft before the concurrence review copy comes to us – will let you know.

Bob

From: McBride, Mark
Sent: Wednesday, March 20, 2013 10:00 AM
To: Schaaf, Robert
Subject: Waste Confidence Rule EIS: Copy of Appendix E for Review

As discussed at our last meeting on 3/11/13, I am giving Appendix E a more thorough review than was possible before the meeting. I have only a rather poor .pdf copy of the appendix – Would it be possible to get a Word copy to make commenting easier?

-- Mark

Mark McBride - Hydrologist
NRC - Hydrology and Meteorology Branch (NRO/DSEA/RHMB)
Office T7F32, Mail Stop T7E18
(301) 415-0670

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Schaperow, Jason

From: Mrowca, Lynn
Sent: Friday, March 22, 2013 5:03 PM
To: Schaperow, Jason
Subject: FYI: Permanent Relocation Presentation
Attachments: Untitled.pdf

From: Ader, Charles
Sent: Friday, March 22, 2013 4:42 PM
To: Holahan, Gary
Cc: Hawkins, Kimberly; Mrowca, Lynn
Subject: FW: Permanent Relocation Presentation

FYI – my suggestion to RES on a different way to present this type of information. Probably too late since the earlier report went to the ACRS last summer, but worth considering for any future report.

From: Ader, Charles
Sent: Friday, March 22, 2013 4:41 PM
To: Gibson, Kathy
Subject: Permanent Relocation Presentation

Kathy,

This is an example of how the results were presented in NUREG/CR-6295, Reassessment of Selected Factors Affecting Siting of Nuclear Power Plants.

Same information, but a different way to present.

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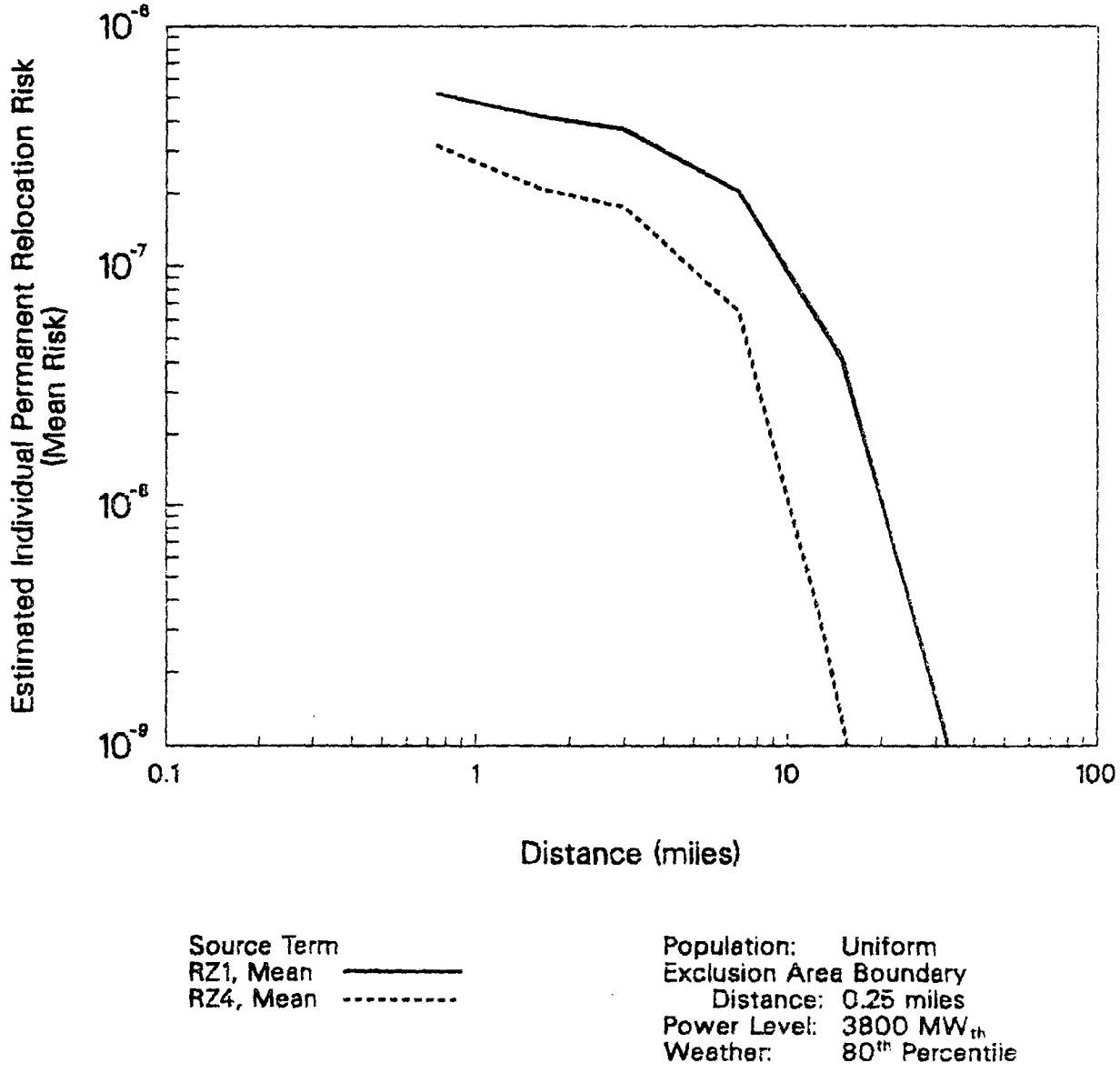

Figure 4.7 Estimated Individual Risk of Permanent Relocation vs. Distance (Mean) (1/yr)

Schaperow, Jason

From: Schaperow, Jason
Sent: Friday, March 22, 2013 1:41 PM
To: Ader, Charles; Hawkins, Kimberly
Cc: Mrowca, Lynn; Powell, Eric; Clark, Theresa
Subject: Revised comments on Spent Fuel Pool Scoping Study
Attachments: SFPSS comments 03 22 2013.doc

Based on discussions I had today with Lynn Mrowca and Hossein Esmaili, I have revised my comments on the Spent Fuel Pool Scoping Study. My revised comments are attached. Lynn is reviewing my revised comments and, as a result, they may be further refined.

Thanks,
Jason

From: Schaperow, Jason
Sent: Friday, March 22, 2013 9:02 AM
To: Ader, Charles; Hawkins, Kimberly
Cc: Mrowca, Lynn; Powell, Eric; Clark, Theresa
Subject: Comments on Spent Fuel Pool Scoping Study

Hi Charlie and Kim,

Per Lynn's request, I am forwarding you my comments on the Spent Fuel Pool Scoping Study. These are the same the same comments I left in your (Charlie's) in-basket late yesterday.

Thanks,
Jason

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March 22, 2013

Spent Fuel Pool Scoping Study comments

1. SRM-SECY-08-0029 directed the SOARCA study to use individual cancer fatality risk as its latent cancer health-effects metric. I recommend that the Spent Fuel Pool Scoping Study (SFPSS) follow the same approach by using this metric and not reporting the total number of cancer deaths.

2. A memorandum to the Commission dated April 3, 2007 (OUO-SII), stated that the staff would not report land contamination/economic consequences in SOARCA because of modeling and policy issues. SRM-COMPBL-08-0002/COMGBJ-08-0003 directed the staff to develop an improved economic consequence model for MACCS. It also stated that the resulting model may be applied to the SOARCA results if so directed by the Commission. I recommend that the SFPSS follow the same approach by not reporting land contamination.

3. Chapter 11, conclusion 5, footnote 43 gives the timeframe during which the fuel cannot be cooled by air. I recommend that the Information Security Branch of NSIR be consulted to confirm that this information is not security-related SUNSI, because the SFPSS is intended to be made publicly available.

4. Chapter 11, conclusion 6 seems to imply that the additional spent fuel pool instrumentation required by Order EA-12-051 is not effective for mitigating spent fuel pool accidents. I recommend that text be added to this conclusion to explain the technical basis for it.

5. Chapter 11, conclusion 7 seems to imply that the additional mitigation capabilities required by Order EA-12-049 were not credited in the study. I recommend that the addition mitigation capabilities required by Order EA-12-049 be credited to improve the study's realism.

6. Chapter 11, conclusion 16 states the study demonstrates that past SFP risk estimates from large seismic events are similar to this study for most consequence metrics. I recommend that text be added to this conclusion to explain the technical basis for it.

Schaperow, Jason

From: Ader, Charles
Sent: Saturday, March 23, 2013 10:30 PM
To: Schaperow, Jason; Hawkins, Kimberly; Mrowca, Lynn; Powell, Eric; Clark, Theresa
Subject: SFPSS comments 03 22 2013.doc
Attachments: SFPSS comments 03 22 2013.doc

Jason,

Gave your second comment some more thought and unclear on how this comment was implemented in SOARCA. Not sure how latent fatalities are calculated in you can't consider land contamination (ground shine, relocation criteria, reentry criteria, etc.). Not clear what is different between SFPSS and SOARCA other than reporting a number of acres.

I did not have a chance to review the report, but these comments look reasonable. I noted a few comments.

Need to discuss with RES the basis for not following the Commission guidance on SOARCA. If clearly not following Commission guidance then I question sending to ACRS and then you probably need to raise with Gary. Also assess the difficulty for RES to revise to follow the Commission guidance.

I would discuss with Gary before final "yea or nay" goes to RES

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March 22, 2013

Spent Fuel Pool Scoping Study comments

1. SRM-SECY-08-0029 directed the SOARCA study to use individual cancer fatality risk as its latent cancer health-effects metric. I recommend that the Spent Fuel Pool Scoping Study (SFPSS) follow the same approach by using this metric and not reporting the total number of cancer deaths.

Comment [c1]: Suggest citing at least one *example where this was done.*

2. A memorandum to the Commission dated April 3, 2007 (OUO-SII), stated that the staff would not report land contamination/economic consequences in SOARCA because of modeling and policy issues. SRM-COMPBL-08-0002/COMGBJ-08-0003 directed the staff to develop an improved economic consequence model for MACCS. It also stated that the resulting model may be applied to the SOARCA results if so directed by the Commission. I recommend that the SFPSS follow the same approach by not reporting land contamination.

Comment [c2]: Probably too late since the earlier report has it, but unless Commission guidance changed, would suggest deleting in this version, or raising to the Commission the question.

3. Chapter 11, conclusion 5, footnote 43 gives the timeframe during which the fuel cannot be cooled by air. I recommend that the Information Security Branch of NSIR be consulted to confirm that this information is not security-related SUNSI, because the SFPSS is intended to be made publicly available.

4. Chapter 11, conclusion 6 seems to imply that the additional spent fuel pool instrumentation required by Order EA-12-051 is not effective for mitigating spent fuel pool accidents. I recommend that text be added to this conclusion to explain the technical basis for it.

5. Chapter 11, conclusion 7 seems to imply that the additional mitigation capabilities required by Order EA-12-049 were not credited in the study. I recommend that the addition mitigation capabilities required by Order EA-12-049 be credited to improve the study's realism.

6. Chapter 11, conclusion 16 states the study demonstrates that past SFP risk estimates from large seismic events are similar to this study for most consequence metrics. I recommend that text be added to this conclusion to explain the technical basis for it.

McBride, Mark

From: Palmrose, Donald
Sent: Monday, March 25, 2013 12:34 PM
To: Giacinto, Joseph
Cc: McBride, Mark
Subject: RE: DEP comments so far on Appendix E

Joe,

I need any comments back to the WCD Directorate by COB Wednesday. So pass on to Mark for both of you to pass on this. At this point, I'll only ask you two for comments if Dan Barnhurst sees anything significant.

Thanks for the quick reply,
Don

From: Giacinto, Joseph
Sent: Monday, March 25, 2013 12:30 PM
To: Palmrose, Donald
Subject: RE: DEP comments so far on Appendix E

Don, Mark and I are at training through Wednesday. Thursday would be OK for me – I am taking Friday off and I believe Mark is off Friday too.

Joe

From: Palmrose, Donald
Sent: Monday, March 25, 2013 8:44 AM
To: Barnhurst, Daniel; McBride, Mark; Giacinto, Joseph
Subject: DEP comments so far on Appendix E

Gents,

The attachment has my comments so far on a new version of Appendix E for the WCD GEIS. Please review and I would like to know if we can meet Tuesday afternoon to discuss this document.

Thanks,
Don

Donald Palmrose, PhD
Sr. Project Manager
NRO/DSEA/RENV
301-415-3803
T7-F38

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Schaperow, Jason

From: Schaperow, Jason
Sent: Monday, March 25, 2013 12:59 PM
To: Hawkins, Kimberly; Mrowca, Lynn
Cc: Powell, Eric; Clark, Theresa
Subject: DSRA proposed comments on the SFPSS
Attachments: DSRA proposed comments on the SFPSS 03 25 2013.doc

Hi Kim,

Per your request, attached is an electronic copy of the DSRA proposed comments on the SFPSS. I added a statement at the beginning of the attachment to note the risk-communication challenge posed by the SFPSS.

I hope this meets the need.

Thanks,
Jason

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March 25, 2013

DSRA Proposed Comments on the Spent Fuel Pool Scoping Study

Recognizing the risk-communication challenge posed by the Spent Fuel Pool Scoping Study, DSRA offers the following comments to help address this challenge:

1. SRM-SECY-08-0029 directed the SOARCA study to use individual cancer fatality risk as its latent cancer health-effects metric. DSRA recommends that the Spent Fuel Pool Scoping Study (SFPSS) follow the same approach by using this metric and not reporting the total number of cancer deaths. For example, Chapter 7, Table 29 reports total latent cancer fatalities per year. Also, Chapter 11, conclusion 11 states "For scenarios with large releases, significant numbers of latent cancer fatalities are predicted when using a dose-response model based on the linear-no threshold hypothesis; however, this would be a small fraction compared to cancer fatalities from all causes."

2. A memorandum to the Commission dated April 3, 2007 (OUO-SII), stated that the staff would not report land contamination/economic consequences in SOARCA because of modeling and policy issues. SRM-COMPBL-08-0002/COMGBJ-08-0003 directed the staff to develop an improved economic consequence model for MACCS. This SRM also stated that the resulting model may be applied to the SOARCA results if so directed by the Commission. DSRA recommends that the SFPSS follow the same approach by not reporting land contamination.

3. Chapter 11, conclusion 5, footnote 43 gives the timeframe during which the fuel cannot be cooled by air. DSRA recommends that the Information Security Branch of NSIR be consulted to confirm that this information is not security-related SUNSI, because the SFPSS is intended to be made publicly available.

4. Chapter 11, conclusion 6 seems to imply that the additional spent fuel pool instrumentation required by Order EA-12-051 is not effective for mitigating spent fuel pool accidents. DSRA recommends that text be added to this conclusion to explain the technical basis for it.

5. Chapter 11, conclusion 7 seems to imply that the additional mitigation capabilities required by Order EA-12-049 were not credited in the study. DSRA recommends that the additional mitigation capabilities required by Order EA-12-049 be credited to improve the study's realism.

6. Chapter 11, conclusion 16 states the study demonstrates that past SFP risk estimates from large seismic events are similar to this study for most consequence metrics. DSRA recommends that text be added to this conclusion to explain the technical basis for it.

From: Tabatabai, Sarah
To: Chokshi, Nilesh; Flanders, Scott; Jackson, Diane
Cc: Karas, Rebecca
Subject: SFPSS comments
Date: Thursday, March 28, 2013 11:21:00 AM
Attachments: SFPSS Doc (ground motion related comments).docx
Importance: High

I've attached the comments for your review before sending to Theresa (by noon).
Thanks.
--Sarah

Comment 1

Table 3 (last table entry on page 19):

Vertical spectral accelerations as high as horizontal accelerations are justified on the bases that nearby earthquakes control the ground motions spectra for this event and that the frequencies of interest for the study are frequencies near or above 10 Hz.

In sentence above, you state that nearby earthquakes control the ground motions for this event and that the frequencies of interest for this study are near or above 10 Hz. What information is this conclusion based on? Is it based on the SSE PGA? Specifically, on page 33 (2nd paragraph) you state that "The PGA for the reference plant SSE is 0.12 g. (This is about a magnitude 5.3 earthquake at about 25 km.)" Is this controlling earthquake consistent with USGS 2008 hazard results? What annual exceedance frequency does this controlling earthquake correspond to?

Comment 2

Table 3 (first table entry on page 20):

The current seismic assessment uses a model and code generated by the US Geological Survey (USGS, 2008). The USGS 2008 information is being further developed and updated by a group of stakeholders, including the NRC, in a collaborative study which includes (a) the seismic source zone characterization, and (b) the ground motion attenuation models. In addition, the NRC is developing independent methods and computer codes, which will be publicly available when completed, to combine (a) and (b). Although part (a) of this updating effort has been completed in early 2012, part (b) and the computer code development are still ongoing. Therefore, this study used the earlier USGS information instead of the ongoing update program.

The sentence "The USGS 2008 information is being further developed and updated by a group of stakeholders, including the NRC, in a collaborative study which includes (a) the seismic source zone characterization, and (b) the ground motion attenuation models" is incorrect. Instead of saying "The USGS 2008 information is being further developed and updated by a group of stakeholders, including the NRC, in a collaborative study" should reference the CEUS SSC model (which is a new seismic source model independent from the USGS 2008 model). Also change ground motion attenuation models to "ground motion prediction equations (GMPEs)". The GMPE update effort was not part of the CEUS SSC model and it is an industry effort, which is still in progress.

In addition, add a sentence to justify the use of the USGS 2008 model for the purposes of this scoping study, since the USGS hazard model is not endorsed by the NRC in licensing new reactors (currently the CEUS SSC is the NRC approved starting model). Also need to add a disclaimer saying something like the use of the USGS hazard is not consistent with how new reactors are licensed.

Comment 3

Table 3 (first table entry on page 22)

In general, for an aftershock to cause subsequent additional damage to a structure, it would have to occur much closer to the site than the main event and with characteristics, for example frequency content, that would make the structure especially vulnerable to it. The earthquake ground motion considered in the SFP scoping study is a probabilistic quantity that aggregates motions from events with various magnitudes and distances to the site. For this site, this probabilistic ground motion already tends to be controlled by relatively close events in the larger magnitude range for the credible seismic sources. . This main shock cracks the SFP studied but its structure is still stable after the earthquake and it cracks in a manner that allows for additional loading cycles at this level. Under these conditions, earthquake ground motions greater than those for the main shock would be needed to further damage the SFP. This is unlikely given that the ground motion considered is already controlled by close events with magnitudes near the credible upper magnitudes for the site.

There is no discussion of the controlling earthquakes for the site in the report. Is the information on controlling earthquakes in the above paragraph based on the USGS 2008 hazard results? There should be some discussion in the text regarding describing how the controlling earthquakes and the associated annual exceedance frequencies. Furthermore, aftershocks can be numerous and substantial (especially if we are considering very low probability events). Also, the aftershocks could in fact be closer to the site than the main shock and that could be significant since according to above paragraph "ground motion considered is already controlled by close events...".

Comment 4

Section 3.1 (page 29, 2nd paragraph)

The USGS 2008 model is being further developed and updated by a group of stakeholders, including the NRC, in a collaborative study which includes (a) the seismic source zone characterization, and (b) the ground motion attenuation models. In addition, the NRC is developing independent methods and computer codes, which will be publicly available when completed, to combine (a) and (b). Although part (a) of this updating effort has been completed (NRC, 2012b), part (b) and the computer code development are still ongoing. Therefore, this study used the earlier USGS model instead of the ongoing update program.

See comment 2

Comment 5

Section 3.1 (page 29, 3rd paragraph)

Comparisons of hazard estimates for the reference site, a rock site, obtained with those three information sources are graphically shown in **Error! Reference source not found.** *(PGA) and in* **Error! Reference source not found.** *(1, 5 and 10 Hz spectral acceleration), which supports the following observations:*

The reference site selected for this study is a rock site, however what are the implications of a soil site and possible high frequency resonances? Also, what are the implications of sites with different controlling earthquakes. How do spent fuel pool characteristics vary between different operating plant and what are the implications of this?

Comment 6

Section 3.1 (page 29, paragraphs 4 to 6)

These bullets compare the USGS 2008 hazard estimates for the reference site with the LLNL and EPRI results. What is the purpose of these comparisons? It seems more appropriate to justify using the USGS 2008 model and compare to CEUS SSC results (e.g. the CEUS SSC report compares USGS 2008 with CEUS SSC model at several demonstration sites).

Comment 7

Section 3.1 (Figures 4 and 5, page 31)

Should indicate in the figure captions that these are hard rock hazard curves.

Comment 8

Section 3.2 (last paragraph page 33)

In addition to the PGA, ground motions at a site are also characterized by their frequency content expressed in terms of response spectra. Based on the USGS 2008 model, a uniform hazard site Ground Motion Response Spectrum (GMRS) (NRC, 2007b) was derived for the GI-199 study and used in this study.

It is incorrect to combine term uniform hazard response spectra with ground motion response spectra. In addition, the related footnote states that that the term GMRS has a specific meaning in the context of Regulatory Guide 1.208 (NRC, 2007b). In this report, the term GMRS is used more generally. Please describe how the response spectra for the selected site was developed and if it is not consistent with the definition of the GMRS in RG 1.208, then use a different name. Is the response spectra for the reference site shown in Figure 7 a uniform hazard response spectra?

Comment 9

Section 3.3 (1st and 2nd paragraphs, page 34)

In first paragraph change "Peach Bottom" to "reference site" and in second paragraph rename GMRS if not consistent with RG 1.208.

Comment 10

Page 35 (2nd paragraph)

Vertical spectral accelerations and the vertical PGA are taken to be the same as the horizontal spectral accelerations and PGA. This is assumed on the bases that nearby earthquakes would control the ground shaking spectra for this event and that the frequencies of interest for this study are frequencies above 5 Hz (ASCE, 1999) (McGuire, Silva and Costantino, 2001).

How were controlling earthquakes determined?

Comment 11

Page 35 (Section 3.3, 2nd paragraph)

Other ground motion response spectra of interest for this study are the free-field response spectra used in the seismic PRA for the NUREG-1150 study (Lambright et al., 1990). **Error! Reference source not found.** *provides a comparison of the frequency content of the horizontal response spectra (5-percent damping) for the SSE, the median response spectrum used in NUREG-1150 study, and the site GMRS based on the USGS 2008 model. For this comparison, all spectra are scaled to a PGA of 1.0 g. When the three response spectra under consideration are scaled to the same PGA, the information in* **Error! Reference source not found.** *supports the following observations:*

Please clarify which response spectra will actually be used in the structural analysis described later on in the report. It's not clear from in the text. In addition, if GMRS is not consistent with the RG 1.208 definition of site GMRS, then need to use an alternative description.

Comment 12

Page 38 (Bullet 1)

Obtain free-field GMRS (horizontal and vertical) for the site considered (a rock site and a reactor building with small embedment).

Make sure that the definition of GMRS is either consistent with RG 1.208 or use a different terminology to distinguish it from the GMRS. The GMRS is referred to in Section 4 as well. Furthermore, what is the link between 1) and 2). As written, it appears that there is no connection between the response spectra obtained in step 1) and the ISRS obtained in step 2).

Comment 13

Page 38 (Bullet 2)

Change "Peach Bottom" to "reference site"

Correa, Yessie

From: Clark, Theresa
Sent: Friday, March 29, 2013 2:12 PM
To: Pope, Tia; Algama, Don; Gibson, Kathy; Richards, Stuart; Lee, Richard
Cc: RidsNroMailCenter Resource; Penny, Melissa; Hawkins, Kimberly; Ader, Charles; Schaperow, Jason; Mrowca, Lynn; Flanders, Scott; Chokshi, Nilesh; Tabatabai, Sarah; Jackson, Diane; Shams, Mohamed; Shuaibi, Mohammed; Berrios, Ilka; Willingham, Michael; Bergman, Thomas; Mayfield, Michael; Matthews, David; Powell, Eric; Tegeler, Bret; Holahan, Gary
Subject: NRO Response: YT-2013-0053 - Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor [OUO]
Attachments: spent fuel pool scoping study (NRO comments - OUO).doc

Good afternoon,

The attached file provides high-level **comments from NRO divisions on the spent fuel pool scoping study**, based on the relatively short time available for review. This does not reflect division concurrence. The comments should be addressed before sending the report to ACRS, and we suggest that any comments that are not addressed be shared with ACRS. Incorporation of these comments should help with the risk-communication challenge posed by presenting the information in this report. We look forward to further office review after the ACRS subcommittee meeting.

This completes action on **YT-2103-0053**.

--

Theresa Valentine Clark
Technical Assistant
Division of Safety Systems and Risk Assessment
U.S. NRC Office of New Reactors
T-10F10 | 301-415-4048
Theresa.Clark@nrc.gov

From: RidsNroMailCenter Resource
Sent: Tuesday, March 19, 2013 12:09 PM
To: Penny, Melissa; Hamilton, LaJuan
Cc: Clark, Theresa; Berrios, Ilka; Correa, Yessie; RidsNroMailCenter Resource
Subject: ACTION W/ SUB TASK: YT-2013-0053 - Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor
Importance: High

DSRA (Lead) w/ DE (sub task)
YT-2013-0053 - Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor

Due Dates:
DE (YT-2013-0053-A) to DSRA – **3/26/13**
DSRA (YT-2013-0053) to RES – **3/29/13**

Please see additional details and guidance below.

Thank you,
NRO Correspondence Team
Electronic Distribution Only

From: Pope, Tia
Sent: Wednesday, March 13, 2013 3:56 PM

34

NRO Division-Level Comments on "Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor" March 2013

The comments below represent a high-level review of the "Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor" dated March 2013 (also known as the spent fuel pool (SFP) scoping study) by divisions in the Office of New Reactors. These comments should be addressed before sending the report to the Advisory Committee on Reactor Safeguards (ACRS). It is suggested that any comments that are not addressed be shared with the ACRS. Incorporation of these comments should help with the risk-communication challenge posed by presenting the information in this report.

1. The report needs to describe how its results could be useful in making regulatory decisions on matters including the Japan lessons-learned Tier 3 recommendation on assessment of the transfer of spent fuel to dry-cask storage and recent Commission direction on economic consequences. In responding to this comment, a fuller characterization of the purpose and usefulness of the report should be added, including an explanation of how the study's point-estimate approach is appropriate in the context described above.

2. The report needs to describe the relationship between the study results and our current approach to approving nuclear power plant sites and designs. In addition to describing this approach, a column could be added to the assumptions in Chapter 2 to provide context relative to the current regulatory approach for licensing nuclear power plants and plants' licensing bases. Accordingly, the conclusions could also be reframed to highlight the robustness of our regulatory framework for the safe operation of nuclear power plants, e.g., that mitigation strategies provide a significant reduction in release rates.

3. The Staff Requirements Memorandum (SRM) on SECY-08-0029 directed the State-of-the-Art Reactor Consequence Analyses (SOARCA) to use individual cancer fatality risk as its latent cancer health-effects metric. The study should follow the same approach by using this metric and not reporting the total number of cancer deaths. For example, Chapter 7, Table 29 reports total latent cancer fatalities per year. Also, Chapter 11, conclusion 11 states "For scenarios with large releases, significant numbers of latent cancer fatalities are predicted when using a dose-response model based on the linear-no threshold hypothesis; however, this would be a small fraction compared to cancer fatalities from all causes."

4. A memorandum to the Commission dated April 3, 2007 (OUO-SII), stated that the staff would not report land contamination/economic consequences in SOARCA because of modeling and policy issues. SRM-COMPBL-08-0002/COMGBJ-08-0003 directed the staff to develop an improved economic consequence model for the MELCOR Accident Consequence Code System (MACCS). This SRM also stated that the resulting model may be applied to the SOARCA results if so directed by the Commission. The study should follow the same approach by not reporting land contamination.

5. Table 3 (the last entry on page 19) includes this sentence: "Vertical spectral accelerations as high as horizontal accelerations are justified on the bases that nearby

earthquakes control the ground motions spectra for this event and that the frequencies of interest for the study are frequencies near or above 10 Hz." Provide the basis for the assumption that nearby earthquakes control the estimated ground motions at the reference site.

6. Table 3 (the first entry on page 20) includes this paragraph:

The current seismic assessment uses a model and code generated by the US Geological Survey (USGS, 2008). The USGS 2008 information is being further developed and updated by a group of stakeholders, including the NRC, in a collaborative study which includes (a) the seismic source zone characterization, and (b) the ground motion attenuation models. In addition, the NRC is developing independent methods and computer codes, which will be publicly available when completed, to combine (a) and (b). Although part (a) of this updating effort has been completed in early 2012, part (b) and the computer code development are still ongoing. Therefore, this study used the earlier USGS information instead of the ongoing update program.

   a. It seems that the intent of this paragraph is to reference the recently published Central and Eastern United States Seismic Source Characterization (CEUS SSC) model. Instead of saying: "The USGS 2008 information is being further developed and updated by a group of stakeholders, including the NRC, in a collaborative study," the paragraph should reference the CEUS SSC model and note that it is a new seismic source model cosponsored by EPRI, DOE, and NRC. Also, clarify that CEUS SSC is independent of the USGS 2008 model.

   b. Change "ground motion attenuation models" to "ground motion prediction equations (GMPEs)" and make the distinction that the GMPE update effort was not part of the CEUS SSC model and it is an industry effort, which is still in progress.

   c. Add a sentence to justify the use of the USGS 2008 model for the purposes of this scoping study, since the USGS hazard model is not endorsed by the NRC in licensing new reactors (currently the CEUS SSC model is the NRC approved starting model).

   d. Add a disclaimer stating that the use of the USGS hazard is not consistent with the hazard defined in the licensing basis for new reactors.

   This comment also applies to Section 3.1 (page 29, 2nd paragraph).

7. Table 3 (the first entry on page 22) includes this paragraph:

In general, for an aftershock to cause subsequent additional damage to a structure, it would have to occur much closer to the site than the main event and with characteristics, for example frequency content, that would make the structure especially vulnerable to it. The earthquake ground motion considered in the SFP scoping study is a probabilistic quantity that aggregates motions from events with various magnitudes and distances to the site. For this site, this probabilistic ground motion already tends to be controlled by relatively close events in the larger magnitude range for the credible seismic sources. . This main shock cracks the SFP studied but its structure is still stable after the earthquake and it cracks in a manner that allows for additional loading cycles at this

level. Under these conditions, earthquake ground motions greater than those for the main shock would be needed to further damage the SFP. This is unlikely given that the ground motion considered is already controlled by close events with magnitudes near the credible upper magnitudes for the site.

It would be better to just state that current probabilistic seismic hazard analysis (PSHA) models do not consider aftershocks and that is why they were not considered in this study. Otherwise the statements in the above paragraph would lead to the following comments that should be clarified:

a. There is no discussion on the controlling earthquakes and the associated annual exceedance frequencies to support the statement that "[f]or this site, this probabilistic ground motion already tends to be controlled by relatively close events in the larger magnitude range for the credible seismic sources."

b. Aftershocks can be numerous and substantial (especially if the study is considering very low probability events).

c. Aftershocks could in fact be closer to the site than the main shock, and that could be significant since the report stated previously that the estimated ground motions at the reference site are controlled by nearby events.

8. Section 3.1 (page 29, 3rd paragraph) mentions the hazard estimates for a rock site. The report should discuss the implications for soil sites, as well as the implications of sites with different controlling earthquakes. Clarify how SFP characteristics vary between different operating plants and what are the implications of this variation.

9. Section 3.1 (page 29, paragraphs 4 to 6) includes bullets that compare the USGS 2008 hazard estimates for the reference site with the LLNL and EPRI results. The report should clarify the purpose of these comparisons.

10. Section 3.1 (page 31, Figures 4 and 5) should indicate in the figure captions that these are hard rock hazard curves.

11. Section 3.2 (page 33, last paragraph) includes this statement: "In addition to the PGA, ground motions at a site are also characterized by their frequency content expressed in terms of response spectra. Based on the USGS 2008 model, a uniform hazard site Ground Motion Response Spectrum (GMRS) (NRC, 2007b) was derived for the GI-199 study and used in this study." It is incorrect to combine the term uniform hazard response spectra with the term GMRS. In addition, Footnote 5 states that "the term GMRS has a specific meaning in the context of Regulatory Guide (RG) 1.208 (NRC, 2007b). In this report, the term GMRS is used more generally." The report should describe how the response spectrum for the selected site was developed. If it is not consistent with the definition of the GMRS in RG 1.208, then use a different name. Clarify whether the response spectrum for the reference site shown in Figure 7 is a uniform hazard response spectrum. In addition, do a global search for "GMRS" because it is used throughout the report.

12. In Section 3.3 (page 34, 1st and 2nd paragraphs), change "Peach Bottom" to "reference site" and do a global search for further changes because "Peach Bottom" appears in

multiple places.

13. The second paragraph on page 35 includes this statement:

Vertical spectral accelerations and the vertical PGA are taken to be the same as the horizontal spectral accelerations and PGA. This is assumed on the bases that nearby earthquakes would control the ground shaking spectra for this event and that the frequencies of interest for this study are frequencies above 5 Hz (ASCE, 1999) (McGuire, Silva and Costantino, 2001).

The report should describe how controlling earthquakes were determined.

14. Section 3.3 (page 35, 2nd paragraph) describes other "ground motion response spectra of interest for this study." Clarify which response spectra were used in the structural analysis described later in the report.

15. Chapter 11, conclusion 5, footnote 43 gives the timeframe during which the fuel cannot be cooled by air. The Information Security Branch of NSIR should be consulted to confirm that this information is not security-related sensitive unclassified non-safeguards information, because the study is intended to be made publicly available.

16. Chapter 11, conclusion 6 seems to imply that the additional spent fuel pool instrumentation required by Order EA-12-051 is not effective for mitigating spent fuel pool accidents. Text should be added to this conclusion to explain its technical basis.

17. Chapter 11, conclusion 7 seems to imply that the additional mitigation capabilities required by Order EA-12-049 were not credited in the study. The additional mitigation capabilities required by Order EA-12-049 should be credited to improve the study's realism.

18. Chapter 11, conclusion 16 states the study demonstrates that past spent fuel pool risk estimates from large seismic events are similar to this study for most consequence metrics. Text should be added to this conclusion to explain its technical basis.

Ader, Charles

From: Ader, Charles
Sent: Wednesday, May 01, 2013 5:36 PM
To: Gibson, Kathy
Subject: FW: Permanent Relocation Presentation - SFPSS
Attachments: Untitled.pdf

Kathy,

Back in March I sent you a suggestion for an alternate way to present results of the SFPSS which you said you would consider. During yesterday's briefing of the JLD, in responding to one of the questions, I thought you indicated that there were problems presenting the results on an individual risk basis, although you do have tables of consequences/year. If there are problems with this approach, I would be interested in the concerns.

Two observations from yesterday, one of which may become an additional comment and the other regards how results are presented (separate from my early comment). If, as Brian said, the purpose of the study is to help inform a decision of whether to move fuel from a pool, then I would only include the mitigated case, not the unmitigated case. (SOARCA had an objective to show the effectiveness of the b.5.b measures, which is not an objective of this study). A best estimate would be one case that assumes operators will attempt to mitigate, with a high likelihood of success (some smaller likelihood of failure). Your slide 13 should have the last box titled something like "conditional probability of "hot" fuel" and then another box titled "conditional probability of failure of mitigation" of ??. The other comment regards presenting total numbers from this event without putting it in the context of total numbers for individuals impacted by a 1 in 60,000 year initiating event (i.e., what is the impact of the seismic event on surrounding populations, likely displaced individuals due to chemical hazards, fires, etc.) It is only part of the story and rather misleading presentation of the impact of a rather significant seismic event. Permanent land condemnation may be more appropriate than interdiction.

I would be happy to discuss further.

From: Ader, Charles
Sent: Friday, March 22, 2013 4:41 PM
To: Gibson, Kathy
Subject: Permanent Relocation Presentation

Kathy,

This is an example of how the results were presented in NUREG/CR-6295, Reassessment of Selected Factors Affecting Siting of Nuclear Power Plants.

Same information, but a different way to present.

35

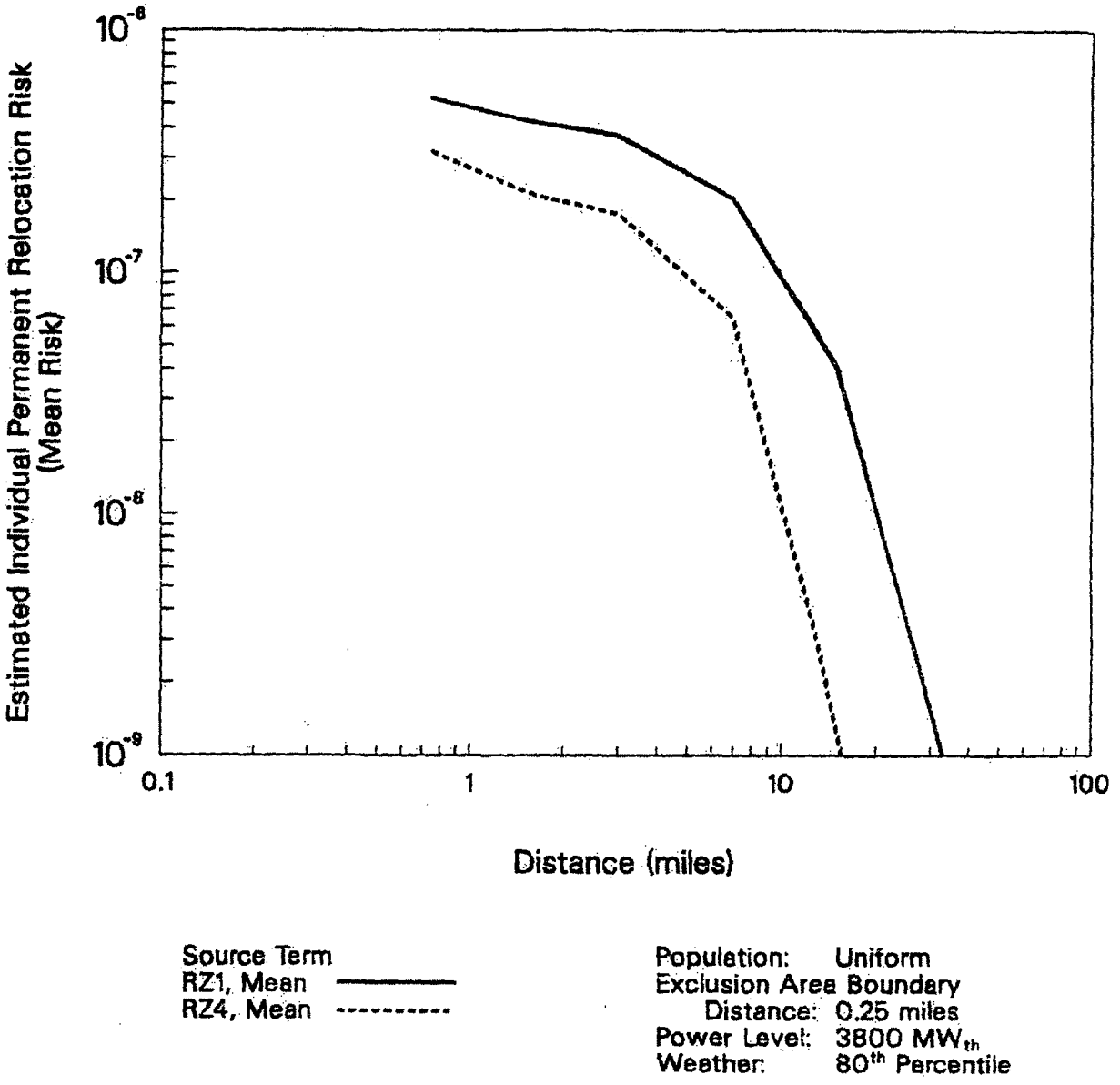

Figure 4.7 Estimated Individual Risk of Permanent Relocation vs. Distance (Mean) (1/yr)

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Protecting People and the Environment

Spent Fuel Pool (SFP) Scoping Study

Office of Nuclear Regulatory Research

January 18, 2012

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Overview

- Focus: reexamination of the potential advantages associated with moving older fuel stored in the SFP to dry cask storage in an expedited manner
- Study to be completed by: June 2012
- Explicit consideration of past SFP safety and security studies in project planning/conduct
  - Resolution of GI-82 (1989)
  - NUREG-1738 (2001)
  - Post-9/11 SFP Security Assessments

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Spent Fuel Pool Scoping Study

- **Technical Objectives**:
  - Estimating the effect on SFP accident consequences of a reduced amount of spent nuclear fuel being stored in the SFP based on realistic analysis using readily available and technically defensible deterministic methods and assumptions as well as probabilistic insights.
  - Capturing the relevant technical considerations and findings in a form that is conducive to informing deliberation on the issue of SFP accident consequences and whether to expedite movement of spent fuel to dry storage while not addressing any specific regulatory user need.
  - Expediting attainment of preliminary insights by (1) relying exclusively on available information and methods, (2) focusing on a single SFP/operating cycle, and (3) using past experience to narrow the scope to a particular scenario of interest.

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Technical Approach

- Two conditions to be considered:
  - Representative of the current situation for the selected site (i.e., high-density loading and a relatively full SFP)
  - Representative of expedited movement of older fuel to a dry cask storage facility (i.e., low-density loading)
- Elements of the study include
  - A single site/single operating cycle
  - Seismic and structural assessments based on available information to define initial and boundary conditions
  - SCALE analysis of reactor building dose rates
  - MELCOR accident progression analysis (effectiveness of mitigation, fission product release, etc.)
  - Emergency planning assessment
  - MACCS2 offsite consequence analysis (land contamination and health effects)
  - Probabilistic considerations

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Boundary Condition Specification

- Basic scenario prescribed
  - Seismic event: 0.5 g to 1.0 g Peak Ground Acceleration (PGA)
    - Challenging but very low frequency occurrence (PGA > 6 times the SSE and beyond the seismic design basis for Eastern US plants)
    - USGS hazard models (2008) being used as starting model for ground motion response spectra
  - Peach Bottom Unit #3 (PB3)
  - 5 operating cycle snapshots
    - e.g., latter half of refueling outage represented by 13 days since reactor shutdown with fuel in a non-dispersed configuration

Note: Considers consequences associated with the SFP itself (i.e., consequences associated with fuel cask handling, transportation, and storage are not within the scope)

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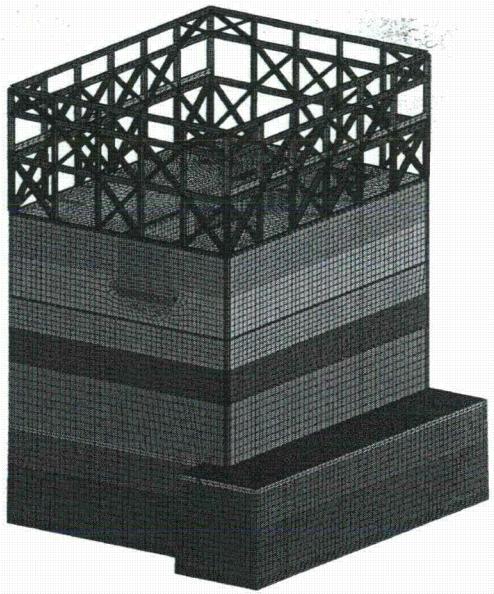
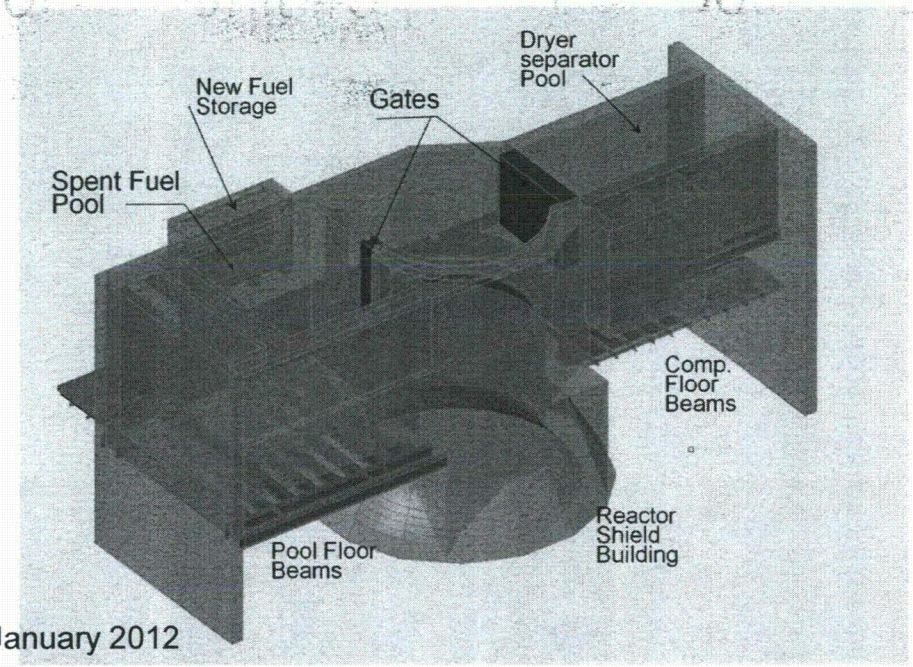
Seismic & Structural Inputs

- Seismic Assessment
  - Challenging damage not expected below seismic bin 3 (0.5 – 1.0 g)
    - Level of 0.71g studied - Challenging but very low frequency of occurrence
  - USGS hazard assessments (2008) as starting model for ground motion response spectra (GMRS)
- Structural Assessment - To determine starting point for subsequent accident progression analysis
  - Damage to the spent fuel pool structure and liner
  - Other damage states: loss of water from sloshing; damage to SFP penetrations, reactor building and other relevant structures, racks and fuel, and reactor shutdown cooling systems





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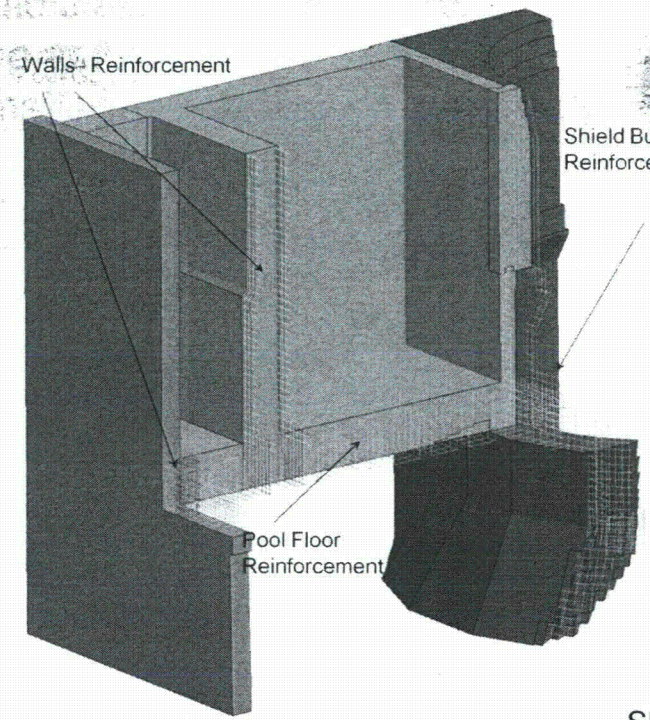
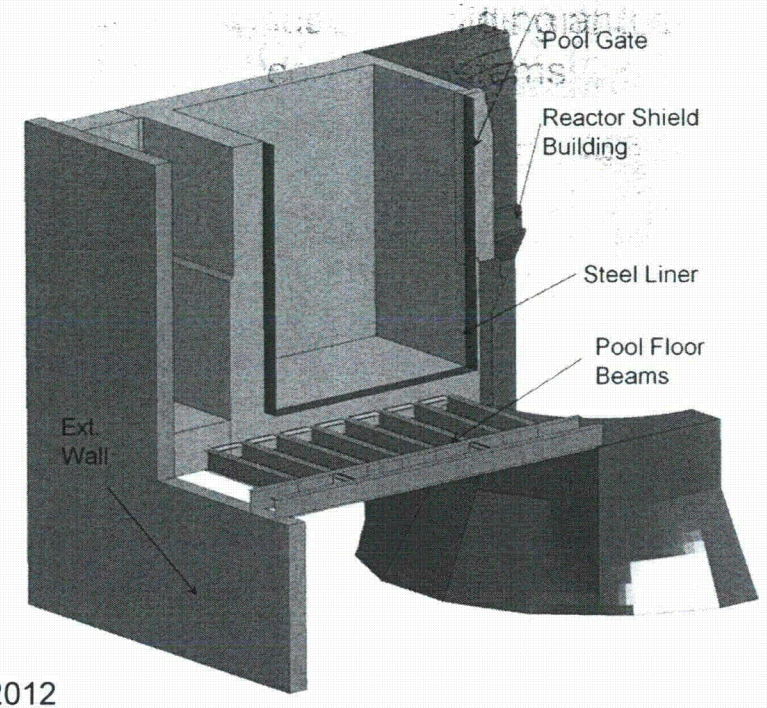
Seismic & Structural Inputs

- Structural Assessment
  - Follows the general approach used for GI-82 (NUREG/CR-5176) together with scaled in-structure spectra from the NUREG-1150 PRA for calculation of demands
  - Finite element analysis to calculate SFP frequencies, verify hydrodynamic loads and calculate distortions of the SFP structure, concrete cracking, and liner strains.





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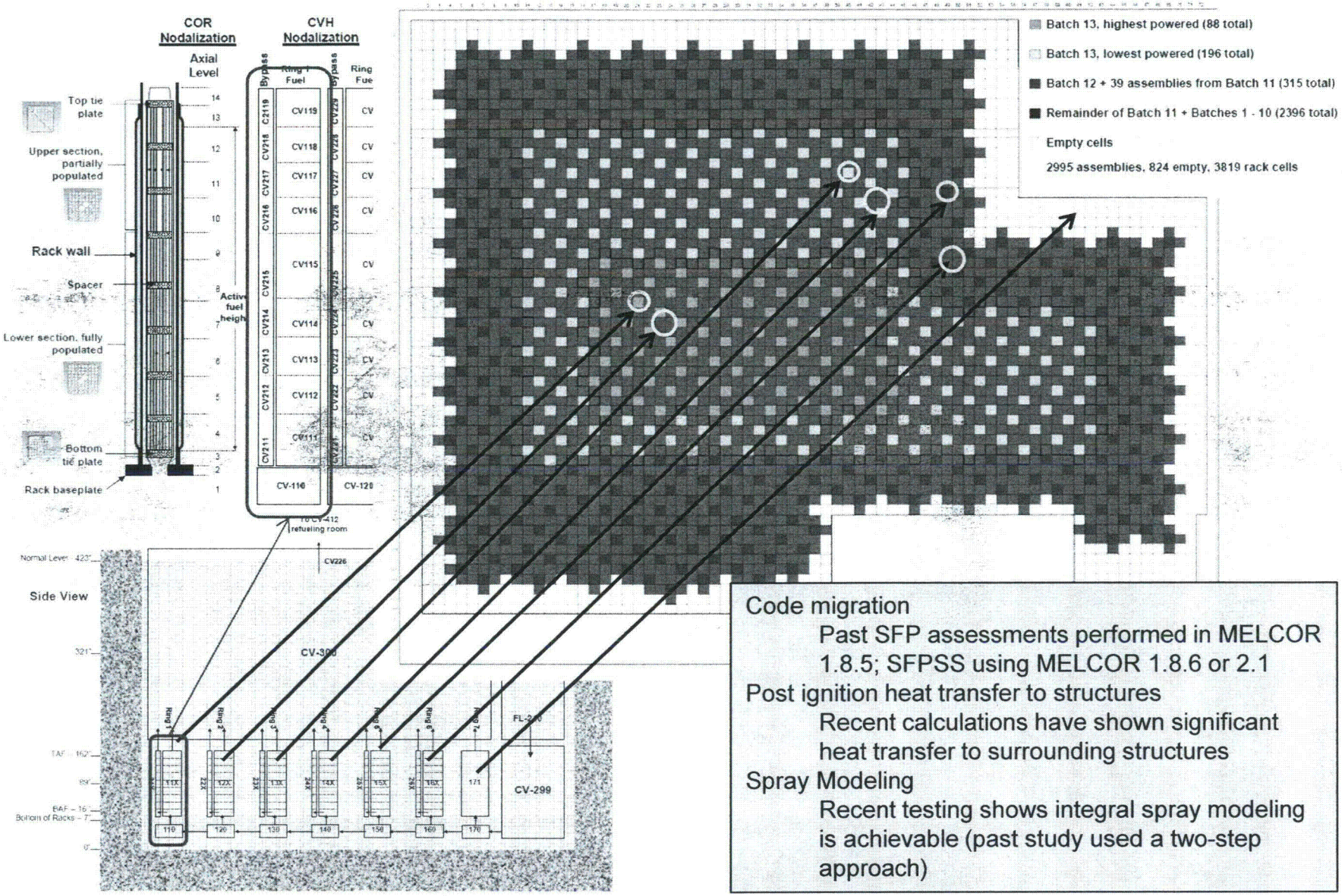
High-Density Post-Outage SFP MELCOR Model



Code migration
- Past SFP assessments performed in MELCOR 1.8.5; SFPSS using MELCOR 1.8.6 or 2.1

Post ignition heat transfer to structures
- Recent calculations have shown significant heat transfer to surrounding structures

Spray Modeling
- Recent testing shows integral spray modeling is achievable (past study used a two-step approach)

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Offsite Consequence Analysis

- MACCS2 Computer Code will be used
- Many of the input values for offsite release and consequence modeling will be based on the same sources used by NUREG-1935 (SOARCA)
- Scope and modeling decisions will be similar to SOARCA, except in instances where a severe accident from a spent fuel pool is expected to differ from a reactor
- Consequence Reporting
  - Health Effects
    - Current Plan: Report the conditional risk of early fatalities and latent cancer fatalities for the surrounding area. (Ideal for informing individual members of the public)
    - Alternative: Report the population dose. (Ideal for informing decision-makers)
  - Land Contamination
    - Current Plan: Report total land contamination above a specified dose level (e.g., the habitability criterion for the selected site of 500 mrem/year)

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Coordination within NRC

- Monthly/bimonthly meetings with points of contact within other offices to gather input from technical staff
- A communication plan has been drafted
- Briefings for Senior Management and Commissioners
- Future ACRS Involvement
  - Planned issuance of draft report in June 2012

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Backup Slides

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Comparison of PB IE frequency from various sources

Source	Estimated initiating event frequency of a large seismic event	Notes
USGS 2008	$1.7 \cdot 10^{-5}$/yr (one event in 61,000 years)	Frequency of seismic bin #3
Peach Bottom 3 SPAR-EE Model (v3.21, Rev. 1)	$1.3 \cdot 10^{-5}$/yr (one event in 77,000 years)	Frequency of seismic bin #3 (of 3)
NUREG-1488 / NUREG-1738	$1.1 \cdot 10^{-5}$/yr (one event in 91,000 years)	Frequency of seismic hazard between 0.51g to 1.02g
INEL-96/0334	$3.2 \cdot 10^{-6}$/yr (one event in 310,000 years)	Frequency of seismic events > 0.6g

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Past studies –Frequency of SFP Uncovery (/yr)

Initiating Event Class	**NUREG-1353** (1989) [BWR, best-estimate[1]]	**NUREG-1738** (2001)
Seismic events	$7 \cdot 10^{-6}$	$2 \cdot 10^{-6}$ (LLNL) $2 \cdot 10^{-7}$ (EPRI)
Cask / heavy load drop	$3 \cdot 10^{-8}$	$2 \cdot 10^{-7}$
LOOP – severe weather	-	$1 \cdot 10^{-7}$
LOOP – other	-	$3 \cdot 10^{-8}$
Internal fire	-	$2 \cdot 10^{-8}$
Loss of pool cooling	$6 \cdot 10^{-8}$	$1 \cdot 10^{-8}$
Loss of coolant inventory	$1 \cdot 10^{-8}$	$3 \cdot 10^{-9}$
Inadvertent aircraft impacts	$6 \cdot 10^{-9}$	$3 \cdot 10^{-9}$
Missiles – general	$1 \cdot 10^{-8}$	-
Missiles - tornado		$< 1 \cdot 10^{-9}$
Pneumatic seal failures	$3 \cdot 10^{-8}$	-

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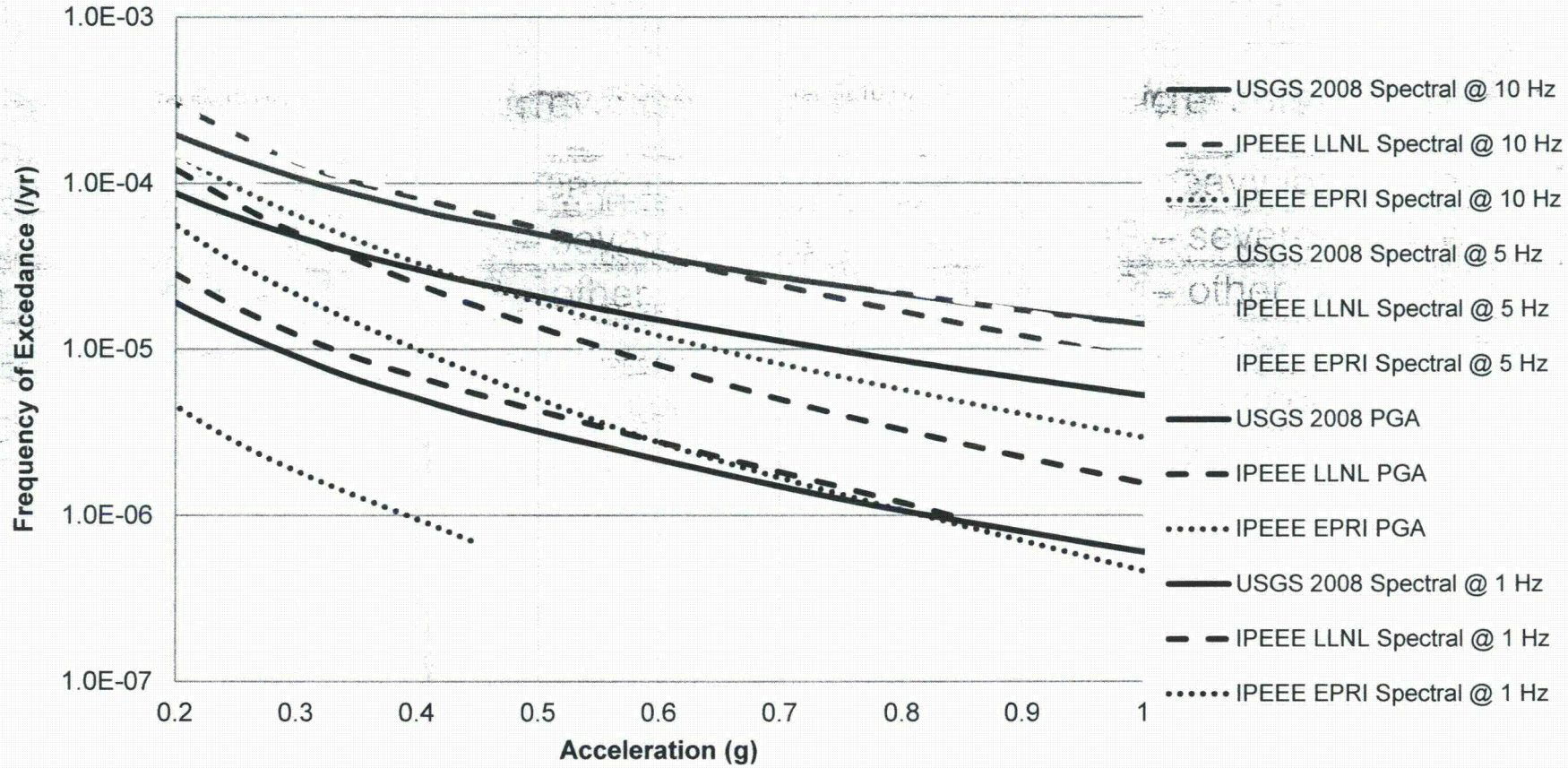
Peach Bottom Seismic Hazard
(for various spectra/information sources)



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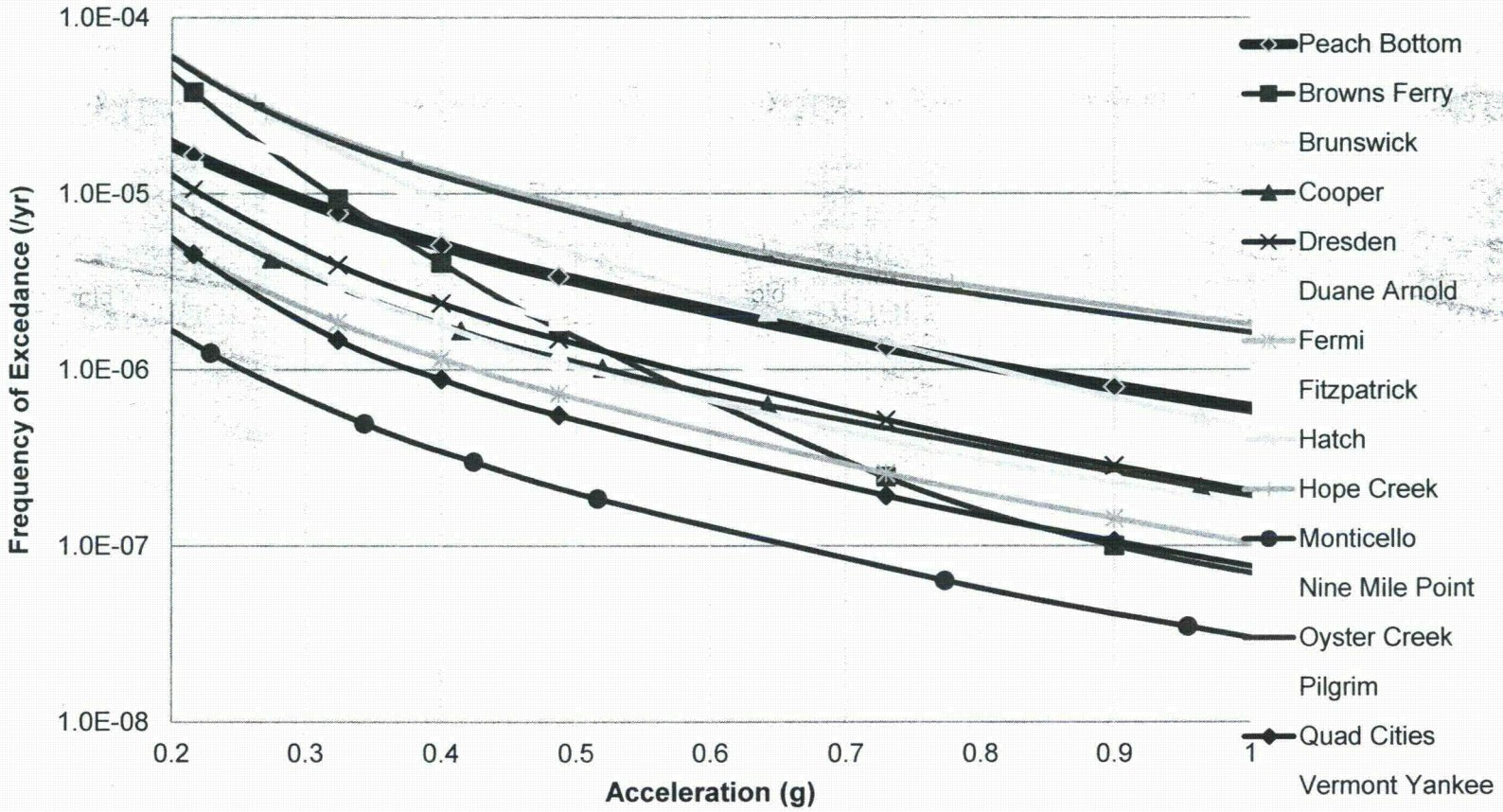
Peach Bottom Seismic Hazard versus Other Mark Is (based on pga)



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Seismic Bins and IE Frequencies (Based on USGS 2008 for Peach Bottom)

			~ Initiating Event Frequency for USGS 2008 (/yr)			
Bin #	**Bin**	**Bin Accel.**	**PGA**	**10 Hz**	**5 Hz**	**1 Hz**
1	0.05 < a < 0.3	0.12	5.20E-04	1.15E-03	1.12E-03	2.44E-04
2	0.3 < a < 0.5	0.39	2.70E-05	5.79E-05	4.44E-05	6.45E-06
3	0.5 < a < 1.0	0.71	1.65E-05	3.75E-05	2.67E-05	2.64E-06
4	1.0 < a < 2.1	1.2*	4.90E-06			
	1.0 < a < 5.0			1.43E-05		
	1.0 < a < 4.9				9.04E-06	
	1.0 < a < 3.7					6.26E-07

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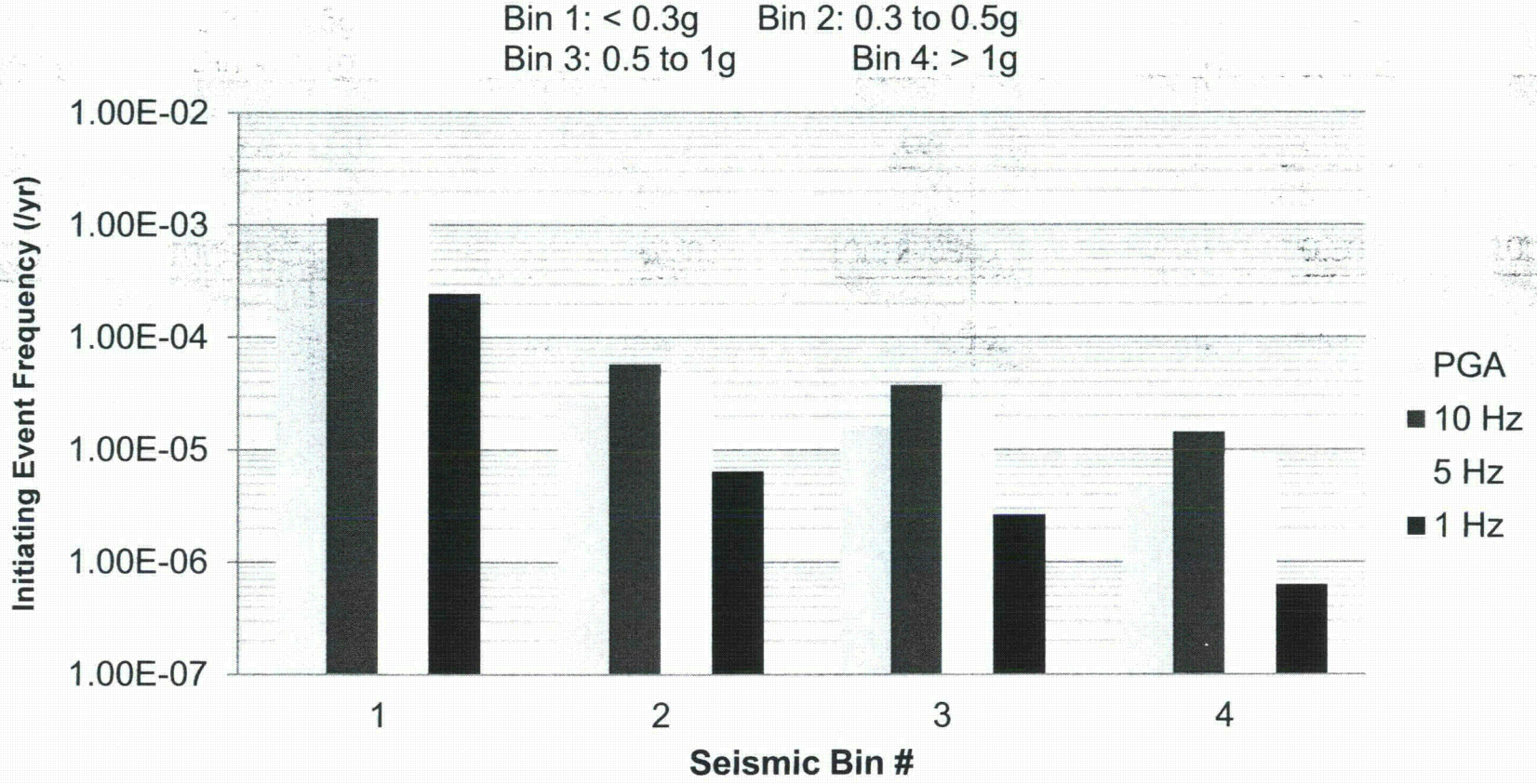
Seismic IE Frequency Comparison (Based on USGS 2008)



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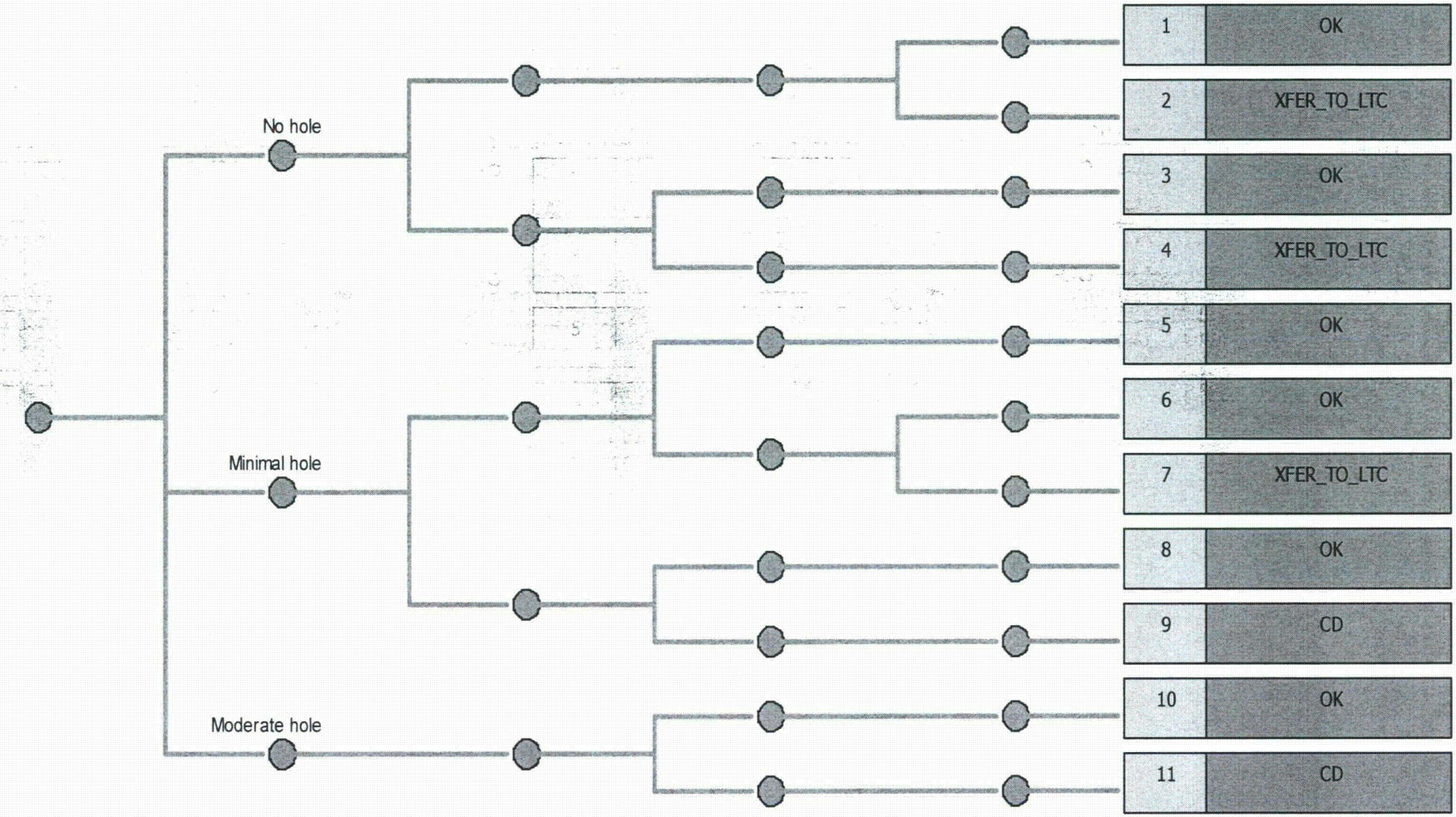
Sample event tree
(For scenario delineation)

0.71g seismic @ 12 days	Equivalent size of the hole, which is located at fuel height	SFPC fails or not due to IE (fail-to-start)	EDMG equipment is successfully deployed in allotted time	SFPC works for a while but stops working after x hours	#	End State (Phase - PH1)
INIT-EV	HOLE-SIZE	SFPC-AVAILABLE	EDMG	SFPC-FAILS-TO-RUN		



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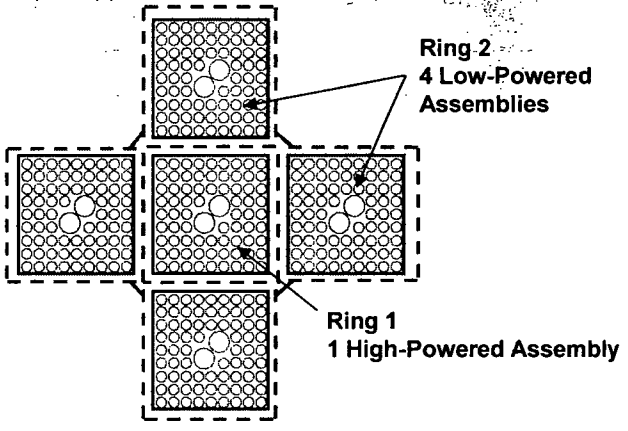
SFP MELCOR Models

- New records (SFP-BWR, SFP-PWR)
  - Rack component introduced in M1.8.5 version RO to model heat transfer within a ring and also from ring to ring
- Breakaway air oxidation model (ANL data)
- Hydraulic resistance model (SNL experiments)

Introduced in MELCOR 1.8.5 Version RO
"SFP-BWR" Model Representation of the 1x4 Model
(Cell wall is represented by new rack component)



Convective Heat Transfer Surfaces:

Ring 1 — Clad, Canister / Water Rods, Rack | *Ring 2* — Clad, Canister / Water Rods, Racks

Radiative Heat Transfer Flow Path:

Fuel → Clad → Canister / Water Rods → Rack → Canister / Water Rods → Clad → Fuel

Ring 1 — Racks — *Ring 2*

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MELCOR Versions

- M1.8.5 version RP (Dec 17 2007)
  - Used in the Whole Pool Model for Peach Bottom (SAND Letter Report, June 2008, Rev. 3)
  - New record CORHTR (user defined radiation path, 'RADIATE2')
  - Increase number of core control volumes (NCVOL > 100)
- M1.8.6 version YV (~ June 2011)
  - Main difference between M1.8.6 and M1.8.5 is introduction of curved lower head – requires input conversion
  - SFP models
  - Latest version with (NCVOL > 100)
  - CORHTR ('RADIATE2' implemented as default)

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Problem Benchmarking

- Focus on single accident sequence (10" bottom LOCA) for the high density loading (15 day aging)
- Converted M1.8.5 model to M1.8.6
- Additional records required to model lower head (a few had to be removed)
- Ring 7 channel in the M1.8.6 input had to be CV171 to be consistent with the new core models (consistency between COR/CVH cell volumes). Core channel was specified as CV299 in M1.8.5 model. Specifying CV171 as core channel in Ring 7 for M1.8.5 produces more consistent results with M1.8.6

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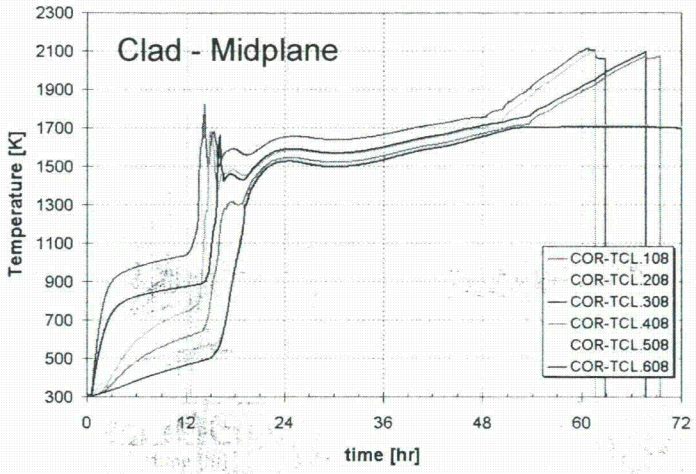
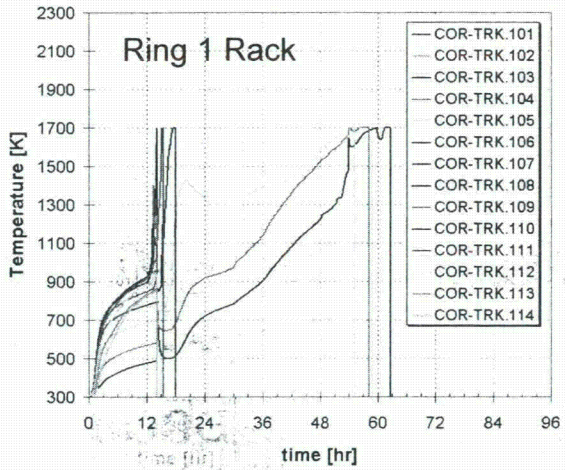
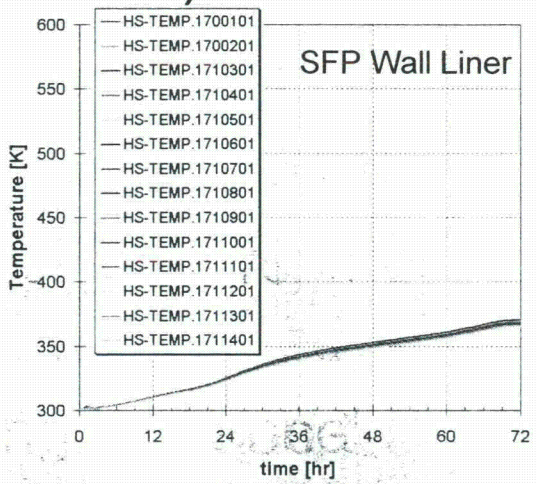
Benchmarking Results

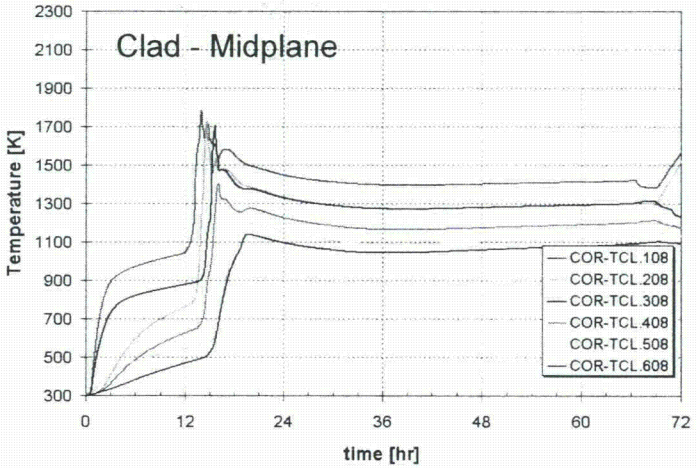
MELCOR 1.8.5 (Ring 7 Channel CV299)

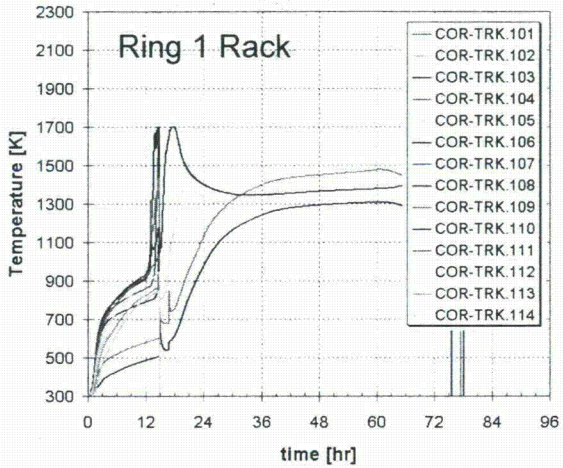


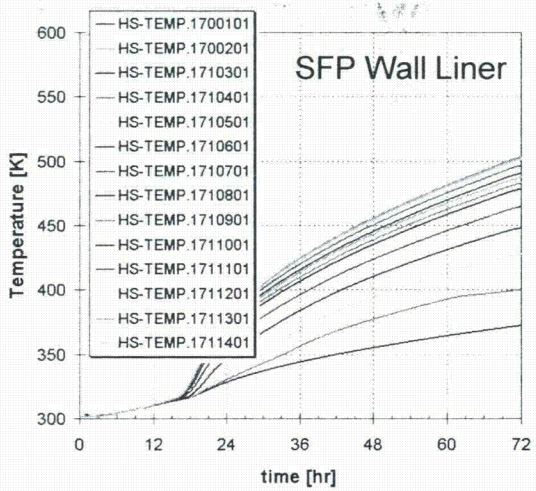




MELCOR 1.8.6 (Ring 7 Channel CV171)







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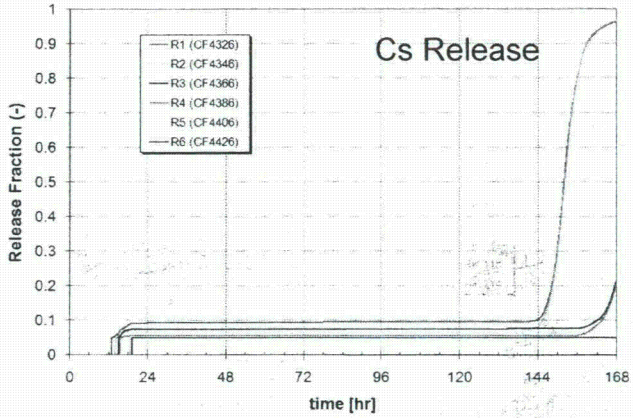
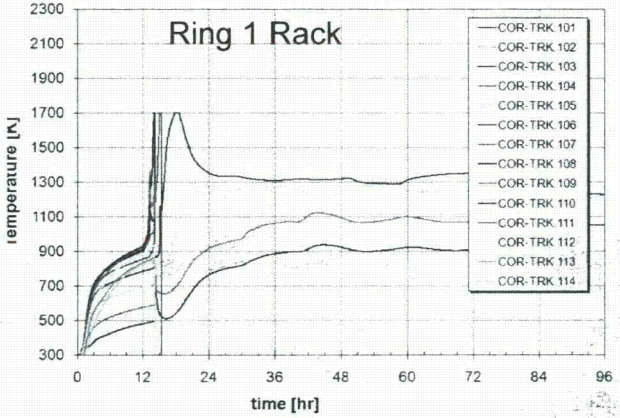
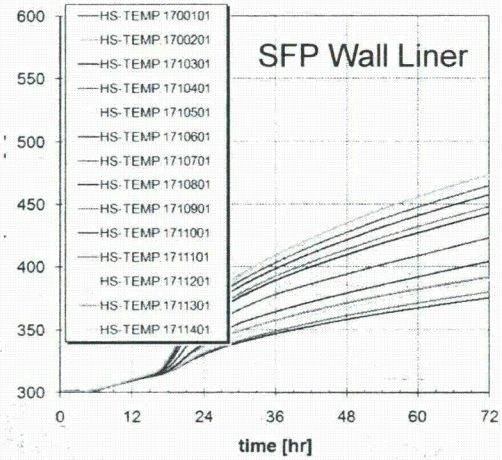
Benchmarking Results

MELCOR 1.8.5 (Ring 7 Channel CV171)







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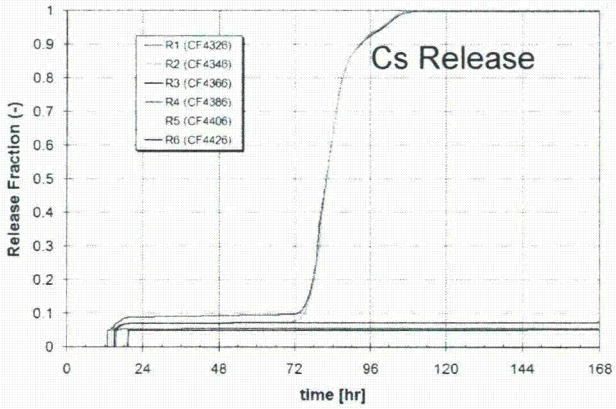

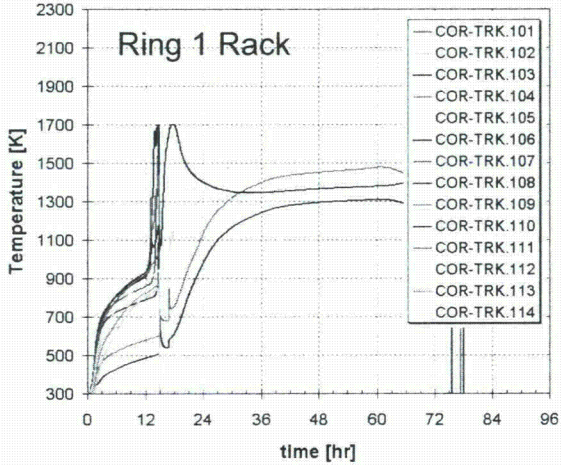

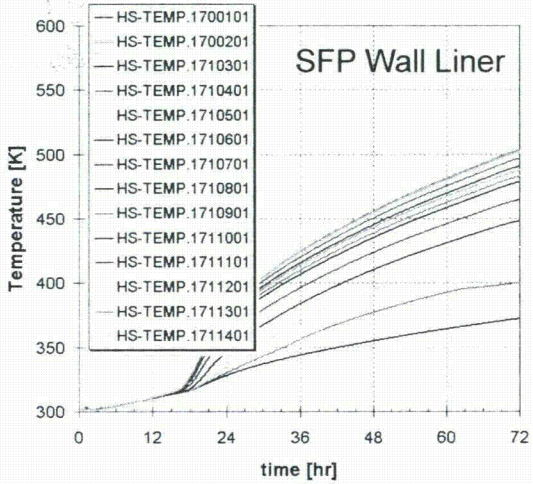

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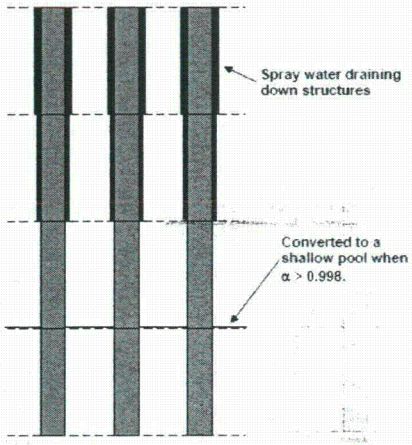
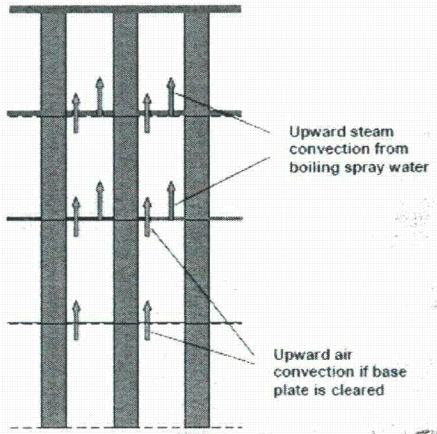
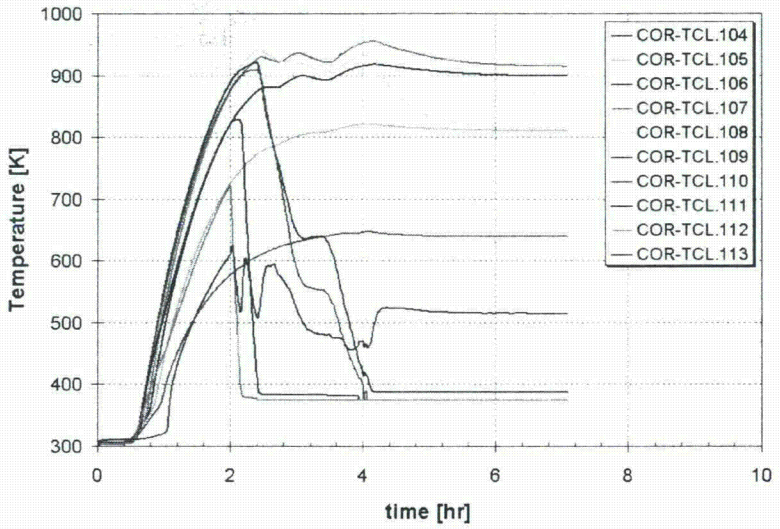
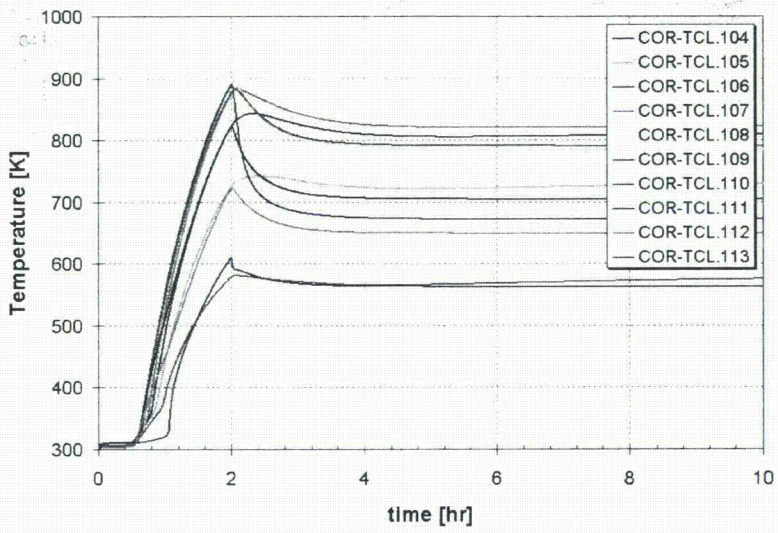
Spray Modeling Results

- Very large LOCA for high density loading (15 day aging)









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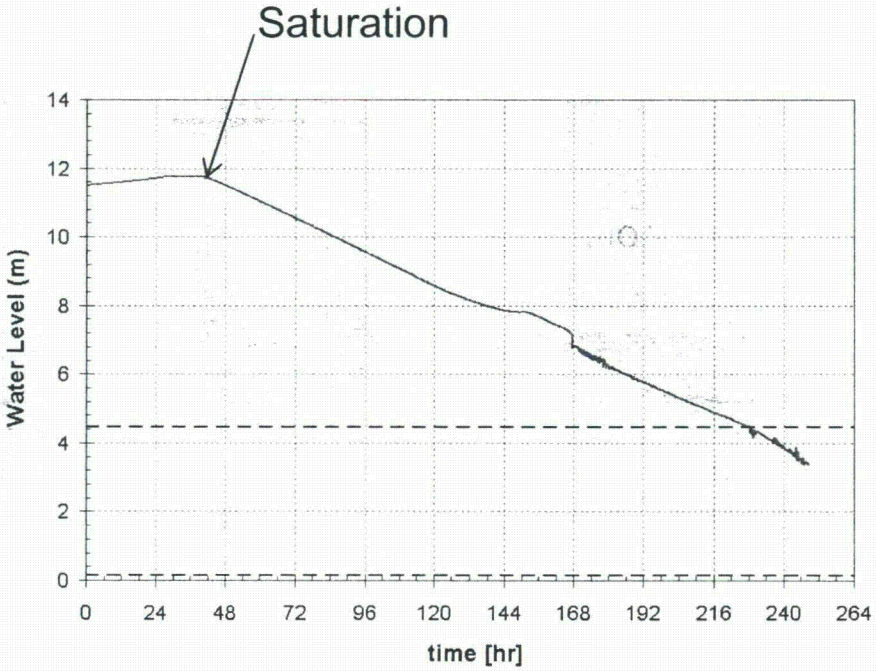
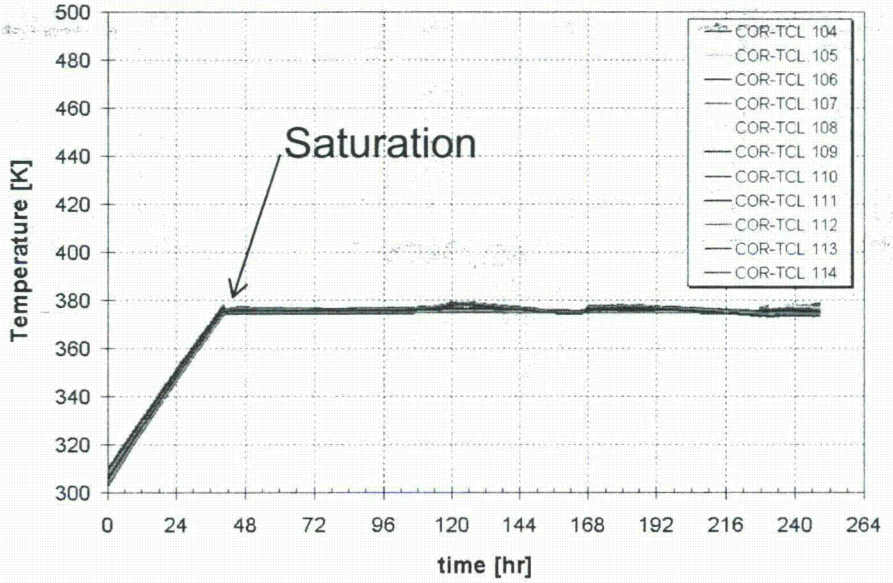
Boil-off Results

- High density loading (15 day aging)





ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
Spent Fuel Pool Scoping Study

Thursday, April 12, 2012
Rockville, MD
- Table of Contents -

Cognizant ACRS Member: S. Armijo

Cognizant ACRS Staff Member: C. Brown

Advisory Committee on Reactor Safeguards
Meeting of the Subcommittee on Materials. Metallurgy, and Reactor Fuels
Spent Fuel Pool Scoping Study
Rockville, MD
Thursday, April 12, 2012
DRAFT AGENDA

Cognizant Staff Engineer: Christopher L. Brown (301)-415-7111, Christopher.Brown@nrc.gov)

Item	Topic	Presenter(s)	Time
1	Opening Remarks and Objectives	Dr. Sam Armijo, ACRS	10:15 – 10:20 a.m.
2	Staff Opening Remarks and Introduction to the Spent Fuel Pool Scoping Study	Kathy Gibson/Brian Sheron, RES	10:20 - 10:25 a.m.
3	Project Background and Past Studies	Katie Wagner, RES	10:25 - 10:30 a.m.
4	Seismic and Structural Methods and Preliminary Results	Jose Pires, RES	10:30 - 10:45 a.m.
5	Scenario Delineation	Don Helton, RES	10:45 - 11:10 a.m.
6	Accident Progression Methods and Preliminary Results	Hossein Esmaili, RES	11:10 - 11:25 a.m.
7	Consequence Analysis Methods	Andrew Nosek, RES	11:25 – 11:35 a.m.
8	Summary and Questions	Katie Wagner, RES	11:35 – 11:40 a.m.
9	Committee Discussion	Dr. Armijo, ACRS	11:30 – 11:45 a.m.
10	Adjourn		11:45 a.m.

Notes:

- During the meeting, 301-415-7360 should be used to contact anyone in the ACRS Office.
- Presentation time should not exceed 50 percent of the total time allocated for a given item. The remaining 50 percent of the time is reserved for discussion.
- Thirty five (35) hard copies (2 B&W slides per page) of each presentation or handout should be provided to the Designated Federal Official 30 minutes before the meeting.
- 10 full page colored copies for the ACRS members and the court reporter.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

SPENT FUEL POOL SCOPING STUDY
April 12, 2012
Rockville, Maryland
- Status Report -

PURPOSE

The purposes of this information briefing is concerning the development of a project plan and reflect on other necessary considerations associated with conducting a spent fuel pool (SFP) scoping study focusing on reexamination of the potential advantages associated with moving older fuel stored in the SFP to dry cask storage in an expedited manner.

OBJECTIVE

The technical objectives of the proposed SFP scoping study include:

- Estimating the effect on SFP accident consequences of a reduced amount of spent nuclear fuel being stored in the SFP (by removing older fuel in an expedited manner and placing it in dry storage). This estimation would be made with a focus on realistic analysis using readily available and technically defensible deterministic methods and assumptions as well as probabilistic insights (to the extent practical).

- Capturing the relevant technical considerations and findings in a form that is conducive to informing deliberation on the issue of SFP accident consequences and whether to expedite movement of spent fuel to dry storage while not addressing any specific regulatory user need.

- Expediting attainment of preliminary insights by (1) relying exclusively on available information and methods, (2) focusing on a single SFP/operating cycle, and (3) using past experience to narrow the scope to a particular scenario of interest.

BACKGROUND

- SFP risk is low, due to the low frequency of events that could damage the thick reinforced pool walls
  - Frequency of fuel uncovery; 6E-7 to 2E-6/yr – NUREG-1738
  - Consequences have been assessed to be large due to the potential for heat-up of all the fuel in the pool
  - Heat-up of the fuel in the pool can lead to "zirconium fire" initiation and propagation
  - Large inventory of Cs-137
- The above has prompted questions as to whether older fuel should be moved to casks

  Past SFP risk studies indicate that seismic hazard is the most prominent contributor to SFP fuel uncovery. While these studies have known limitations, this is sufficient motivation to focus on this class of hazards in the SFP Seismic Study. The focus of the study will be concerned with reexamination of the potential advantages

associated with moving older fuel stored in the SFP to dry cask storage in an expedited manner
    - Emphasis is given to acquiring timely results for ongoing deliberations and external stakeholder interest. The project is using:
        - Available information / methods
        - A representative operating cycle for a BWR Mark I (Peach Bottom)
        - Past studies to narrow scope

TECHNICAL APPROACH

- Two conditions to be considered:
    - Representative of the current situation for the selected site (i.e., high-density loading and a relatively full SFP)
    - Representative of expedited movement of older fuel to a dry cask storage facility (i.e., low-density loading)
- Elements of the study include
    - Seismic and structural assessments based on available information to define initial and boundary conditions
    - SCALE analysis of reactor building dose rates
    - MELCOR accident progression analysis (effectiveness of mitigation, fission product release, etc.)
    - Emergency planning assessment
    - MACCS2 offsite consequence analysis (land contamination and health effects)
    - Probabilistic considerations

SEISMIC AND STRUCTURAL METHODS AND PRELIMINARY RESULTS

Probabilistic Seismic Hazard Assessment

Probabilistic Seismic Hazard Assessment (PSHA) is a tool that can provide damage footprints for SSCs following a seismic event including the relative likelihood of the different damage footprints for user-specified bins (i.e., slices of the spectrum). One form of the output from the PSHA is a plot of the occurrence of ground motion versus frequency of occurrence. The ground motion is initially expressed in the form of peak ground acceleration (PGA), but for a specific site, if its characteristics are known, the PGA can be convolved with the characteristic to obtain a ground motion spectrum.

For Peach Bottom, the frequency-dependent ground motion can be obtained from existing legacy sources. These have been used for past NRC studies such as the IPEEE in which the various NPPs were binned by ground motion according to the level of IPEEE investigation each nuclear power plant was to complete. If the level of information provided by these legacy studies is acceptable, the level of effort and expense are minimized. The studies that provided this information are from the late 1980s.

The bases for the legacy information described above are currently being updated by a consortium of stakeholders—the U.S. Department of Energy, the Electric Power Research Institute, and the U.S. Nuclear Regulatory Commission (NRC) - with the U.S. Geological Survey as a contractor to NRC. This consortium will provide a revised source zone map and a tectonic model of the central and eastern United States (CEUS) by the end of 2011. This represents the first of three parts leading to a modern PSHA for the CEUS. The other

two parts, a computer code and a ground motion propagation model, are scheduled to be available at the end of each of the following 2 years.

The choice is to use seismic hazard data for the site as used for the NRC seismic risk estimates for GI-199 in conjunction with scaling methods for the calculation of a representative ground shaking response spectra at the site, as well as related seismic demands, specifically seismic coefficients and in-structure spectra (as needed). This is to be updated when the new seismic hazard models and analysis codes become available at the end of 3 years. The choice has also been made to use a single seismic bin (or range of peak ground accelerations).

Structural Damage Assessment

To reduce the uncertainty of the methods used, structural analyses will need to be conducted for the seismic demands on the spent fuel structure and its supports. These analyses will involve simplified pushover analyses using as input seismic loads coefficients and in-structure response spectra as needed, determined by scaling of legacy data. The above analysis will be corroborated using a mixture of other supporting analysis (e.g., reassessment of the analysis done for NUREG-1738), event experience (e.g., large seismic events near Japanese nuclear power plants), and practitioner judgment.

The results of the structural engineering assessments will provide the following boundary conditions estimates:

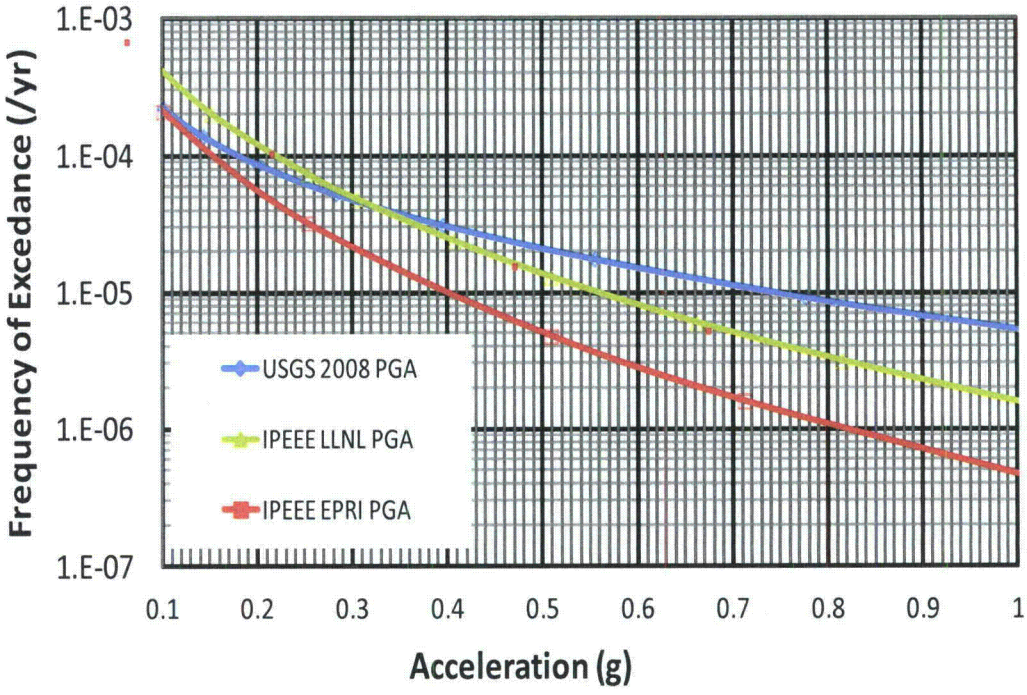
- Potential locations, if any, of leakage from cracking of the concrete and related liner tearing with tentative ranking of more likely locations
- Interval estimates / binning of leakage areas at the potential leakage locations
- The damage, if any, to the spent fuel and the racks sufficient for specifying geometry changes and/or blockage effects that could affect coolability.
- The amount of water, if any, displaced from the pool directly as a result of the seismic event.
- The damage to the overlying building including any debris that might affect fuel coolability considerations.
- Damage to supporting systems and penetrations.
- Damage to other SSCs necessary for accident mitigation (e.g., the building that a portable diesel pump is stored in).

Seismic event: 0.5 g to 1.0 g peak ground acceleration (PGA)

- Challenging but very low frequency of occurrence (one event in 61,000 years)
  - PGA 6 times greater than that for the SSE and beyond the seismic design basis for Eastern US plants
- USGS hazard models (2008) being used as starting model for ground motion response spectra

Site Seismic Hazard

PGA exceedance probabilities from various models



Preliminary Results

- Concrete cracking – Results indicate a through-the-wall concrete crack at the junction of the spent fuel pool walls to the pool floor
  - Crack thickness varies along this junction
- Liner strains – Maximum liner strains at the junction of the walls and floor
  - Analysis for the base finite element model (element size ~ 6 to 8 inches) shows strains in the range
    - 1.5E-4 to 1.9E-3 (0.0019)
    - Yield strain: about 1.2E-3 (0.0012)
  - Detailed liner analysis done to assess strain concentrations

- Penetrations, other structures and equipment
  - Penetrations
    - The analysis shows that the pool is stiff and the distortions small
    - Damage to penetrations is not expected
  - Reactor building
    - Fragility estimates for the NUREG-1150 study (in NUREG/CR-4550) indicate a median fragility of about 1.6 g
    - Natural frequencies of the reactor building are not in the high frequency region of the GMRS used for the site (except for the vertical mode which does not control the building fragility)

        - On these bases, it is concluded that the reactor building would survive this event
    - Building structure above the spent fuel pool and overhead crane
        - Building structure above the pool is not vulnerable to earthquakes (it has low mass and was designed to carry heavy crane loads of 125 tons)
        - Overhead crane, even if were to collapse, would not fall inside the pool
        - On these bases, damage to the pool is not expected from this part of the building from the ground shaking
- Penetrations, other structures and equipment
    - Building housing B5B equipment
        - Water treatment building, which is not a seismic Category I structure
        - Fragility of older buildings in the Central and Eastern United States (CEUS) not especially detailed to resist earthquakes is expected to be less than that of Seismic Category I buildings
        - Estimated median fragility for the turbine building (NUREG 1150 PRA) is about 0.5 g (not Category I)
        - Building collapse would not necessarily prevent access to equipment
        - Absent other information suggest considering equal likelihood of access and non-access to the equipment for this event
- Spent fuel racks and assemblies
    - The condition of spent fuel racks for the event was not calculated
    - Sliding, rocking and possibly contact between racks might occur for this event
        - Would not necessarily imply damage to the racks and assemblies (accelerations less than those expected in cask accidents) but would result in some re-arrangement of relative rack positions and clearances

SCENARIO DELINEATION

- Two SFP loading conditions to be considered:
    - Representative of the current situation for the selected site (i.e., high-density loading and a relatively full SFP)
    - Representative of expedited movement of older fuel to a dry cask storage facility (i.e., low-density loading)
- Successful deployment of mitigation and unsuccessful deployment of mitigation considered for each scenario
- Three SFP damage states considered
- Seismically-induced rupture to reactor piping considered
- Operating cycle is dissected in to 5 phases

SCALE Analysis

- SCALE computer suite used to estimate refueling floor "shine" dose rates from fuel in SFP prior to radioactive release (to provide additional context)
    - Not used to directly affect mitigation assumptions since 50.54(hh)(2) includes consideration of deploying capabilities in high-radiation environments

- Leveraging a capability Oak Ridge National Labs used to inform Fukushima response – "low hanging fruit"
- Results indicate:
  - Worst-case locations exceed 25 rem/hour when SFP water level is 1-2 feet above top of fuel
  - Time since discharge has a limited effect on dose rates
- Additional analysis ongoing:
  - A single lifted assembly (event occurs during fuel handling)
  - Lower-density pool loading

Mitigation Assumptions

- Re-arrangement of fuel to favorable configuration (for high-density loading) assumed to occur at the end of the outage
  - Represents a compromise between pre-configuring (which we believe the site did last outage) and waiting the full exemption time (which is a non-public value) or longer
- For scenarios not including mitigative actions:
  - No operator action is considered
- For scenarios including mitigative actions:
  - Diagnosis is assumed to take until SFP level drops 5 feet + 30 minutes for observation/decision-making (recall assumption of no AC power)
  - Capacities / timings follow underlying endorsed guidance in NEI-06-12, Revision 2
    - Except where an additional 3 hour delay is permitted above and beyond the 2 hour delay
- Mitigation parameters:
  - 200 gpm spray (delivered; uniformly throughout SFP) or 500 gpm makeup
  - Higher site-specific flow rates not credited based on uncertainty in pump speed, pool coverage, etc.
  - Commences 2 hours <u>after diagnosis</u>
  - Mode is determined based on SFP water level at time of deployment
    - Later switching of mode is not considered based on modeling simplification and complications of initiating event
  - Once deployed, equipment runs indefinitely
    - Represents successful arrival of offsite support or deployment of other onsite assets
  - <u>Effectiveness is determined by MELCOR</u>

<u>ACCIDENT PROGRESSION METHODS AND PRELIMINARY RESULTS</u>

MELCOR is an ideal tool for this type of application because (1) its capabilities have been recently developed and validated for treating spent fuel pool accidents and (2) it is able to model the accident progression, and radionuclide release and in-building transport/retention. Accident progression analysis requires initial decay heat and radionuclide inventories. The time-dependent decay heats, masses, and specific activities are input data to MELCOR. For reactor applications, the MELCOR user may choose from a built-in table of normalized decay powers after shutdown based on pre-calculated ORIGEN

calculations for a typical boiling-water reactor and pressurized-water reactor or directly input the decay heat power for a given element. In addition, the radionuclide inventories can be specified on a core-cell basis using user-specified multipliers, and the total inventory or the user can directly input the inventory for each cell. For SFP applications, the time-dependent decay heat must be provided, and the inventories are specified based on the number of assemblies and the inventory for each assembly based on plant-specific ORIGEN calculations.

Preliminary Results

	OCP3			
	High Loading (2.57 MW)		Low Loading (2.15 MW)	
	No mitigation	Mitigated	No mitigation	Mitigated
Low water level signal (hr)	0.47	0.47	0.47	0.47
Water Level at top of Racks (hr)	2.61	2.61	2.61	2.61
End of mitigation deployment (hr)	-	2.97	-	2.97
Water level at racks base plate (hr)	5.3	5.3	5.8	5.8
Time of Release (hr)	16.77	-	14.47	-
Time of spray actutation (hr)	-	5.98	-	5.98
Peak cladding temperature (K)	1823	876	1557	898

	OCP4			
	High Loading		Low Loading	
	No mitigation	Mitigated	No mitigation	Mitigated
Low water level signal (hr)	0.47	0.47	0.47	0.47
Water Level at top of Racks (hr)	2.61	2.61	2.61	2.61
End of mitigation deployment (hr)	-	2.97	-	2.97
Water level at racks base plate (hr)	5.3	5.3	5.8	5.8
Time of Release (hr)	-	-	-	-
Time of spray actutation (hr)	-	5.98	-	5.98
Peak cladding temperature (K)	943	720	896	730

CONSEQUENCE ANALYSIS METHODS

Using the offsite releases calculated with MELCOR, MACCS2 modeling will be performed to predict offsite consequences and conditional individual risk. Given the focus on a particular initiator, binning of rele``ases (to keep the number of MACCS2 analyses tractable) will probably not be necessary. If binning is performed, it will consider release magnitude, timing, and duration as well as emergency-preparedness-related sequence

differences (most notably emergency action level declaration timing and evacuation time estimate differences).

The MACCS2 site modeling developed as part of the SOARCA project should generally be adequate for the purpose of this study. With regard to the production and documentation of results using the Linear No Threshold (LNT) dose response model versus other consensus threshold models, this project will report the same metrics reported by the SOARCA study. Depending on the sensitivity of the offsite consequence results to the site-specific population density, a sensitivity can be performed representing a high-population density site.

Because some past studies have addressed economic consequences of spent fuel pool accidents and because economic consequences could be more significant than health consequences (due to large, delayed releases) the project will consider economic consequences based on existing MACCS modeling. Note that the updated MACCS2 economic model under development will not be completed in time for use in this study.

- MACCS2 code will be used
  - Input: Accident source term, weather, population and economic data, protective measures
  - Output: Consequences (e.g. contamination, health effects) from atmospheric release
  - Modeling will leverage best practices from draft NUREG-1935 (SOARCA)

- Dispersion based on Gaussian plume model (with provisions for meander and surface roughness effects). Phenomena not treated in detail in this model are: Irregular terrain, spatial variations in wind field, temporal variations in wind direction

- Meteorological data required. Wind direction and speed, Pasquill stability category, precipitation, (seasonal [PM,AM]) mixing layer height, and boundary weather

- Multiple weather sequences (accounts for uncertainty in weather conditions at the time of the accident)

Issue raised at the March 6, 2012, Subcommittee Meeting
1. How is thermal risk described in the study? Member Skillman stated there is only a certain amount of that fuel that is considered a real thermal hazard.
2. Past studies have indicated that spent fuel pool seismic hazard is an important piece of overall spent fuel risk. Member Stetkar asked why the frequency of a seismic hazard (initiating event) is considered important. Member Stetkar stated that earthquakes do not care whether or not what the inventory of fuel assemblies are in spent fuel pool.
3. Why selection of a BWR for the study? Member Stetkar said why not a PWR. PWRs have potentially more vulnerabilities to drain-down events than BWRs.
4. Member shack commented on the damage state of the spent fuel pool. Most likely case – no liner tearing and so no water leaking except from sloshing. Small relative likelihood of: extensive liner tearing with drainage controlled by size of concrete cracking and liner tearing localized at the

backup plates.
5. Staff stated that building collapse would not necessarily prevent access to equipment. Member Stetkar challenged this statement and said that if the building is partially collapsed, why can equipment be accessed.
6. Seismic events that disable cooling and makeup systems. Member Stetkar's concern was about non-safety, non-seismic systems, i.e., cooling and purification systems.
7. Staff discussed the two spent fuel pool loading conditions to be considered: Representative of the current situation for the selected site (i.e., high-density loading and a relatively full spent fuel pool) and representative of expedited movement of older fuel to a dry cask storage facility (i.e., low-density loading). Members and staff had a discussion on the differences between and analysis for the high-density and low-density loading cases.
8. How does the study handle the variability for instance between a plant that's on a 24-month fuel cycle, and it's run hard for six years. Member Skillman added why wouldn't you add this detail into the analysis, so that you don't have to do it again later?
9. Member Stetkar asked can you ever get enough sloshing that it creates a problem even if you haven't damaged the structure of the pool.
10. Discussion about whether the staff looked at leak rates by several members. Member Abdel-Khalik asked several follow-up questions relating to leak and flow rates.
11. Chairman Armijo asked did the analysis consider the shape of the axial power profile at the end of cycle, (top peaked versus bottom peaked on the fuel assemblies). Related discussion about fuel assembly location with respect to the benefit of heat transfer to the walls in case of a loss of water
12. Member Bley asked does the study include the dose from activated hardware.
13. Member Skillman asked how does a low-density, post-outage model account for a required offload for an operating event?
14. Discussion on the mitigation assumptions. Member Abdel-Khalik asked about recovery mode where you have air cooling by natural circulation. Staff responded by saying you can have particular sets of conditions where the 500 gpm will cause you to build up a water level that is high enough to cut off natural circulation flow but low enough not to provide steam cooling.
15. Staff will be using the MACCS2 code for the offsite consequence analysis. Input: Accident source term, weather, population and economic data, protective measures. Output: Consequences (e.g. contamination, health effects) from atmospheric release. Member Ryan asked the staff to comment on the uncertainty analysis and sensitivity studies.
17. Member Skillman asked about the use the use of one year of meteorological data in the Consequence Analysis Methods. The meteorological data required are wind direction and speed, Pasquill stability category, precipitation, mixing layer height, and boundary weather. Staff response centered on the same process used for the SOARCA uncertainty analysis.

REFERENCES

1. NUREG-1738, "Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants," February 2001.
2. Memorandum to Brian W. Sheron, Director, Office of Nuclear Regulatory Research, from Michael J. Case, Director, Division of Engineering, Office of Nuclear Regulatory Research, Richard P. Correia, Director, Division of Risk Analysis Office of Nuclear Regulatory Research, Kathy Halvey Gibson, Director, Division of Systems Analysis, Office of Nuclear Regulatory Research, Subject, "Project Plan for Spent Fuel Pool Scoping Study," (ML111570370)

July 26, 2011

MEMORANDUM TO: Brian W. Sheron, Director
Office of Nuclear Regulatory Research

FROM: Michael J. Case, Director */RA/*
Division of Engineering
Office of Nuclear Regulatory Research

Richard P. Correia, Director */RA/*
Division of Risk Analysis
Office of Nuclear Regulatory Research

Kathy Halvey Gibson, Director */RA/*
Division of Systems Analysis
Office of Nuclear Regulatory Research

SUBJECT: PROJECT PLAN FOR SPENT FUEL POOL SCOPING STUDY

This memorandum responds to your request for the Office of Nuclear Regulatory Research (RES) staff to develop a project plan and reflect on other necessary considerations associated with conducting a spent fuel pool (SFP) scoping study focusing on reexamination of the potential advantages associated with moving older fuel stored in the SFP to dry cask storage in an expedited manner. To address your request, the three RES technical divisions have established a working group whose members are shown in Table 3 of this document. This memorandum and its enclosures lay out the technical objectives of the proposed study, background information of relevance, proposed technical approach, items not covered in the proposed plan, resource requirements and impacts, and coordination needs.

Technical Objectives

The technical objectives of the proposed SFP scoping study include:

- Estimating the effect on SFP accident consequences of a reduced amount of spent nuclear fuel being stored in the SFP (by removing older fuel in an expedited manner and placing it in dry storage). This estimation would be made with a focus on realistic analysis using readily available and technically defensible deterministic methods and assumptions as well as probabilistic insights (to the extent practical).

- Capturing the relevant technical considerations and findings in a form that is conducive to informing deliberation on the issue of SFP accident consequences and whether to expedite movement of spent fuel to dry storage while not addressing any specific regulatory user need.

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- Expediting attainment of preliminary insights by (1) relying exclusively on available information and methods, (2) focusing on a single SFP/operating cycle, and (3) using past experience to narrow the scope to a particular scenario of interest.

Background

Various risk studies (most recently NUREG-1738, "Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants," February 2001) have shown that storage of spent fuel in a high-density configuration in SFPs is safe and that the risk is appropriately low. These studies used simplified and sometimes bounding assumptions and models for characterizing the likelihood and consequences of beyond-design-basis SFP accidents. As part of NRC's post-9/11 security assessments, SFP modeling using detailed thermal-hydraulic and severe accident progression models integrated into the MELCOR code were developed and applied to assess the realistic heatup of spent fuel under various pool draining conditions. Moreover, in conjunction with these post-9/11 security assessments, NRC has required enhancements via 10 CFR 50.54(hh) for operating reactor SFP storage that are directed at further improving the coolability of spent fuel under event conditions in which a substantial amount of water has drained from the storage pool.

Recently, the agency has restated its views that spent fuel is stored safely in high-density configurations in a response to Petition for Rulemaking (PRM)-51-10 and PRM-51-12 as well as the revision to NUREG-1437 (the Generic Environmental Impact Statement for License Renewal, Draft Report for Comment). However, this position relies in part on the findings of the aforementioned security assessments, which are not publicly available. The renewed interest in spent fuel storage engendered from the changes in path forward of the planned geologic repository and from the events in Japan following the March 2011 earthquake has rekindled interest in capturing the consequences from postulated accidents associated with high-density SFP storage in an updated safety study. An SFP risk study is being considered as part of a larger initiative involving the conduct of a site Level 3 probabilistic risk assessment (PRA), and SECY- 011-0089 issued July 7, 2011 paper is forthcoming on this issue. In the interim, a desire exists to produce updated consequence estimates for a particular scenario of interest that can in part act as a bridge between the current state-of-knowledge (much of which is greater than 10 years old or security-related and, thus, nonpublicly available) and future studies.

To that end, a scoping study is being proposed that would combine structural and seismic assessments, more detailed mechanistic modeling of fuel heatup and potential fission product behavior, realistic treatment of offsite emergency preparedness, and the current SFP regulatory requirements. Since recently offloaded fuel must be stored in the spent fuel pool (thus requiring low-density pool storage, at a minimum), the project will also identify potential advantages with this configuration as it impacts accident progression and source term. By doing so, benefits associated with low-density pool storage can be assessed.

The older fuel removed from the pool in the low-density pool storage configuration has potential accident consequences associated with its movement and storage in a cask. As a result, those potential accident consequences should be considered along with the low-density pool storage potential accident consequences. In particular, increased spent fuel pool consequences associated with a cask drop event can be significant, whereas fuel handling accidents (which are within the design-basis) would be expected to have much lower consequences. Initiating event cask drop frequencies from the NUREG-1738 study will be reviewed, along with any available data since that time. In the case of cask drop events, this will include a scoping review of readily-available and relevant contemporary information (e.g., heavy load lift programs, etc.)

to determine if the NUREG-1738 estimates can be updated. Depending on the results of this assessment and the results of the seismic consequence study, additional work may be pursued to develop more realistic estimates of the structural and accident progression/consequences of a cask drop event.

Technical Approach

As described above, the following two conditions will be considered:

(1) A condition representative of the current situation for the selected site (i.e., high-density racking, a relatively full SFP, and current regulatory requirements with respect to fuel configuration and preventive/mitigative capabilities); and

(2) A condition representative of expedited movement of older fuel to a dry cask storage facility (i.e., low-density racking, and current applicable regulatory requirements with respect to fuel configuration and preventive/mitigative capabilities).

For purposes of obtaining a near-term perspective on the issue, a single site and single assumed operating cycle will be used. The site characterization (seismic response, decay heat, radionuclide inventory, etc.) will be based on readily available information that will primarily stem from the Individual Plant Examination: External Events (IPEEE), seismic information developed by the United States Geological Survey (USGS), and the post 9/11 security assessments[1]. As discussed further in Enclosure 1, the example site that will be studied for the current project is the Peach Bottom Atomic Power Station.

Accident progression and consequence characterization of the two conditions cited above will be affected by the potential for a "zirconium fire" (i.e., rapid, propagating oxidation reaction of the fuel cladding with air or steam), which leads to significant heatup of the fuel and the release of a substantial quantity of the volatile radionuclides. Owing to the better "coolability" that can be achieved during complete loss-of-pool-inventory events by spreading out the recently offloaded (higher decay heat) fuel, it is expected that the low-density packing configuration has a reduced potential for a "zirconium fire." The potential of having a "zirconium fire" for either situation will change over time due to changes in the overall decay heat load in the SFP and the individual decay heat level of the highest-powered assembly. As such, it will be necessary to study multiple quasi-steady snapshots in time.

In addition to the differences in likelihood, there may be a difference in the magnitude and timing of the potential radionuclide release depending on the situation (i.e., high-density versus low-density racking). For a condition where all of the older fuel has been removed, the SFP fuel inventory is reduced and the initial inventory of long-lived radionuclides (e.g., cesium-137) will be substantially lower. The inventory of short-lived radionuclides (e.g., iodine-131) depends only on the inventory of fuel from the last reactor offload and does not vary between the two conditions.

To limit the time and resources needed to develop a preliminary set of information, a single seismic event will be pursued. The 2008 USGS site-specific seismic hazard data has been discretized into bins based on reactor probabilistic risk assessment (PRA) best practice

[1] The post 9/11-security assessments included consideration of SFPs and resulted in the collection of information and the development of a MELCOR model for a BWR Mark I plant that is very similar to the one being proposed for use in this project.

guidance. The bin that will be studied here corresponds to peak ground accelerations in the range of 0.5g to 1g.[2]

Using the chosen seismic event demands, specifically lateral load coefficients and in-structure spectra (as needed), related damage states for relevant structures, systems, and components (SSCs) will be estimated by scaling seismic demands from legacy analyses (namely those in the Final Safety Analysis Report (FSAR) and the IPEEE) and performing static pushover-like analysis. The results of this analysis will be corroborated via other analyses, event experience, and practitioner judgment. This damage state will be the starting point for subsequent accident progression analysis.

A number of different technical elements need to be considered to assess the potential accident consequences associated with SFP storage. Enclosure 1 describes each of these technical elements. A preliminary project plan has been developed which includes site and outage specification (e.g., time at which refueling operations are assumed to commence), seismic assessment, structural assessment, accessibility scoping, MELCOR accident progression analysis, MELCOR Accident Consequence Code System (MACCS2) offsite consequence analysis, emergency planning assessment, consideration of probabilistic aspects, review, and documentation. Enclosure 2 provides this complete project plan, and Table 1 provides a brief summary of the major phases. As has been discussed at recent briefings for you on the project status, we have begun work on this project to support the schedule shown in Enclosure 2.

Table 1. Project Phases

Phase 1: Project Formulation & Vetting – April through July 2011
Phase 2: Model Development and Boundary Condition Specification – July through October 2011
Phase 3: Production Analysis – November 2011 through April 2012
Phase 4: Documentation & Review – April through June 2012

Enclosure 1 also provides a discussion of items potentially deferred to a future study, should one be authorized. Per the initial specification, this scoping study relies on the use of deterministic methods in a parametric framework for the evaluation of accident consequences. This approach simplifies the overall analysis and allows for the generation of preliminary results in a shorter timeframe.

Resource Requirements and Impacts

The current approach is (1) the development of a parametric scoping study using probabilistic insights focused on a single initiating event (an event associated with a particular range within the seismic hazard spectrum) and (2) the use of deterministic methods. During the planning of this work, the working group also considered the resources needed to accomplish a broader limited-scope SFP risk assessment including the treatment of multiple initiators, new site information, licensee involvement, PRA methods, human reliability analysis, and uncertainty. Resources for this broader study are also provided here for comparison.

[2] The geometric mean acceleration for this bin is 0.71g; the corresponding initiating event frequency for the peak ground acceleration is $1.7 \cdot 10^{-5}$ per year; the range of corresponding initiating event frequencies for this bin based on spectral accelerations from 1 Hz to 10 Hz is $2.6 \cdot 10^{-6}$ to $3.8 \cdot 10^{-5}$ per year.

The working group estimates that total resource requirements for the scoping study would be about 3.65 FTE and $475K over roughly the next 12 months. For comparison, the limited scope risk assessment would require about 5.6 FTE and $950K over roughly 19 months.

Currently, most of the in-house and contractor staff who were involved in prior SFP analyses are engaged in other higher-priority work (e.g., the State-of-the-Art Reactor Consequence Analyses (SOARCA) project). Accessing these individuals' expertise prior to the first quarter of calendar year 2012 without affecting other high-priority work, and relying on other staff who are technically proficient in their disciplines but are less familiar with SFP issues, represent a major challenge.

In addition, the normal delays associated with issuing U.S. Department of Energy contracts (greater than 2 months) or commercial contracts (greater than 6 months) further complicate project planning. The preliminary project plan outlined in Enclosure 2 attempts to accommodate these constraints by designing a project that relies primarily on in-house expertise for the first half of the project's duration.

Table 2 decomposes the resource estimates stated above into their respective organizations and identifies the impacts of reallocating these resources to this project.

Table 2. Impacts of Reallocating Resources

Organization	FY11 FTE / $	FY12 FTE/$	Impacts
RES/DRA/PRAB	0.35 / $0K	0.3 / $50K	Delays of 3-4 months in: • completing Standardized Plant Analysis Risk (SPAR) model success criteria and integrated capabilities model work (user need NRR-2010-017), and • new reactor SPAR model development (NRO-2008-002)
RES/DE/SGSEB	0.55 / $0K	0.25 / $50K	• Note: Start of work deferred to August due to other commitments • Other impacts will be assessed later but likely minor
RES/DSA/FSTB	0.76 / $0K	0.6 / $275K	• Reduce various in-house MELCOR assessments (minor impact overall) • Re-stack fiscal year (FY) 2012 budget (could impact other projects)
RES/DSA/SPB	0.4 / $0K	0.45 / $100K	• Delay NUREGs on WinMACCS for NSIR and NUREG on MACCS2 parameters (2 months delay unless work can be shifted to other staff) • Possible delay on support for this plan or other severe accidents topics • Re-stack FY 2012 budget based on SOARCA priorities (could impact SFP projects)
NMSS/SFST NRR/DE, DRA, DSS NSIR/DPR	Resources needed (and impacts incurred) by these entities for project support (e.g., providing input on specific issues, reviewing interim products, etc.) are not explicitly accounted for at this point.		
Total	2.06 / $0K	1.6 / $475K	

Coordination

Close coordination among the technical areas involved in this project will be essential to the successful completion of the work. Table 3 lists the individual leads in RES along with participation from other offices. The cognizant staff from other offices will be identified via interactions between RES management and management from the respective organizations. RES leads will contact staff in other offices (as needed) to solicit input and feedback. In addition, a communication plan will be developed.

Table 3. Project Coordination and Cognizant Staff

Office	Division	Cognizant staff
RES	DRA DSA DE	D. Helton K. Wagner (overall lead), H. Esmaili A. Murphy
NMSS	DSFST	A. Barto
NSIR	DPR	R. Sullivan
NRR	DE DSS DRA DPR	TBD S. Jones J. Mitman E. Bowman
NRO	DSRA DE	TBD B. Tegeler

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Enclosure 1. Description of the Technical Elements

Site Selection and Information Gathering

One of the first steps in doing a site-specific analysis is to identify the site. Presently, the working group is planning to evaluate one of the spent fuel pools (SFPs) at the Peach Bottom Atomic Power Stations (PBAPS). This site has been studied in a number of different reactor studies including the NUREG-1150 study and the recent State-of-the-Art Reactor Consequence Analyses (SOARCA) project. In addition, some of the necessary supporting information/models for this plant have already been developed. Even so, a significant amount of information and model development would still be required to comprehensively model the current as-designed, as-operated plant. Because of the desire for near-term results, this effort will be minimized and readily available information will be used (e.g., detailed decay heat and radionuclide inventory estimates developed in 2003 as part of a different project).

RES staff will work with the Offices of Nuclear Reactor Regulation (NRR) and Nuclear Material Safety and Safeguards (NMSS) to identify readily available information. If key pieces of information are missing and could be readily produced by the licensee, the NRR plant project manager will be engaged to request this information (if deemed appropriate). Examples of the type of information needed to conduct a study such as this and how they will be addressed here include:

- SFP current spent fuel loading configuration. Loading patterns will be assumed and, prior to the time period at which fuel must be dispersed, it will be assumed to be in a "uniform" pattern (barring additional substantive information).

- Fuel design and operating history information for all fuel in the SFP. This information, which is used for decay heat and radionuclide inventory estimation, will be estimated based on the collection of similar data in 2003.

- Typical outage characteristics. Typical characteristics will be proposed and provided to NRR/DSS for review and comment.

- Refueling practices and cask-loading practices and associated procedures. For the most part, the scope of the scoping study limits the need for this information, and any follow-on work would be potentially highly dependent on this information.

- Site, regional, and structure, system, or component (SSC) information necessary to perform seismic hazard assessments. For offsite infrastructure, the assessment performed for SOARCA will be used; for onsite and SSC assessments, past information (relying heavily on the Individual Plant Examination of External Events [IPEEE]) will be used.

- Seismic Structural Assessment: Information needs include the references in Appendix A2B-12 of NUREG-1738; structural drawings, seismic coefficients and in-structure spectra for the PBAPS reactor building as well as related Final Safety Analysis Report (FSAR) information; IPEEE report for the PBAPS; seismic hazard and seismic spectra estimates for PBAPS used for the GI-199 study; structural aspects and weight of spent fuel racks; structural aspects of rack spacers.

Regarding other information needs not cited above, it is anticipated that some of the information needed may not be readily available, and therefore assumptions will have to be made and documented.

An additional information-related issue is the design of a contemporary low-density rack. Fundamental assumptions will have to be made about this design if information is not available. Some of these assumptions (e.g., an open-rack design versus the closed-cell rack designs currently used for high-density racking for criticality prevention reasons) can have important effects on the coolability of high-decay-heat spent fuel. One could choose to assume that the spent fuel pool is not reracked upon removal of the older fuel and that the licensee will simply spread out the remaining spent fuel in the existing racks, but again this assumption could have a significant effect on the results. Time and resources permitting, a sensitivity study will be performed to investigate this uncertainty.

Treatment of Probabilistic Considerations:

A key characteristic of the scoping study plan is the reliance on deterministic methods being exercised in a parametric manner. The completeness and rigor associated with a limited-scope risk assessment was judged to be incommensurate with the goal of producing results in a short timeframe. However, the decision to not treat risk does not preclude the consideration of probabilistic aspects.

To that end, a parallel effort (i.e., one that is not critical path to the conduct of the deterministic, parametric scoping study) will be undertaken to inform and contribute to the deterministic study. The objectives of this parallel effort will be to (i) provide additional context to the deterministic results since only a subset of the initiators/scenarios will be treated and (ii) assess the relative importance of modeling assumptions and limitations on the overall results, where possible.

This closely-coupled activity will consider probabilistic aspects by looking at:

1. Initiating event timing effects (e.g., the relative likelihood of having an event during the various operating cycle phases)
2. Initiating event frequency (e.g., the frequencies used in NUREG-1738 and whether newer information is available)
3. Event tree development (e.g., the development of event trees for depicting accident scenarios and performing limited, conditional probability figures of merit)
4. Relative likelihood of damage state characteristics (e.g., the study of the effects of different probability distribution functions on the boundary conditions provided by the seismic/structural assessment)
5. Conditional probabilities associated with offsite consequence analysis (e.g., meteorological sampling in MACCS2 analysis)

Item #1 above is addressed in the description of the definition of operating cycle phases. Regarding item #2, initiating event frequencies from the NUREG-1738 study will be reviewed, along with any available data since that time. In the case of cask drop events, this will include a scoping review of readily-available and relevant contemporary information (e.g., heavy load lift programs, etc.) to determine if the NUREG-1738 estimates can be updated. In the case of initiators such as seismic and loss-of-offsite power, this will include assessing whether updated frequencies developed for reactor applications can be directly utilized to update the NUREG-1738 frequencies.

Regarding item #3, scoping accident progression event trees will be developed to delineate the various accident progression pathways being considered. They will also be used to keep track of (and quantify) the relevant conditional probabilities and conditional release frequencies, to the extent practical. These event trees will go from the initiating event to the point of offsite release. The need for binning of releases for limiting the number of required MACCS2 calculations will be assessed at a later time. The event trees will be developed and quantified using the NRC's Systems Analysis Programs for Hands-On Integrated Reliability Evaluations Version 8 (SAPHIRE8) computer code. They will be developed iteratively; that is they will be continually refined throughout the project as the deterministic evaluations proceed.

Item #4 will involve the investigation of assumptions regarding the relative probabilities associated with the seismic and structural input. As such, it is intended to assess the effects of the large source of uncertainty stemming from this portion of the analysis on the end results. Item #5 will be addressed in a standard manner (e.g., stratified sampling), consistent with the current state-of-practice in offsite consequence modeling.

By looking at these probabilistic aspects, the results can be placed in better context, via the limited treatment of relative likelihood. While these elements provide some of the benefits of an actual risk assessment, several elements of a risk assessment are specifically not considered (due to a lack of time, information, or applicability). These are:

- Failure modes and effects analysis
- Data analysis and component reliability
- Effects of dependencies
- Human reliability analysis
- System fault tree development and quantification
- Full event tree quantification

Operating Cycle Phases

During a given operating cycle, the spent fuel pool:

- will change configurations from being an isolated pool to being hydraulically connected to the reactor vessel (and back again)—these will be referred to as pool-reactor configurations to distinguish from the different spent fuel loading configurations;
- may have spent fuel offloaded temporarily from the reactor;
- will have spent fuel offloaded permanently from the reactor;
- will likely have spent fuel moved around within the SFP (as part of the compliance with regulatory requirements and/or otherwise);
- may have older spent fuel offloaded in to casks;
- will experience changes in the peak assembly decay power (of interest for draindown events and spray mitigation) due to the above as well as radioactive decay; and
- will experience changes in the total decay power of all assemblies (of interest for pool heatup/boiling and makeup mitigation) because of the above as well as radioactive decay.

To faithfully represent these temporally changing conditions, one would need to break up the operating cycle into numerous small periods of time or Operating Cycle Phases (OCPs). However, the number of OCPs considered is nearly a linear multiplier on the amount of resources needed because each period of time requires its own set of accident progression and consequence analyses. As such, the definition of the OCPs becomes a minimization /

optimization problem (i.e., it needs to minimize the number of OCPs while optimizing the resulting OCPs' accuracy in representing the above pool-reactor configurations / spent fuel loading configurations/decay heat levels).

For initial planning purposes, it will be assumed that six OCPs are sufficient as outlined in Table 1, based on preliminary assumptions. However, once the site and outage specifications have been finalized, the minimization/optimization problem will need to be re-solved to obtain the actual OCPs. To start, the analyses will be performed for an earlier OCP (e.g., OCP #2) and a later OCP (e.g., OCP #5).

Table 1. Sample OCP Definition for a Boiling-Water Reactor Operating Cycle (Assumes Fuel Shuffling)

OCP #	OCP Description	Time period (days)[1]	% of operat. cycle	Pool-reactor configuration	Spent fuel config.[4]	Total decay power[3]	Peak assembly power
1	Defueling of the reactor (1/3rd core)	2 - 8	0.8	Refueling	Non-dispersed	Existing[2] + (1/2 of offloaded assemblies) @ 3.6 days[1]	Highest powered offloaded assembly @ 3.6 days
2	Reactor T&M / inspection and refueling	8 - 20	1.6	Refueling	Non-dispersed	Existing[2] + (offloaded assemblies) @ 12 days[1]	Highest powered offloaded assembly @ 12 days
3	Remainder of outage	20 – 30	1.4	Unconnected	Non-dispersed	Existing[2] + (offloaded assemblies) @ 24.2 days[1]	Highest powered offloaded assembly @ 24.2 days
4	Highest decay power portion of non-outage period	30 – 60	4.1	Unconnected	Dispersed	Existing[2] + (offloaded assemblies) @ 41.3 days[1]	Highest powered offloaded assembly @ 41.3 days
5	Highest decay power portion of non-outage period	60 – 240	24.7	Unconnected	Dispersed	Existing[2] + (offloaded assemblies) @ 107 days[1]	Highest powered offloaded assembly @ 107 days
6	Remainder of operating cycle	240 – 730; 0 – 2	67.4	Unconnected; cask loadings	Dispersed	Existing[2] + (offloaded assemblies) @ 389 days[1]	Highest powered offloaded assembly @ 389 days

[1] t= 0 is set to the time subcriticality in the reactor is achieved.
[2] Refers to the fuel residing in the SFP at t = 0 (prior to offload).
[3] Offloading of fuel in to casks is not considered as it would have a secondary effect on total decay power.
[4] For the sake of demonstration, this assumes that the pool is not preconfigured to disperse high-decay heat fuel and that this action is taken at 30 days. Note that a sensitivity could be performed to look at alternative assumptions.

Decay Heat and Radionuclide Inventory Assessment

Accident progression analysis requires initial decay heat and radionuclide inventories. The time-dependent decay heats, masses, and specific activities are input data to MELCOR. For

reactor applications, the MELCOR user may choose from a built-in table of normalized decay powers after shutdown based on pre-calculated ORIGEN calculations for a typical boiling-water reactor and pressurized-water reactor or directly input the decay heat power for a given element. In addition, the radionuclide inventories can be specified on a core-cell basis using user-specified multipliers, and the total inventory or the user can directly input the inventory for each cell. For SFP applications, the time-dependent decay heat must be provided, and the inventories are specified based on the number of assemblies and the inventory for each assembly based on plant-specific ORIGEN calculations

For the MACCS2 consequence code, data on fluid flows and radionuclide transport to the environment must be specified. A MACCS2 interface has been integrated into MELCOR by specifying the release paths and associated data (e.g., fluid temperature, released mass).

Probabilistic Seismic Hazard Assessment

Probabilistic Seismic Hazard Assessment (PSHA) is a tool that can provide damage footprints for SSCs following a seismic event including the relative likelihood of the different damage footprints for user-specified bins (i.e., slices of the spectrum). One form of the output from the PSHA is a plot of the occurrence of ground motion versus frequency of occurrence. The ground motion is initially expressed in the form of peak ground acceleration (PGA), but for a specific site, if its characteristics are known, the PGA can be convolved with the characteristic to obtain a ground motion spectrum.

For Peach Bottom, the frequency-dependent ground motion can be obtained from existing legacy sources. These have been used for past NRC studies such as the IPEEE in which the various NPPs were binned by ground motion according to the level of IPEEE investigation each nuclear power plant was to complete. If the level of information provided by these legacy studies is acceptable, the level of effort and expense are minimized. The studies that provided this information are from the late 1980s.

The bases for the legacy information described above are currently being updated by a consortium of stakeholders—the U.S. Department of Energy, the Electric Power Research Institute, and the U.S. Nuclear Regulatory Commission (NRC)—with the U.S. Geological Survey as a contractor to NRC. This consortium will provide a revised source zone map and a tectonic model of the central and eastern United States (CEUS) by the end of 2011. This represents the first of three parts leading to a modern PSHA for the CEUS. The other two parts, a computer code and a ground motion propagation model, are scheduled to be available at the end of each of the following 2 years.

The choice is to use seismic hazard data for the site as used for the NRC seismic risk estimates for GI-199 in conjunction with scaling methods for the calculation of a representative ground shaking response spectra at the site, as well as related seismic demands, specifically seismic coefficients and in-structure spectra (as needed). This is to be updated when the new seismic hazard models and analysis codes become available at the end of 3 years. The choice has also been made to use a single seismic bin (or range of peak ground accelerations) as noted in page 4 of the memorandum.

Structural Damage Assessment

To reduce the uncertainty of the methods used, structural analyses will need to be conducted for the seismic demands on the spent fuel structure and its supports. These analyses will

involve simplified pushover analyses using as input seismic loads, coefficients and in-structure response spectra as needed, determined by scaling of legacy data. The above analysis will be corroborated using a mixture of other supporting analysis (e.g., reassessment of the analysis done for NUREG-1738), event experience (e.g., large seismic events near Japanese nuclear power plants), and practitioner judgment.

The results of the structural engineering assessments will provide the following boundary conditions estimates:

- Potential locations, if any, of leakage from cracking of the concrete and related liner tearing with tentative ranking of more likely locations
- Interval estimates / binning of leakage areas at the potential leakage locations
- The damage, if any, to the spent fuel and the racks sufficient for specifying geometry changes and/or blockage effects that could affect coolability.
- The amount of water, if any, displaced from the pool directly as a result of the seismic event.
- The damage to the overlying building including any debris that might affect fuel coolability considerations.
- Damage to supporting systems and penetrations.
- Damage to other SSCs necessary for accident mitigation (e.g., the building that a portable diesel pump is stored in).

The results of these assessments will be used to specify the boundary conditions for MELCOR analysis.

Accident Progression Analysis

Using existing models, a "whole-pool" MELCOR model will be developed to describe each of the OCPs for each of the configurations (i.e., low-density and high-density racking) considered. It also may be necessary to further develop the state-of-practice in the area of mitigative spray analysis from both a thermal-hydraulic and aerosol capture standpoint[1]. The resulting models will be tested, and results will be compared to past analyses to ensure they adequately capture all of the relevant phenomena.

A calculation matrix will be developed that will identify the preliminary calculations needed to cover the most important boundary conditions (e.g., leak hole size and location). In addition, scoping boil-off calculations also will be performed.

MELCOR is an ideal tool for this type of application because (1) its capabilities have been recently developed and validated for treating spent fuel pool accidents and (2) it is able to model the accident progression, and radionuclide release and in-building transport/retention.

Offsite Consequence Modeling

Using the offsite releases calculated with MELCOR, MACCS2 modeling will be performed to predict offsite consequences and conditional individual risk. Given the focus on a particular initiator, binning of releases (to keep the number of MACCS2 analyses tractable) will probably

[1] While capabilities have been developed in this area, more work may be required to include this capability in to a general purpose-type model. An example of an issue that arises in this regard is capturing the effect of "shutting off" natural circulation air flow if the combination of spray capacity, leak size, and leak location results in flooding above the rack base-plate but not to a height sufficient for steam cooling.

not be necessary. If binning is performed, it will consider release magnitude, timing, and duration as well as emergency-preparedness-related sequence differences (most notably emergency action level declaration timing and evacuation time estimate differences).

The MACCS2 site modeling developed as part of the SOARCA project should generally be adequate for the purpose of this study. With regard to the production and documentation of results using the Linear No Threshold (LNT) dose response model versus other consensus threshold models, this project will report the same metrics reported by the SOARCA study. Depending on the sensitivity of the offsite consequence results to the site-specific population density, a sensitivity can be performed representing a high-population density site.

Because some past studies have addressed economic consequences of spent fuel pool accidents and because economic consequences could be more significant than health consequences (due to large, delayed releases) the project will consider economic consequences based on existing MACCS modeling. Note that the updated MACCS2 economic model under development will not be completed in time for use in this study.

Internal and External Review

The work breakdown in Enclosure 2 includes an Advisory Committee on Reactor Safeguards (ACRS) subcommittee meeting. The utility of additional interactions with internal and external stakeholders will be re-evaluated as the project proceeds. Additional review, including external review, will need to take place after the initial 12-month project period. The interim reports documenting the results of the specific analysis (e.g., accident progression) in phases 1 through 3 will be internally reviewed by the cognizant staff from the various program offices.

Documentation

As part of the review process, different components of the study need to be documented. The documentation will include objectives, assumptions, limitations, technical approach and analysis, preliminary results, and conclusions. At the end of the study, a summary report will be prepared to document the main findings of the study as it relates to the differential consequence assessment in response to the objectives of the study. The various detailed elements of the study from each discipline will be documented as appendices to the summary report.

Study Scope Limitations

Resource and schedule constraints associated with this project have certain scope limitations that impact the ways in which the results of the study can be applied. For example, the inclusion of probabilistic aspects within the current scoping study (described earlier in this enclosure) will allow for consideration of some aspects of likelihood, but will not support definitive statements on risk. The limited-scope parametric scoping study proposed here promotes the development of underlying capabilities needed to perform a SFP risk study, and produces a subset of the overall information needed to inform the issue of SFP storage requirements in a short timeframe. However, it is important that these results be viewed as a stepping stone to a more comprehensive set of results to be generated as part of follow-on work or a different study.

The working group has included in the project scope sufficient aspects of the problem to support preliminary insights on the question of whether expedited movement of spent fuel to dry cask storage would provide a significant safety benefit. This section points out the important pieces

that presently fall outside the scope of the project plan. It is recommended that the following aspects receive additional attention as part of follow-on work or as part of other planned initiatives (such as the site Level 3 Probabilistic Risk Assessment):

- Additional inclusion of new seismic hazard assessments using updated geological data and fragility information

- Assessment of risk (as opposed to potential accident consequences with treatment of probabilistic considerations) for the purpose of being able to make statements regarding the comparison of the absolute risk to the agency's quantitative health objectives and assessment of changes in risk

- Assessment of potential accident consequences beyond a single site and a single postulated fuel cycle, for the purpose of extrapolating the results to other sites or significantly different outage specifications

- Treatment of site-specific details beyond those which are readily available (e.g., detailed structural information, procedural information, current SFP inventory/decay heat loads, etc.)

- Rigorous treatment of the detailed design of a contemporary, low-density rack

- Consideration of increased potential accident consequences from additional cask movements (and thus heavy load drops) associated with expedited movement of older fuel out of the SFP

- Cost/benefit considerations (beyond offsite economic impacts that are part of the set of results produced via MACCS2 analysis) from the standpoint of both generic cost/benefit estimations in the context of a regulatory backfit as well as a consideration of cost-beneficial actions other than expedited fuel offloading to casks (akin to a severe accident mitigation alternative analysis)

- Use of the updated economic model used in the MACCS2 computer code once it is available

- Consideration of other initiators[2]

- Quantification of uncertainty in the results beyond parametric sensitivity studies

- Inclusion of human reliability analyses (HRAs)[3]

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[2] A single bin of seismic events is considered as part of the initial effort. A preliminary investigation of what initiators might be relevant was performed based on existing information. Some initiators (particularly cask drop events and severe weather-induced loss-of-offsite power events) could reasonably be expected to contribute to the overall risk, but are excluded from the scope of the initial scoping study for resource and schedule considerations. Other initiators (loss of spent fuel pool cooling, internal and external fires, etc.) would contribute some fraction (believed to be small based on NUREG-1738) to the overall risk but cannot be accommodated within the given schedule/resource constraints.

[3] HRA models and procedures (both for initiating-event-related failure event identification and quantification as well as mitigative action quantification) are primarily centered on reactor applications and the emergency operating procedures. Recent work in fire HRA, shutdown HRA,

and fuel-handling qualitative HRA would be leveraged if follow-on work brought this element within scope. In the absence of HRA, no quantitative basis will exist for assigning relative likelihood between mitigated and unmitigated accident progression scenarios.

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Spent Fuel Pool (SFP) Scoping Study

Office of Nuclear Regulatory Research

January 18, 2012

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Overview

- Focus: reexamination of the potential advantages associated with moving older fuel stored in the SFP to dry cask storage in an expedited manner
- Study to be completed by: June 2012
- Explicit consideration of past SFP safety and security studies in project planning/conduct
  - Resolution of GI-82 (1989)
  - NUREG-1738 (2001)
  - Post-9/11 SFP Security Assessments

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Spent Fuel Pool Scoping Study

- **Technical Objectives**:
  - Estimating the effect on SFP accident consequences of a reduced amount of spent nuclear fuel being stored in the SFP based on realistic analysis using readily available and technically defensible deterministic methods and assumptions as well as probabilistic insights.
  - Capturing the relevant technical considerations and findings in a form that is conducive to informing deliberation on the issue of SFP accident consequences and whether to expedite movement of spent fuel to dry storage while not addressing any specific regulatory user need.
  - Expediting attainment of preliminary insights by (1) relying exclusively on available information and methods, (2) focusing on a single SFP/operating cycle, and (3) using past experience to narrow the scope to a particular scenario of interest.

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Technical Approach

- Two conditions to be considered:
  - Representative of the current situation for the selected site (i.e., high-density loading and a relatively full SFP)
  - Representative of expedited movement of older fuel to a dry cask storage facility (i.e., low-density loading)
- Elements of the study include
  - A single site/single operating cycle
  - Seismic and structural assessments based on available information to define initial and boundary conditions
  - SCALE analysis of reactor building dose rates
  - MELCOR accident progression analysis (effectiveness of mitigation, fission product release, etc.)
  - Emergency planning assessment
  - MACCS2 offsite consequence analysis (land contamination and health effects)
  - Probabilistic considerations

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Boundary Condition Specification

- Basic scenario prescribed
  - Seismic event: 0.5 g to 1.0 g Peak Ground Acceleration (PGA)
    - Challenging but very low frequency occurrence (PGA > 6 times the SSE and beyond the seismic design basis for Eastern US plants)
    - USGS hazard models (2008) being used as starting model for ground motion response spectra
  - Peach Bottom Unit #3 (PB3)
  - 5 operating cycle snapshots
    - e.g., latter half of refueling outage represented by 13 days since reactor shutdown with fuel in a non-dispersed configuration

Note: Considers consequences associated with the SFP itself (i.e., consequences associated with fuel cask handling, transportation, and storage are not within the scope)

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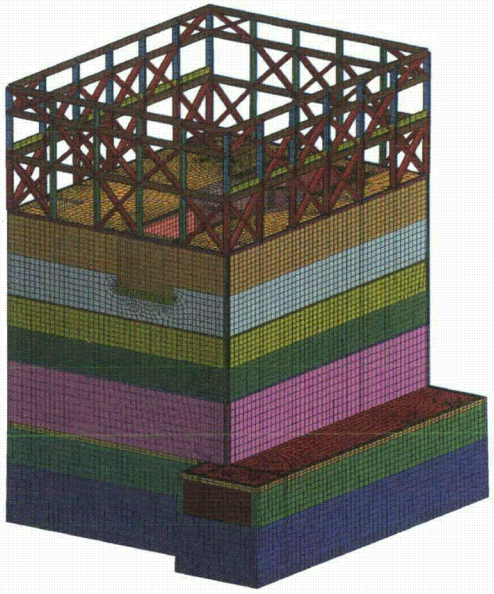
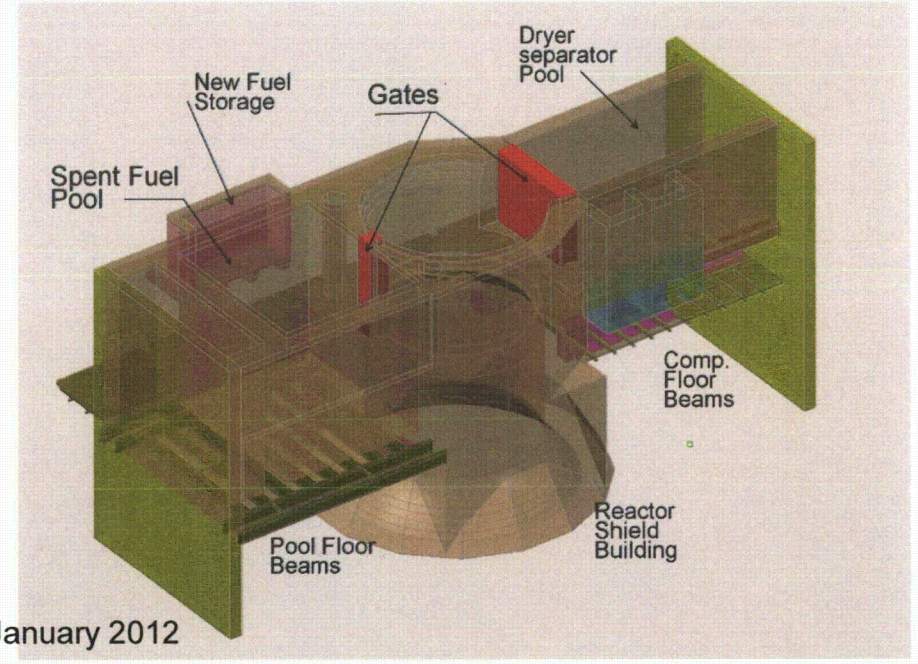
Seismic & Structural Inputs

- Seismic Assessment
  - Challenging damage not expected below seismic bin 3 (0.5 – 1.0 g)
    - Level of 0.71g studied - Challenging but very low frequency of occurrence
  - USGS hazard assessments (2008) as starting model for ground motion response spectra (GMRS)
- Structural Assessment - To determine starting point for subsequent accident progression analysis
  - Damage to the spent fuel pool structure and liner
  - Other damage states: loss of water from sloshing; damage to SFP penetrations, reactor building and other relevant structures, racks and fuel, and reactor shutdown cooling systems





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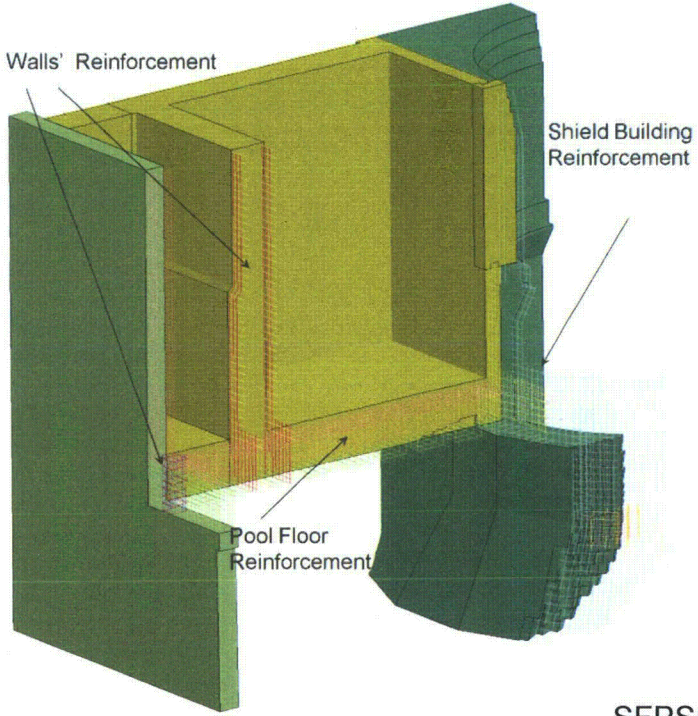
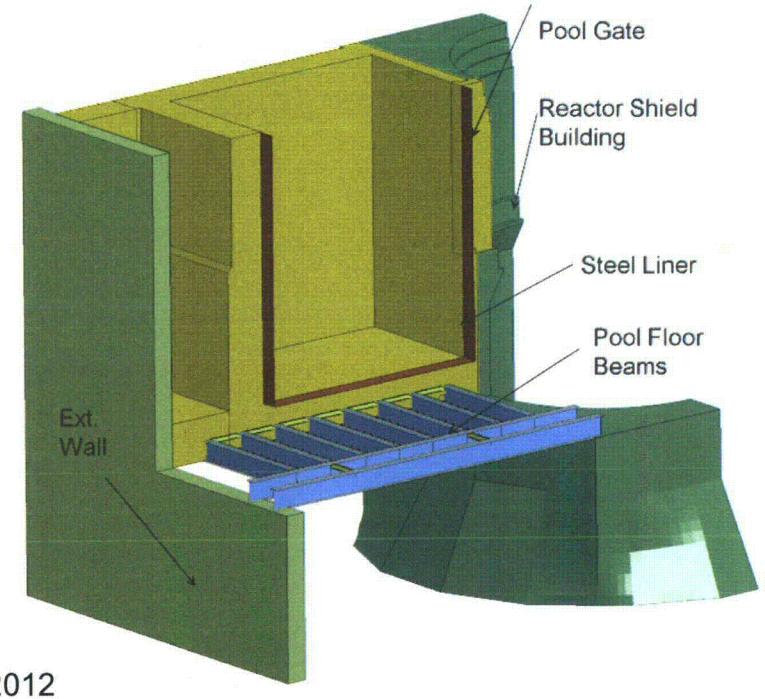
Seismic & Structural Inputs

- Structural Assessment
  - Follows the general approach used for GI-82 (NUREG/CR-5176) together with scaled in-structure spectra from the NUREG-1150 PRA for calculation of demands
  - Finite element analysis to calculate SFP frequencies, verify hydrodynamic loads and calculate distortions of the SFP structure, concrete cracking, and liner strains.





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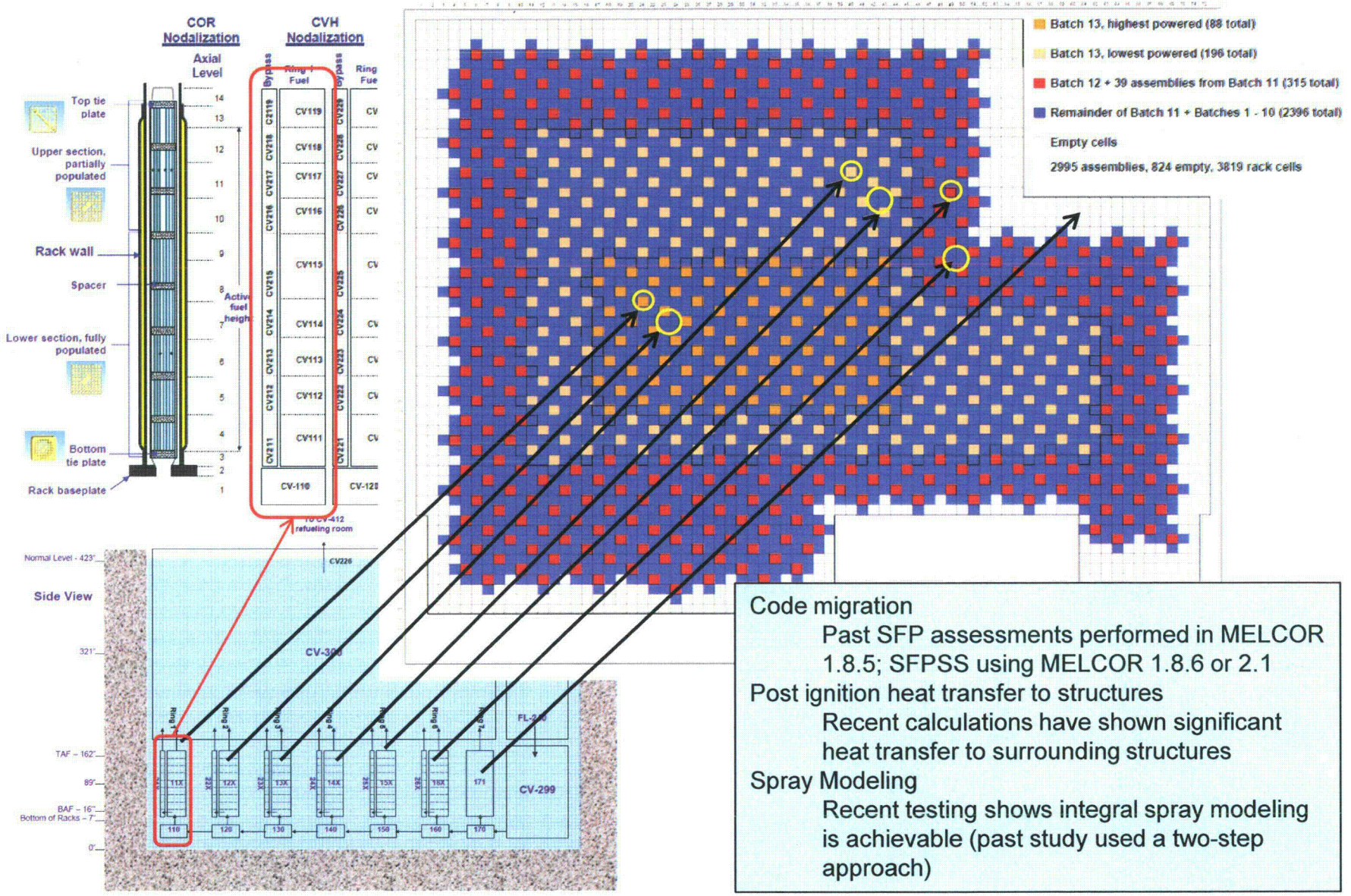
High-Density Post-Outage SFP MELCOR Model



Code migration
- Past SFP assessments performed in MELCOR 1.8.5; SFPSS using MELCOR 1.8.6 or 2.1

Post ignition heat transfer to structures
- Recent calculations have shown significant heat transfer to surrounding structures

Spray Modeling
- Recent testing shows integral spray modeling is achievable (past study used a two-step approach)

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Offsite Consequence Analysis

- MACCS2 Computer Code will be used
- Many of the input values for offsite release and consequence modeling will be based on the same sources used by NUREG-1935 (SOARCA)
- Scope and modeling decisions will be similar to SOARCA, except in instances where a severe accident from a spent fuel pool is expected to differ from a reactor
- Consequence Reporting
  - Health Effects
    - Current Plan: Report the conditional risk of early fatalities and latent cancer fatalities for the surrounding area. (Ideal for informing individual members of the public)
    - Alternative: Report the population dose. (Ideal for informing decision-makers)
  - Land Contamination
    - Current Plan: Report total land contamination above a specified dose level (e.g., the habitability criterion for the selected site of 500 mrem/year)

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Coordination within NRC

- Monthly/bimonthly meetings with points of contact within other offices to gather input from technical staff
- A communication plan has been drafted
- Briefings for Senior Management and Commissioners
- Future ACRS Involvement
  - Planned issuance of draft report in June 2012

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Backup Slides

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Comparison of PB IE frequency from various sources

Source	Estimated initiating event frequency of a large seismic event	Notes
USGS 2008	$1.7 \cdot 10^{-5}$/yr (one event in 61,000 years)	Frequency of seismic bin #3
Peach Bottom 3 SPAR-EE Model (v3.21, Rev. 1)	$1.3 \cdot 10^{-5}$/yr (one event in 77,000 years)	Frequency of seismic bin #3 (of 3)
NUREG-1488 / NUREG-1738	$1.1 \cdot 10^{-5}$/yr (one event in 91,000 years)	Frequency of seismic hazard between 0.51g to 1.02g
INEL-96/0334	$3.2 \cdot 10^{-6}$/yr (one event in 310,000 years)	Frequency of seismic events > 0.6g

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Past studies –Frequency of SFP Uncovery (/yr)

Initiating Event Class	**NUREG-1353** (1989) [BWR, best-estimate[1]]	**NUREG-1738** (2001)
Seismic events	$7 \cdot 10^{-6}$	$2 \cdot 10^{-6}$ (LLNL) $2 \cdot 10^{-7}$ (EPRI)
Cask / heavy load drop	$3 \cdot 10^{-8}$	$2 \cdot 10^{-7}$
LOOP – severe weather	-	$1 \cdot 10^{-7}$
LOOP – other	-	$3 \cdot 10^{-8}$
Internal fire	-	$2 \cdot 10^{-8}$
Loss of pool cooling	$6 \cdot 10^{-8}$	$1 \cdot 10^{-8}$
Loss of coolant inventory	$1 \cdot 10^{-8}$	$3 \cdot 10^{-9}$
Inadvertent aircraft impacts	$6 \cdot 10^{-9}$	$3 \cdot 10^{-9}$
Missiles – general	$1 \cdot 10^{-8}$	-
Missiles - tornado		$< 1 \cdot 10^{-9}$
Pneumatic seal failures	$3 \cdot 10^{-8}$	-

[1] These numbers have not been multiplied by the stated conditional probability of having a Zirconium fire of 0.25.

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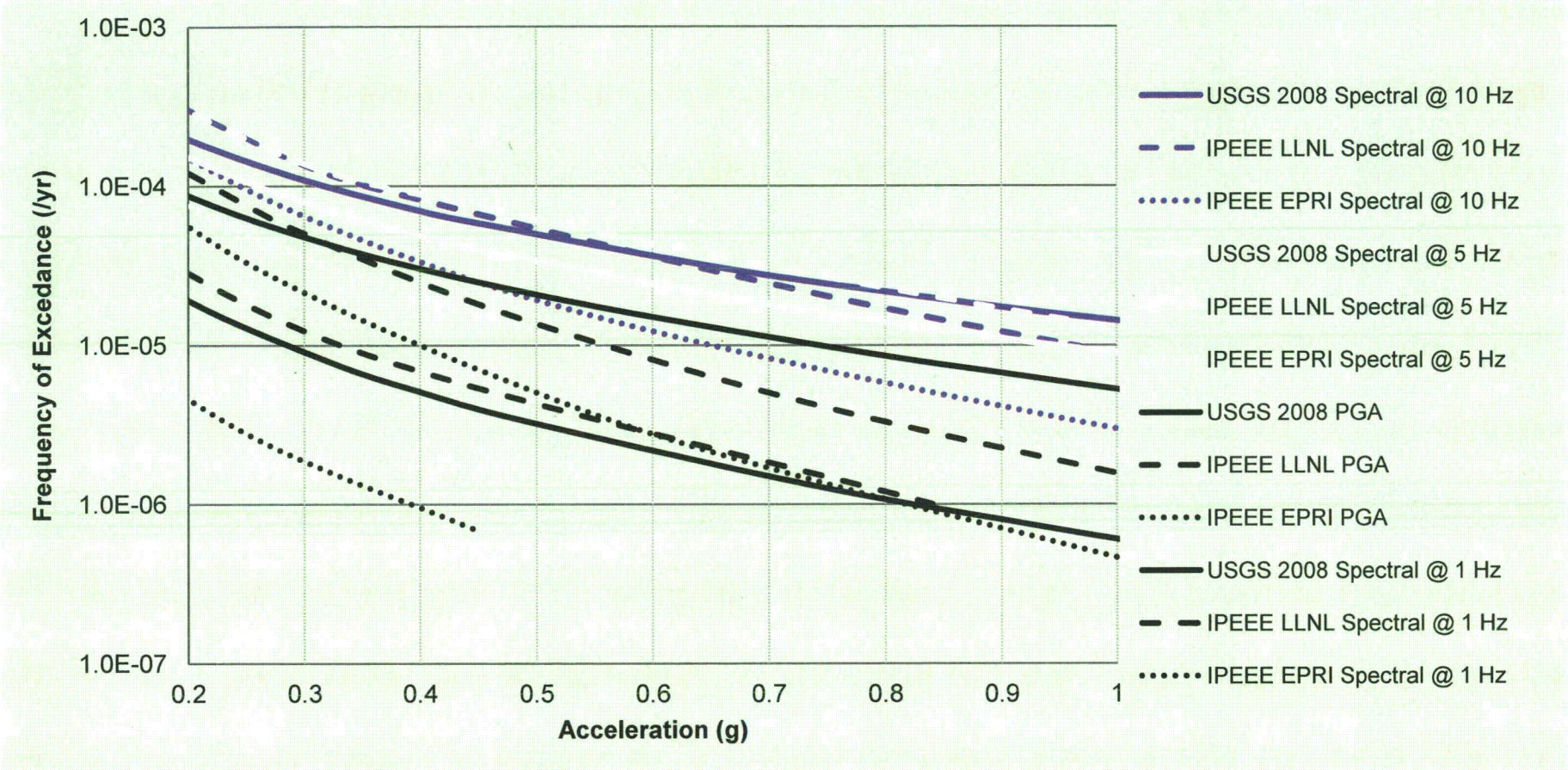
Peach Bottom Seismic Hazard
(for various spectra/information sources)



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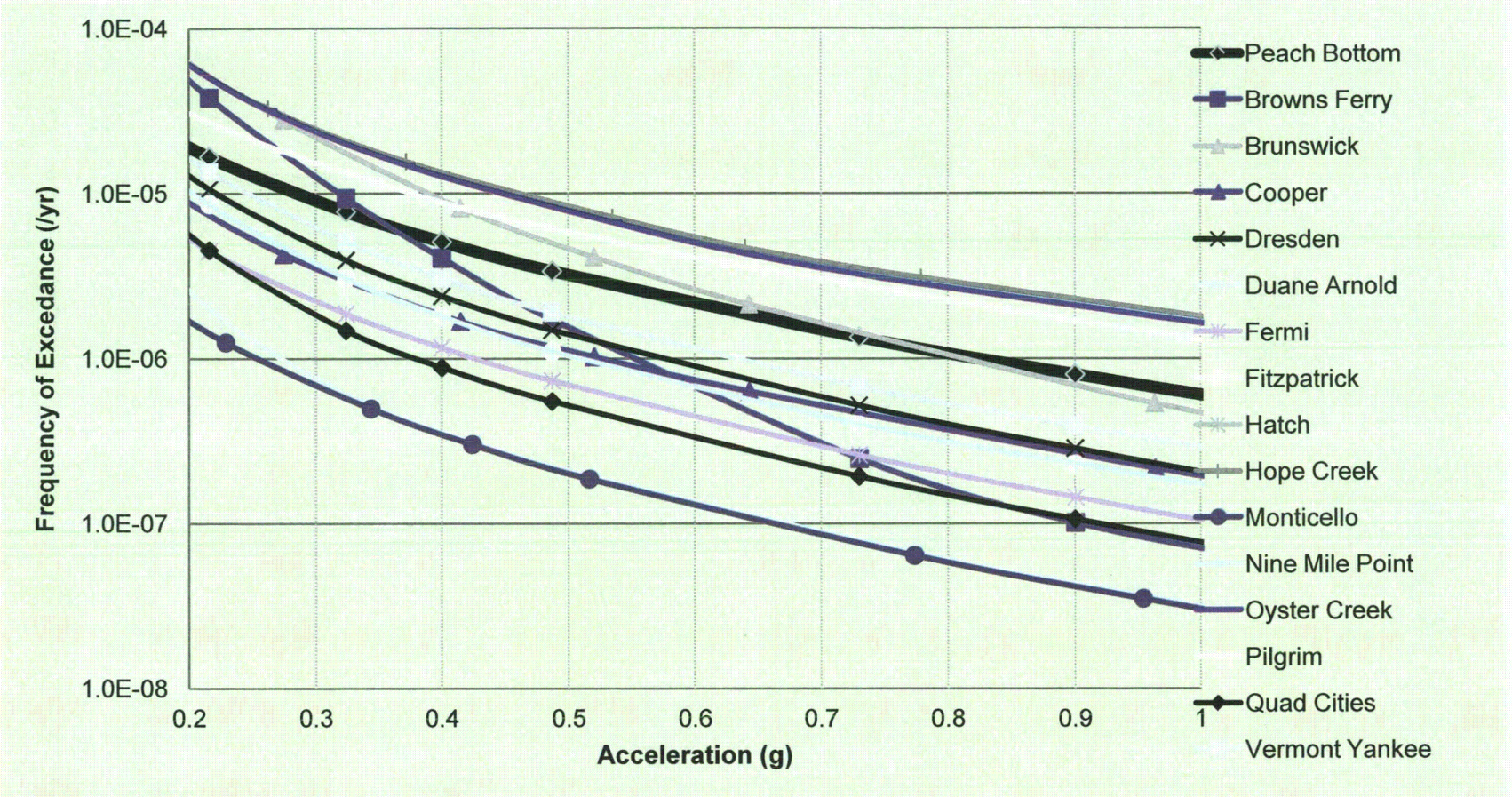
Peach Bottom Seismic Hazard versus Other Mark Is (based on pga)



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Seismic Bins and IE Frequencies (Based on USGS 2008 for Peach Bottom)

			~ Initiating Event Frequency for USGS 2008 (/yr)			
Bin #	**Bin**	**Bin Accel.**	**PGA**	**10 Hz**	**5 Hz**	**1 Hz**
1	0.05 < a < 0.3	0.12	5.20E-04	1.15E-03	1.12E-03	2.44E-04
2	0.3 < a < 0.5	0.39	2.70E-05	5.79E-05	4.44E-05	6.45E-06
3	0.5 < a < 1.0	0.71	1.65E-05	3.75E-05	2.67E-05	2.64E-06
4	1.0 < a < 2.1	1.2*	4.90E-06			
	1.0 < a < 5.0			1.43E-05		
	1.0 < a < 4.9				9.04E-06	
	1.0 < a < 3.7					6.26E-07

* Assumed based on modeling convention

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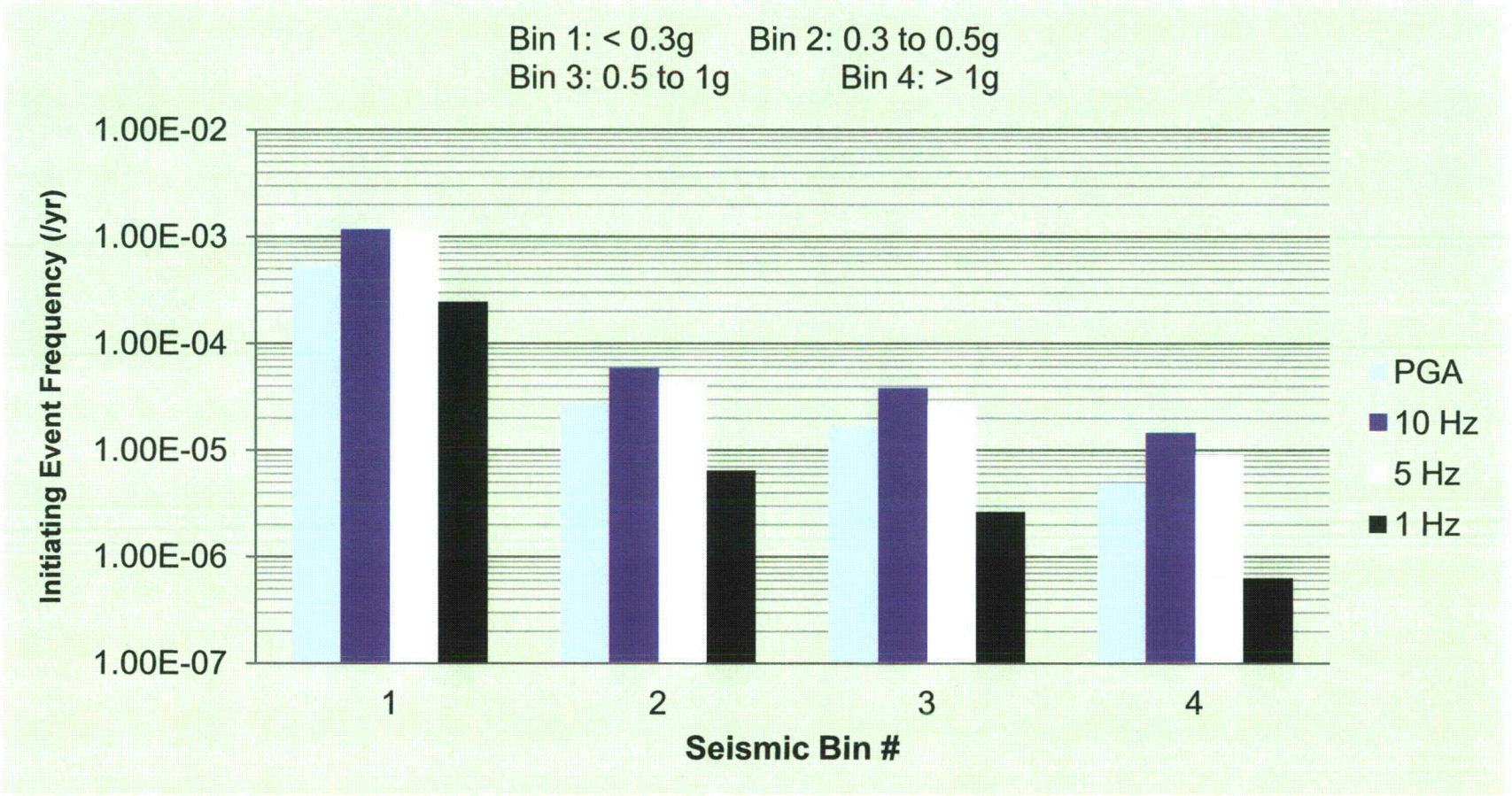
Seismic IE Frequency Comparison (Based on USGS 2008)



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Sample event tree
(For scenario delineation)

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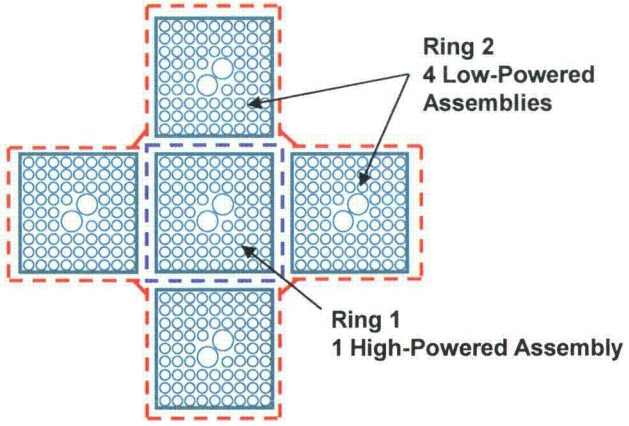
SFP MELCOR Models

- New records (SFP-BWR, SFP-PWR)
  - Rack component introduced in M1.8.5 version RO to model heat transfer within a ring and also from ring to ring
- Breakaway air oxidation model (ANL data)
- Hydraulic resistance model (SNL experiments)

Introduced in MELCOR 1.8.5 Version RO
"SFP-BWR" Model Representation of the 1x4 Model
(Cell wall is represented by new rack component)



Convective Heat Transfer Surfaces:

Ring 1 Clad, Canister / Water Rods, Rack | *Ring 2* Clad, Canister / Water Rods, Racks

Radiative Heat Transfer Flow Path:

Fuel → Clad → Canister / Water Rods → Rack | Canister / Water Rods → Clad → Fuel

Ring 1 | Racks | *Ring 2*

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MELCOR Versions

- M1.8.5 version RP (Dec 17 2007)
  - Used in the Whole Pool Model for Peach Bottom (SAND Letter Report, June 2008, Rev. 3)
  - New record CORHTR (user defined radiation path, 'RADIATE2')
  - Increase number of core control volumes (NCVOL > 100)

- M1.8.6 version YV (~ June 2011)
  - Main difference between M1.8.6 and M1.8.5 is introduction of curved lower head – requires input conversion
  - SFP models
  - Latest version with (NCVOL > 100)
  - CORHTR ('RADIATE2' implemented as default)

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Problem Benchmarking

- Focus on single accident sequence (10" bottom LOCA) for the high density loading (15 day aging)
- Converted M1.8.5 model to M1.8.6
- Additional records required to model lower head (a few had to be removed)
- Ring 7 channel in the M1.8.6 input had to be CV171 to be consistent with the new core models (consistency between COR/CVH cell volumes). Core channel was specified as CV299 in M1.8.5 model. Specifying CV171 as core channel in Ring 7 for M1.8.5 produces more consistent results with M1.8.6

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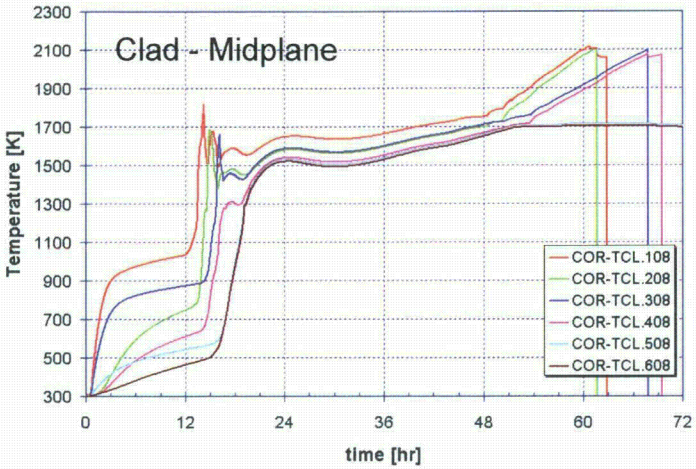
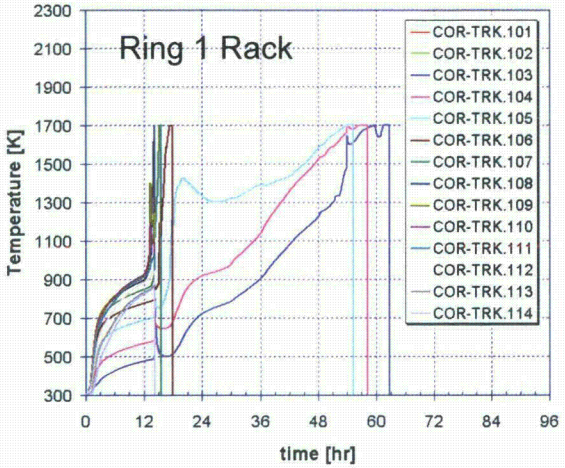
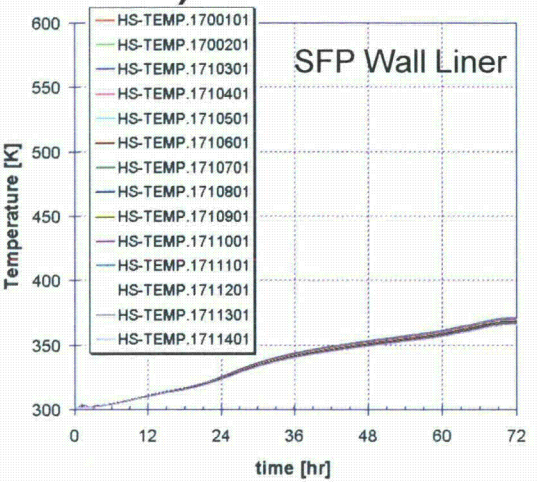
Benchmarking Results

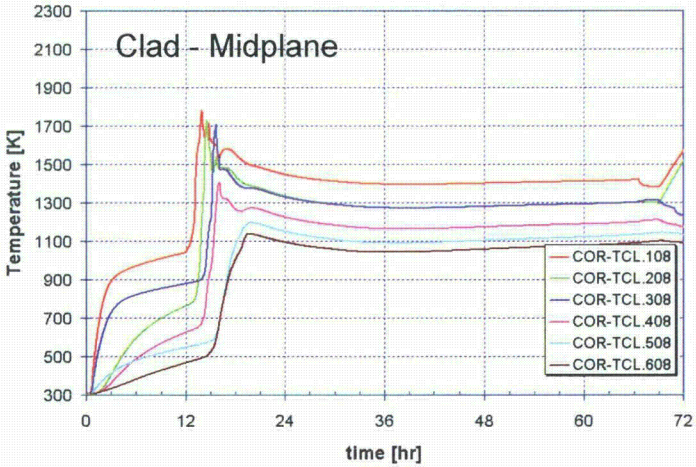
MELCOR 1.8.5 (Ring 7 Channel CV299)

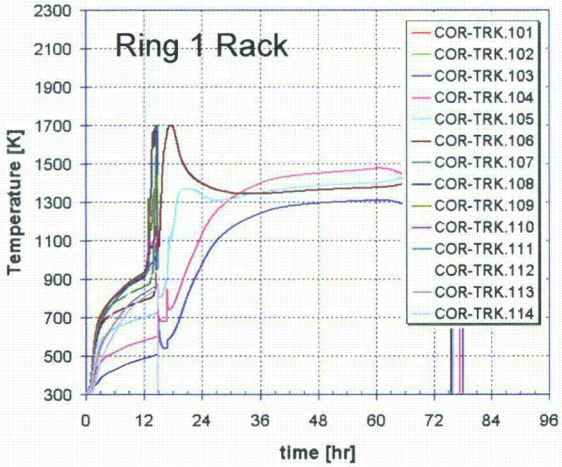


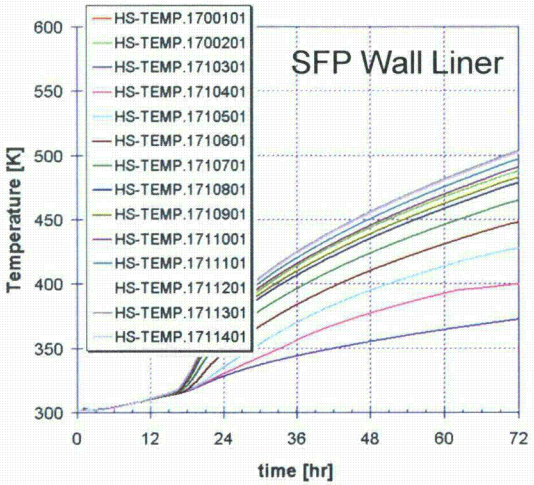




MELCOR 1.8.6 (Ring 7 Channel CV171)







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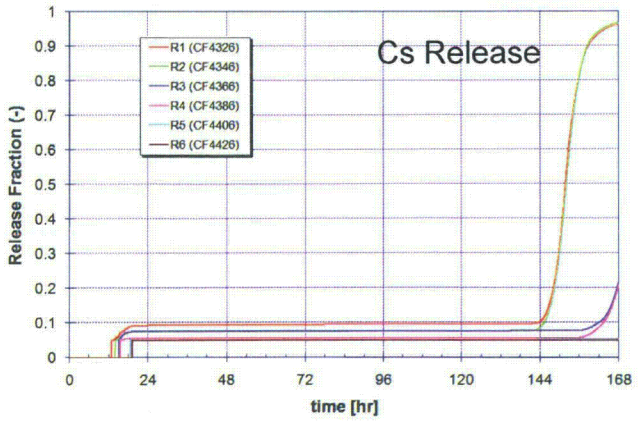
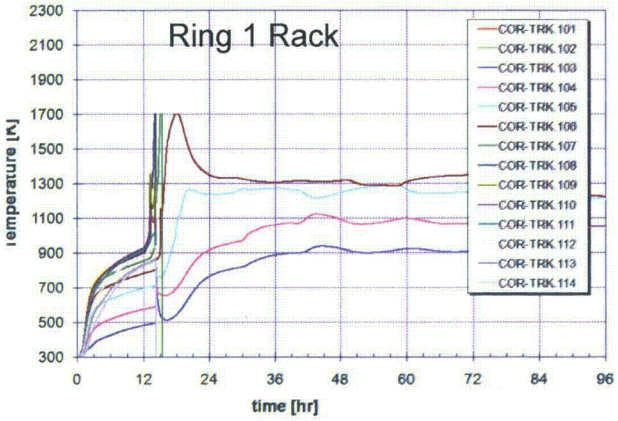
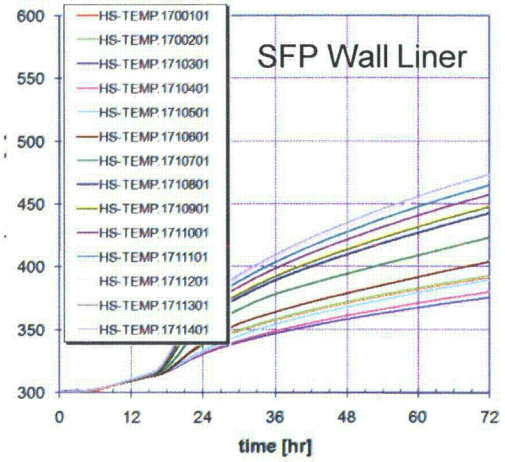
Benchmarking Results

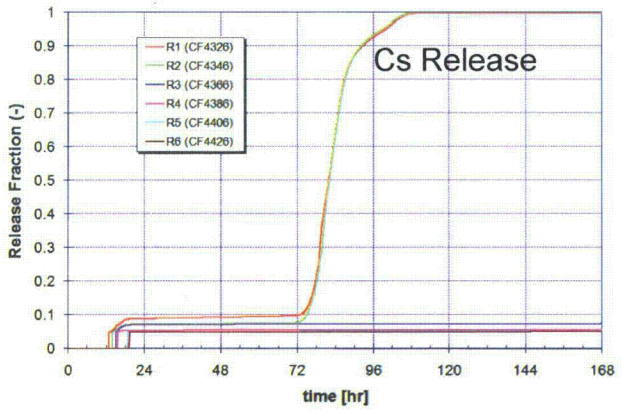
MELCOR 1.8.5 (Ring 7 Channel CV171)

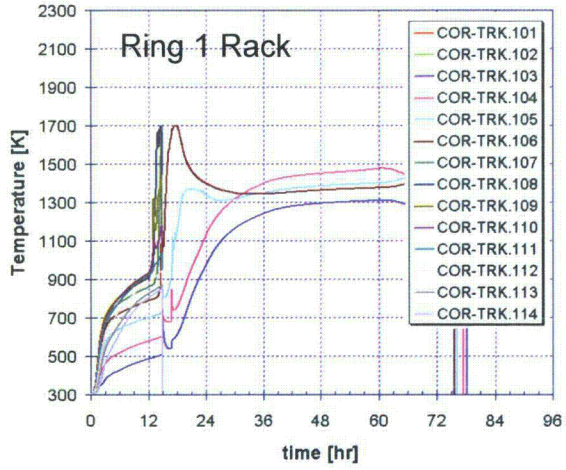


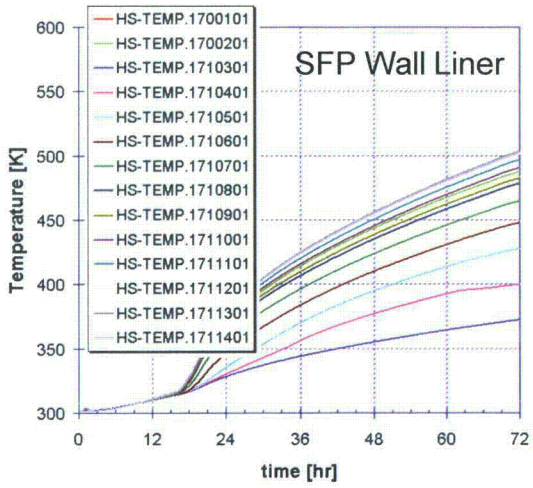




MELCOR 1.8.6 (Ring 7 Channel CV171)







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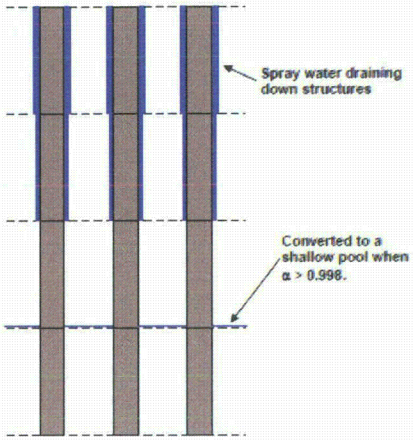
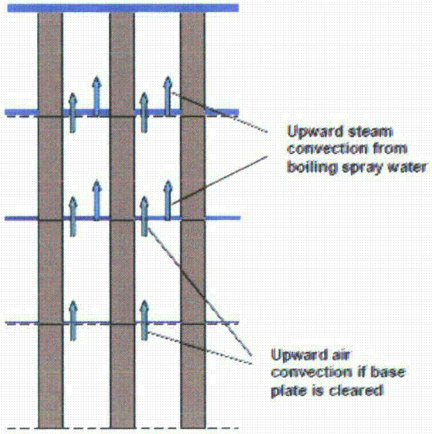
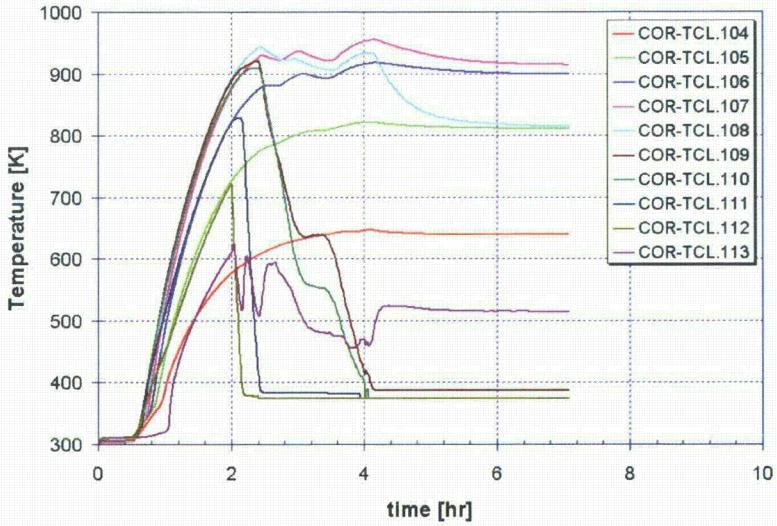
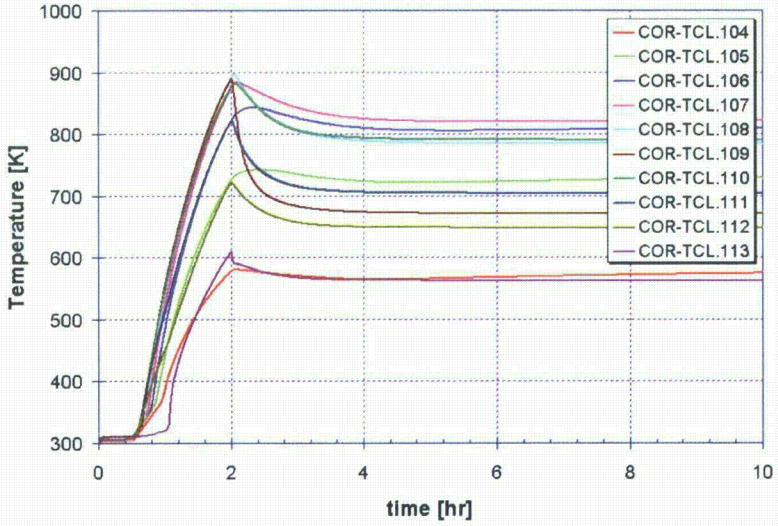
Spray Modeling Results

- Very large LOCA for high density loading (15 day aging)









U.S.NRC
UNITED STATES NUCLEAR REGULATORY COMMISSION
Protecting People and the Environment

Boil-off Results

- High density loading (15 day aging)