UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, DC 20555 - 0001

December 24, 2013

MEMORANDUM TO: ACRS Members

FROM: Christopher L. Brown, Senior Staff Engineer **/RA/**
Technical Support Branch
Advisory Committee on Reactor Safeguards

SUBJECT: CERTIFICATION OF THE MINUTES OF THE MEETING OF THE SUBCOMMITTEES ON MATERIALS, METALLURGY AND REACTOR FUELS ON TIER 3 EXPEDITED SPENT FUEL TRANSFER ON NOVEMBER 19, 2013, IN ROCKVILLE, MARYLAND

The minutes for the subject meeting were certified on December 24, 2013. Along with the transcripts and presentation materials, this is the official record of the proceedings of that meeting. A copy of the certified minutes is attached.

Attachment: As stated

cc w/o Attachment: E. Hackett
C. Santos

cc w/ Attachment: ACRS Members

UNITED STATES
NUCLEAR REGULATORY COMMISSION ADVISORY
COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, DC 20555 - 0001

MEMORANDUM TO: Christopher Brown, Senior Staff Engineer
Technical Support Branch
Advisory Committee on Reactor Safeguards

FROM: J. Sam Armijo, Chairman **/RA/**
Materials, Metallurgy and Reactor Fuels

SUBJECT: CERTIFICATION OF THE MINUTES OF THE MEETING OF THE SUBCOMMITTEES ON MATERIALS, METALLURGY AND REACTOR FUELS ON TIER 3 EXPEDITED SPENT FUEL TANSFER ON NOVEMBER 19, 2013, IN ROCKVILLE, MARYLAND

I hereby certify, to the best of my knowledge and belief, that the minutes of the subject meeting on November 19, 2013, are an accurate record of the proceedings for that meeting.

/RA/ **12/ 24 /13**

J. Sam Armijo, Chairman Date
Materials, Metallurgy and
Reactor Fuels Subcommittee

Certified on: December 24, 2013
Certifed by: J. Sam Armijo

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
MINUTES OF THE ACRS MATERIALS, METALLURGY AND REACTOR FUELS SUBCOMMITTEE MEETING

November 19, 2013

The ACRS Materials, Metallurgy and Reactor Fuels Subcommittee held a meeting on November 19, 2013 in TWF 2B3, 11545 Rockville Pike, Rockville, Maryland. The meeting convened at 830 a.m. and adjourned at 12:00 p.m.

The entire meeting was opened to the public.

ATTENDEES

ACRS Members

- J. Sam Armijo, Chairman
- Ron Ballinger, Member
- Joy Rempe, Member
- Michael T. Ryan, Member
- Stephen P. Schultz, Member
- William J. Shack, Consultant
- Gordon Skillman, Member
- Sanjoy Banerjee, Member
- Charles Brown, Member
- Harold Ray, Member
- Peter Riccardella, Member
- John Stetkar, Member

NRC Staff

- Christopher Brown, Designated Federal Official
- Andrew Barto, NMSS/SFST
- Hossein Esmaili, RES/DSA
- Kathy Halvey Gibson, RES/DSA
- Don Helton, RES
- Steven Jones, NRR/DSS
- A.J. Nosek, RES/DSA
- Jose Pires, RES/DE
- Don Algama, RES
- Kevin Witt, NRO/JLD
- Greg Casto, NRR
- Fred Schofer, NRR
- Tim McGinty, NRR
- Brian Wagner, RES
- Amy Cubbage, OCM
- Rob Taylor, NRR
- Ed Lyman, USC (via telephone)

SUMMARY

The purpose of the meeting was to discuss the staff's extension of the regulatory analysis contained in the Spent Fuel Pool Study (SFPS) reference plant to make it applicable to all Spent Fuel Pools (SFPs). The analysis assesses whether any significant safety benefits (or detriments) would occur from expedited transfer of spent fuel to dry casks for the reference plant as modeled, and the potential costs associated with such expedited transfer.

In SECY-12-0095, the staff submitted a plan to evaluate whether regulatory action is warranted for the expedited transfer of fuel from spent fuel pools to DCSSs. In a memorandum entitled, "Updated Schedule and Plans for Japan Lessons-Learned Tier 3 Issue on Expedited Transfer of Spent Fuel," dated May 7, 2013, the staff updated plans to address Commission directions in staff requirements memoranda (SRMs) M120607C and M120807B to assist in the Tier 3 decision process.

SIGNIFICANT ISSUES	
Issue	Reference Pages in Transcript
1. Staff stated that the regulatory analysis was performed in a conservative manner to maximize what the benefits would be if fuel moved out of SFP from high-density to a low-density configuration. Member Banerjee asked was the regulatory analysis conservative or was it based on best- estimates. In response, the staff indicated that the analysis was performed with a mixture of both conservative assumptions and best estimates. Some assumptions were made out of convenience. Member Banerjee and Chairman Armijo expressed that this was confusing. Staff discussed the assumptions from Table 2 in the COMSECY.	10-13, 50-55, 66-92-122, Regulatory Analysis Table 2
2. Chairman Armijo raised a question concerning seismic analysis in the generic analysis in the SFPS versus the seismic analysis in the regulatory analysis. In particular, he wanted staff to explain why they needed to increase the seismic breadth of the plant over what was analyzed in the SFPS. This question was followed up by additional questions and concerns from Member Stetkar relating to frequency of failures and range of accelerations.	17-23
3. Member Skillman asked about the seismic risk for SFPs for Western Plants. He said what if these plants do not conform to the regulatory analysis. In response, the staff considers the western plants (seismically active areas) to also have robust designs and that the public health and safety are adequately protected.	25-28
4. Concerning seismic initiator frequency assumptions sensitivity, Member Rempe asked about the inconsistencies in the linear fragilities in the COMSECY versus the slides. Other issues relating to fragilities were discussed.	54-56, 81-101

5. The staff issued Order EA-12-049, "Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events," which requires licensees to develop, implement, and maintain guidance and strategies to maintain SFP cooling capabilities, independent of alternating current power, following a beyond-design-basis external event. For the purpose of evaluating the potential benefits of expedited transfer of spent fuel to dry cask storage, the staff used a conservative approach to mitigation by crediting successful mitigation to the low-density SFP storage alternative (i.e., conditions following expedited transfer) and assumed no successful mitigation for the high-density SFP storage regulatory baseline. Chairman Armijo and Members Schultz and Stetkar raised questions relating to mitigation. In particular, the issue of assigning effective mitigation only to the alternative and not to the base case.	58-65
6. Discussions about release fractions. SFPS and previous studies demonstrated that cesium release fractions are generally less in the SFPS when compared to previous studies.	52-56, 60-61, 65, 73-75
7. Sensitivity studies were conducted on key factors such as the dollars per person-rem conversion factor, population density, habitability criteria and consideration of consequences beyond 50 miles. Member Stetkar asked about evacuation plans and evacuation time estimates. In particular, how important are those to the overall results? In response, staff discussed the models used for the sensitivity analysis. The analysis used key insights from operating experience, the October 2013 SFP study, and previous studies on SFP safety.	78-83
8. Member Ballinger mentioned that the study was based on carbon steel and that the properties of carbon steel are different than stainless steel. In particular, the toughness of stainless steel is much higher than carbon steel. Member Ballinger and Chairman Armijo questioned what properties were used for the study. In response, staff indicated that the failure strains were conservative. Member Ballinger discussed the content of NUREG-6706 which contains data and information on steel and concrete containment vessels with corrosion damage.	81-87
9. The staff considers the base case an appropriately conservative analysis for use as the primary basis for the staff's recommendation that additional studies not be pursued and Tier 3 issue be closed. Members asked questions about the following: 1) bin 4 (The SFPS used four bins), 2) mitigation affecting the heat-up frequency, and 3) populations and habitability.	104-124
10. The staff used the quantitative health objectives (QHO) in conducting its safety goal screening evaluation. The QHOs are used as a surrogate for the safety goal as outlined in the Commission's safety goal policy statement. Although the QHOs were developed based on the risk from severe reactor accidents, they provide the only readily available risk criteria for regulatory decision making regarding non-reactor accidents. Chairman Armijo and Member Schultz asked about the QHOs calculation. Staff believes that QHOs are appropriate with the measure in place for SFP.	133-140
11. Public comments were made by Edward Lyman, Union of Concerned Scientist.	153-155

Documents provided to the Subcommittee

1. Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor, Appendix D.
2. COMSECY-13-0030, "Staff Evaluation and Recommendation for Japan Lessons-Learned Tier 3 Issue on Expedited Transfer of Spent Fuel," November 12, 2013, (ML13273A601)
3. ACRS Letter, Subject: Report on the Spent Fuel Pool Study, July 18, 2013 (ML13198A433)
4. ACRS Letter, Subject: Report on the Spent Fuel Pool Scoping Study, April 25, 2012 (ML12108A216)
5. NRC Staff Requirements M120607C - Meeting with the Advisory Committee on Reactor Safeguards, July 16, 2012 (ML121980043)

Official Transcript of Proceedings

NUCLEAR REGULATORY COMMISSION

Title: Advisory Committee on Reactor Safeguards
Materials, Metallurgy and Reactor Fuels

Docket Number: (n/a)

Location: Rockville, Maryland

Date: Tuesday, November 19, 2013

Work Order No.: NRC-426 Pages 1-160

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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

(ACRS)

+ + + + +

MATERIALS, METALLURGY, AND

REACTOR FUELS SUBCOMMITTEE

+ + + + +

TUESDAY

NOVEMBER 19, 2013

+ + + + +

ROCKVILLE, MARYLAND

+ + + + +

The Subcommittee met at the Nuclear Regulatory Commission, Two White Flint North, Room T2B1, 11545 Rockville Pike, at 8:30 a.m., J. Sam Armijo, Chairman, presiding.

COMMITTEE MEMBERS:

J. SAM ARMIJO, Chairman

RONALD G. BALLINGER, Member

SANJOY BANERJEE, Member

CHARLES H. BROWN, JR. Member

HAROLD B. RAY, Member

JOY REMPE, Member

PETER C. RICCARDELLA, Member

MICHAEL T. RYAN, Member

STEPHEN P. SCHULTZ, Member

GORDON R. SKILLMAN, Member

JOHN W. STETKAR, Member

DESIGNATED FEDERAL OFFICIAL:

CHRISTOPHER L. BROWN

ALSO PRESENT:

EDWIN M. HACKETT, Executive Director, ACRS

TIM MCGINTY, Division Director, NRR

RAJ AULUCK, NRR

STEVEN BAGGETT, COMM

ANDREW BARTO, NMSS

PATRICK CASTLEMAN, OCM

GREG CASTO, NRR

AMY CUBBAGE, OCM

HOSSEIN ESMAILI, RES

KATHY HALVEY GIBSON, RES

STEVEN JONES, NRR

IAN JUNG, OEDO

ED LYMAN, UCS*

JOSE PIRES, RES

BILL RECKLEY, NRR

FRED SCHOFER, NRR

ROBERT TAYLOR, NRR

BRIAN WAGNER, RES

KEVIN WITT, NRR

*Present via telephone

T A B L E   O F   C O N T E N T S

P R O C E E D I N G S

8:30 a.m.

CHAIRMAN ARMIJO: Good morning. The meeting will now come to order. This is a meeting of the Materials, Metallurgy and Reactor Fuels Subcommittee. I am Sam Armijo, Chairman of the Subcommittee.

ACRS members in attendance are Sanjoy Banerjee, Ron Ballinger, Harold Ray, Dick Skillman, Steve Schultz, John Stetkar, Mike Ryan, Charlie Brown and Joy Rempe.

I expect Pete Riccardella will show up, but he hasn't yet. Christopher Brown of the ACRS staff is the designated federal official for this meeting.

Today's meeting is open to the public. The purpose of the meeting is to receive a briefing from the Office of Nuclear Reactor Regulations on staff evaluation and recommendation for Japan lessons learned Tier 3 issues on expedited transfer of spent fuel.

The Subcommittee will gather information, analyze relevant issues and facts and formulate proposed positions and actions as appropriate for deliberation

by the full committee.

The full committee meeting on this topic will be on December the 5th, 2013, and will also be open to the public.

The rules for participation in today's meeting were previously published in the Federal Register.

We have received no written comments or requests for time to make oral statements from members of the public regarding today's meeting.

A transcript of the meeting is being kept and will be made available as stated in the Federal Register Notice.

Therefore, we request that participants in this meeting use the microphones located throughout the meeting room when addressing the Subcommittee.

Participants should first identify themselves and speak with sufficient clarity and volume so that they can be readily heard.

I'd like everyone to please silence their cell phones at this time. And also, it is my understanding that members of the public, Mr. Ed Lyman, may be on the bridge line. And the bridge line will be set in listen-only mode during the briefing. After the briefing, we will open the bridge line for public

comments.

We will now proceed with the meeting and I call on - I believe it says Tim McGinty. He will open up the meeting for us and give a brief introduction.

MR. McGINTY: Thank you. Good morning. My name is Ted McGinty, and I'm the director of the Division of Safety Systems in the Office of Nuclear Reactor Regulation at the NRC.

I would like to thank the Chairman and the members of the ACRS for the opportunity to hear the staff's presentation of the near-term task force Tier 3 action to recommend whether further regulatory action is recommended or additional study would be warranted regarding the expedited transfer of spent fuel from wet to dry storage.

To determine whether regulatory action might be warranted, we followed our regulatory decision-making procedures to determine whether there is a substantial safety enhancement.

Additionally, to provide information to the Commission, the staff performed additional cost-benefit analysis, as well as additional sensitivity studies for cases beyond the current regulatory framework.

Based on the feedback that you provided in your October 3rd full committee meeting on the draft

analysis, the staff has reviewed their earlier work and made a number of improvements based on your comments. The staff issued COMSECY-13-0030 to the Commission on November 12th.

For our meeting with you today, Kevin Witt will be covering the Tier 3 plan background and evaluation process, Steve Jones will be covering the Tier 3 analysis, and Fred Schofer will be supporting the discussions on the cost-benefit analysis.

And with that said, I'll turn it over to Kevin Witt.

MR. WITT: Thank you. As Tim said, my name is Kevin Witt. I'm the project manager in the Japan Lessons Learned Project Directorate. I was responsible for coordinating the staff activities on this issue.

Today during our presentation, we'll be going over these following items. I'll be giving a brief background of this issue and talk about the process that we went through on the evaluation of this.

And Steve is going to talk about the regulatory analysis. And Fred will help us out in our discussions.

A little bit of background on how we got here. This issue was identified following the Fukushima accident where there were stakeholder

concerns about spent fuel storage and spent fuel pools.

And the issue came up as to whether spent fuel pools would be safer, and this has been an issue that's been around for quite some time in terms of whether spent fuel pools are safe in a high-density configuration and whether they would be safer in a low-density configuration.

So, what we did following the identification of this issue following the Fukushima incident, is we tried to determine what the best way to determine whether regulatory action might be warranted.

And we have a normal process for doing this on our regulatory analysis guidelines that are outlined in the NUREG/Brochure-0058. And so, this process kind of lays out how we did this analysis.

So, during this analysis we utilized a lot of previous information that we had about spent fuel pools. There's been a broad history of studies on spent fuel pool safety. There was a generic issue back in the 1980s on high-density spent fuel pools.

We also had the Spent Fuel Pool study which was started following Fukushima and we utilized information from that study for our analysis.

So, what the purpose of the paper that we

just recently sent up to the Commission was a high-level look at whether regulatory actions might be warranted. And if our analysis showed that regulatory action might be warranted, then we would do additional studies.

So, when we came up with the plan for this issue, we split it up into three phases. And this first phase is the one that we just recently completed the Commission paper on. It's what we call "Phase 1."

And it's really a conservative analysis. We picked all of our assumptions in a conservative manner to try to maximize what the benefits would be if you did indeed move the fuel out of the pools, move from high-density to low-density spent fuel pools.

MEMBER BANERJEE: Was it a purely conservative analysis, or did it have certain best estimate elements?

MR. WITT: We tried to do it in a conservative manner, but there were a number of places where we did best estimate.

Steve, do you want to -

MEMBER BANERJEE: It seems that it was mixed, right?

MR. JONES: Yes, it is a mix and we'll get to it in detail, I guess, a little later on.

MR. WITT: Yeah, when we talk about all the

inputs that we used, we can see how - we'll talk about how we picked those.

MEMBER BANERJEE: Why did you choose this path instead of either doing something like a best estimate or a very conservative? I mean, this is sort of neither fish nor fowl in some of it.

MR. WITT: Well, really our objective with this analysis was to try to skew it as much as possible towards going further down the road and doing further study on regulatory actions.

So, we tried to figure out whether it would theoretically be possible to have a substantial safety enhancement by having less fuel in the pool than it is currently.

And so, in order to do that, we try to maximize the benefits that we could get out or what type of safety benefits there would be for moving from a high to a low-density pool.

MEMBER BANERJEE: I think the reason is clear. So, I don't want to belabor this, Mr. Chairman. But on the other hand, it is confusing because of the way - it's not clear which assumptions are which. They're not pinned and justified in a way which is -

CHAIRMAN ARMIJO: I share your concern, Sanjoy. The problem is when you're trying to maximize

a benefit of an alternative, the question is how far can you go before you actually create an unrealistic or unjustified benefit when it really isn't there.

And we have that problem with this type of an analysis as opposed to a best estimate, plus some uncertainties are higher and lower.

As you go through your presentation, you just have to keep that in mind that's a concern.

MR. WITT: Right.

MEMBER BANERJEE: And so, if you look at your detailed studies and so on, they were done in great depth in some ways that I must compliment you in that work as well. I think other people might feel that way, but they looked more like in some ways best estimate calculations that you've done if I recall all the materials you put in those.

Now, you can say that maybe the incident was shifted by an hour or two or whatever if you make more conservative estimates, but that's sort of hand waving, you know.

So, this mixture of best estimate and conservative really continues to trouble me on this.

MR. JONES: I'll try to give you a good explanation when we -

MR. WITT: Yeah, another viewpoint on that

is I think that if this analysis did indeed show that there might be a benefit, then we would try to do more of a best estimate analysis on the next phase, but -

MEMBER BANERJEE: Or either, I mean, be conservative. That's fine. Okay. Thank you.

MR. WITT: So, this plan was provided to the Commission back in May of 2013. It outlined the three-phase process that we proposed to follow for this issue.

In terms of stakeholder involvement in our analysis of this issue, we did have two public meetings this past summer. The first was on August 22nd, and the next one was on September 18th.

The first public meeting mainly discussed the Tier 3 issue of expedited transfer and we received some feedback that stakeholders wanted to have some more dialog on the spent fuel pool study. So, we had another meeting on September 18th to talk about the spent fuel pool study and the Tier 3 issue.

There has been a lot of feedback from stakeholders. We received a number of letters on this issue.

Most of the external stakeholder feedback that we have received generally indicates their favor for moving forward with expedited transfer of spent

fuel, but it also outlines a little bit of confusion in terms of what the process we followed was.

We really tried to do our best in terms of revising the document from the draft version to what you have currently, what we sent up to the Commission to lay down in a more logical manner.

The spent fuel pool study as I mentioned several times before, was a major element of this analysis.

We started really doing this analysis with the spent fuel pool study and I'll talk about that in another slide, but this was carried out by the Office of Research. NRR was heavily involved with the conduct of that study. So, there was a lot of collaboration between Research and NRR in terms of how the study was conducted and also on the Tier 3 analysis.

The spent fuel pool study was issued for public comment in June 2013 and that was just recently finalized and sent up to the Commission in October. And the final version had the public comments they received, as well as responses to those public comments.

In terms of the Tier 3 analysis, we did release a draft version of that document before the ACRS meeting which we had with you on October 2nd.

That document was released to support

stakeholder involvement in that meeting or some stakeholders who presented at that meeting.

We did receive, like I said, we received some feedback from individuals and from you about the outline that we had in that document. So, we really took that back and tried to do our best to respond to those concerns or those comments that we received and tried to lay this out in a more logical manner.

So, that's really what you'll see the difference between what we released, what you had back in October and what we just recently sent up to the Commission.

We tried to reformat it to lay it out in a more consistent format in terms of what the process we followed was.

So, this slide gives an overview of the steps that we took to get to this Tier 3 analysis. The bottom level here is the spent fuel pool study. And that was a study to identify the potential consequences of a spent fuel pool accident at a representative plant.

It was really focused on one plant and talked about one specific event that occurred. So, it really went quite in-depth in terms of how the accident progression can occur at a spent fuel pool.

So, subsequent to the completion of the

calculations and all that stuff, we went ahead and added on an appendix to that study. It's Appendix D to the spent fuel pool study. It was a regulatory analysis of that representative spent fuel pool.

And what we did is we wanted to see how those consequences would fit into our regulatory framework. It was kind of like the first step towards getting towards a generic regulatory analysis.

So, the appendix of that study outlined how the consequences from that spent fuel pool study would fit into our regulatory framework in terms of whether there was a substantial safety benefit, and a cost-benefit analysis as well in there.

There was an expanded set of scenarios that that regulatory analysis considered in the spent fuel pool study.

And then finally at the top of this which we're talking about today is the Tier 3 analysis. And that expands it out to all of the plants with the broad side of initiating events that we considered in the analysis.

MEMBER STETKAR: Are you going to talk a little bit about that broad side of initiating events that you considered?

MR. WITT: Yes.

MEMBER STETKAR: Okay.

MR. WITT: Yes.

MEMBER STETKAR: I'll hold my questions until then.

MR. WITT: Yeah, there's a slide on that.

CHAIRMAN ARMIJO: Kevin, I don't remember was it in Appendix D that you broadened the seismic from a 0.7 g to a 1.2 g for the spent fuel pool study?

MR. WITT: Yes.

CHAIRMAN ARMIJO: It was in there?

MR. WITT: Right.

CHAIRMAN ARMIJO: And then you'll use that same set of seismic events in the generic analysis.

MR. WITT: Correct.

CHAIRMAN ARMIJO: Okay. Somewhere along the line if you could explain why you needed to increase the seismic breadth of the plant over what was analyzed in the spent fuel pool study itself, I mean, what was the reason?

Because that was a very, you know, six times the normal design basis.

MR. WITT: I think we can talk - we don't really have a slide about the spent fuel pool study in terms of what we did.

CHAIRMAN ARMIJO: No, we reviewed that. You

don't have to go into that again.

MR. WITT: Right.

CHAIRMAN ARMIJO: But then why did you feel an obligation in the regulatory analysis to crank up the seismic loading?

Was it arbitrary? Was it based on some desire to maximize the -

MR. WITT: It is actually a conscious decision. I believe -

CHAIRMAN ARMIJO: No, I understand it was a -

(Simultaneous speaking.)

CHAIRMAN ARMIJO: No, I'm just saying I'm trying to understand why did you do it?

MR. WITT: That did come out from the formulation of the spent fuel pool study. Because when they first started doing that, they were considering what type of seismic events they were going to analyze in there.

CHAIRMAN ARMIJO: Yeah.

MR. WITT: And so, they had to decide between the Bin 3 and Bin 4 earthquakes. And I guess - I don't know for what reason they chose the Bin 3, but they did identify the Bin 4 in the - when they were starting that study.

And so, we kind of took that from the study itself in terms of how they discussed that Bin 4 earthquake in terms of it being a possibility.

MR. STETKAR: Kevin, maybe at the break you guys can get together and assemble a little bit of background material on this, because I have questions different from what Dr. Armijo has, because your analysis actually underestimates the seismic risk for base case analysis, because you've limited only Bin 4, which has a particular frequency and there's only a 50 percent probability that the two, three, four groups' liner fails at that acceleration.

So, you have not accounted for frequencies - when you convolute the frequency of higher accelerations with the fragilities past the median capacity, you've not accounted for that damage frequency. You've not, you know.

So, you've arbitrarily truncated the upper end at an acceleration that does not span the range of fragilities of the Group 2, 3 and 4 pools.

So, I'd like to better understand why you stopped at Bin 4, why you don't have a Bin 5, because you don't have the frequency of those large earthquakes for which the pools would fail.

CHAIRMAN ARMIJO: Well, John, you totally

confused me.

(Simultaneous speaking.)

MEMBER STETKAR: I do seismic analysis.

CHAIRMAN ARMIJO: Yeah, I understand.

MEMBER STETKAR: And risk analysis is just selecting a couple of arbitrary bins. And what bothers me is there are statements in there that says, well, we assigned 1.2 g to Bin 4 based on PRA convention, and that is not PRA convention.

That is a gross misrepresentation of what is done in modern seismic risk assessments. And that's on the record now.

So, I'd like to really understand why you stopped where you stopped with Bin 4, and why you characterized it the way you did it considering the fragilities that you used for the Group 2, 3 and 4 pools.

Group 1 is fine, because it's guaranteed to fail at Bin 4. Groups 2, 3 and 4, the fragilities are only 0.5. It's a medium capacity of those liners. They will fail at some higher acceleration.

You have not quantified the frequency of that higher acceleration.

MR. WITT: Well, we can definitely have that discussion.

MEMBER STETKAR: Okay. So, you know, maybe

at the break you can think about it a little bit.

MR. WITT: Yeah, unfortunately our seismic expert isn't here yet.

MEMBER STETKAR: Well, that's too bad, because all of this is seismic.

MR. WITT: Yeah, he's going to be here in a little bit.

CHAIRMAN ARMIJO: I think, you know, I want to understand that, you know, we obviously can make anything fail if we crank up the seismic loading sufficiently.

And the question I have is, where do you stop and - to be realistic, these are, you know, we don't want to have a - you find a situation where we just force an answer being the only alternative is -

MEMBER STETKAR: If the frequency of failure is small enough, then it's small enough. If the frequency of failure is not small enough, then it's not small enough. If you've not looked at it, you don't know what the frequency of failure is.

CHAIRMAN ARMIJO: You're presuming it's -

MEMBER STETKAR: I'm not presuming anything. I don't know, because it's not been evaluated.

I don't know the steepness of the assumed seismic fragility curve, nor do I know the shape of the

- I do know the shapes of the seismic -

CHAIRMAN ARMIJO: If we have those curves and we don't have to invent them on the fly, then I think it's a good point. But if it's very high uncertainty, I'm just -

MEMBER STETKAR: There's two issues. One is uncertainty, one is completeness of the range of accelerations in -

CHAIRMAN ARMIJO: You've got to educate some of us that - your opening statement was very complex.

MEMBER STETKAR: It's not easy if you don't - if you do seismic analysis, you know what I'm talking about. If you don't do seismic analysis, you don't know what I'm talking about.

CHAIRMAN ARMIJO: Okay. The staff knows what John is talking about. Okay. Thank you. We'll move on.

MEMBER BALLINGER: The seismic analysis is not disconnected from the fragility and the assumptions you've made there.

So, as you raise the - as you increase the energy in the seismic event, you run up against the conservatisms that you've made with respect to the fragility, especially the pool mechanical properties and those assumptions.

And so, they're not disconnected from one another.

MEMBER STETKAR: No, they're not disconnected. That's my point.

MEMBER BALLINGER: And we have to be sure that we get them both right. Otherwise, we get hoisted by our own petard here.

MR. WITT: I don't want to jump ahead, but I just want to respond quickly that what our analysis showed is that the dominant frequencies are - or what we're talking about in terms of the safety enhancements are really dominated by the event initiator frequencies.

So, that's one of the major contributors.

MR. SCHOFER: That's dominated by seismic.

(Simultaneous speaking.)

MR. WITT: We all agree it's dominated by seismic. Now, the question is -

MEMBER BALLINGER: But it's the liner that fails.

MR. WITT: Correct.

MEMBER BALLINGER: We'll get back to it.

MR. WITT: Yeah, we'll talk more about that.

CHAIRMAN ARMIJO: Okay.

MR. WITT: Okay. So, this is a little bit more about how we did our analysis. We had a broader

set of initiating events in this Tier 3 analysis, as well as the spent fuel pool studies talk about a more severe earthquake, a Bin 4 earthquake, cask drop events, loss of power, loss of coolant inventory.

The Two, Three analysis, as I said before, covers all the reactors in the central and eastern. We did have one caveat in the paper that we sent up to the Commission in that we did not have updated seismic hazard information for the west coast plants, which they are currently working on updating as part of the Japan Lessons Learned 2.1 activity, the seismic reevaluations.

So, what we committed to the Commission is that we're going to go back after the completion of those reevaluations for the west coast plants and look to make sure that they are consistent with the analysis that we conducted in this Tier 3 evaluation.

We also did in our analysis, we covered new reactors, the AP-1000s. That was one of the groups.

And then one of the issues that we've heard about numerous times, a number of stakeholders have brought this up, is security.

And we did have a statement in the paper that we sent up to the Commission that security is not considered in this analysis. It's handled through our

existing processes for security.

And we gave out - there was some effect of security in this analysis in that in the alternative we did consider the mitigating strategies, the B.5.B or the 50.54(hh) equipment. So, that was included in our analysis. We did credit that equipment, as well as the security changes in the regulatory baseline.

So, we pretty much assumed that security is going to be perfect in our analysis.

MEMBER SKILLMAN: Kevin, before you change the slide, let me ask you about the western plants. In the draft letter to the Commissioners - excuse me - in the November 12th letter to the Commissioners, Mark's comment on Page 8, Mark Satorius' comment is at the completion of the NTTF recommendation 2.1 seismic reevaluation, the staff will confirm that the seismic risk for SFPs is consistent with that considered in the enclosed analysis.

And I'm following up on the statement that you made that this be revisited. It sounds to me like the analysis is being closed out on a future promise.

MR. WITT: Well -

MEMBER SKILLMAN: And please explain to us what's going to happen if as a result of the 2.1 evaluations the western plants don't conform with your

analysis.

MR. WITT: Well, I believe that it will be handled through that process in terms of the 2.1 reevaluation process.

MEMBER SKILLMAN: What's that do to your analysis, the risk analysis?

MR. JONES: I guess we're trying to - we are following to a large degree past studies. And NUREG-1738 and NUREG-1353 have the same issues with the west coast plant seismic fragility or seismic hazard information.

So, we are expecting to confirm the same response or similar response as the west coast pools, but there's a difference. And we certainly have to consider that for plant-specific backup process or some other action appropriate for that risk that's identified.

MEMBER SKILLMAN: Okay. Thank you.

MEMBER REMPE: Are there any increases in security costs if you have to expedite transfer because you're going and building new ISFSIs and things like that and there's more cameras and guards for these new facilities, and were they considered?

MR. JONES: There's a discussion in the regulatory analysis about operating costs. Most of the

existing reactors have ISFSIs and we're not assuming that they dramatically need to increase operational expenses.

There might be additional cost for expansion of an existing ISFSI by putting in a new pad or something.

On the other hand, for the new reactors, the Group 3 plants, there was consideration of additional operational costs beginning much earlier in the life of the plant, because there would be earlier transfer.

MEMBER REMPE: The judgment if you thought it was significant, you did consider it. But with the existing plants, you didn't think it was significant.

MR. JONES: The operational costs are - we didn't consider significant.

MR. WITT: Yeah, I would say just a general overarching plan that we had in mind was that we weren't going to try to make it more - I think something like that in terms of additional security would kind of make it more beneficial than cost more. the costs would go up.

So, what we are trying to do is we were not really considering the additional things like, for instance, the risks - additional risks associated with

expedited transfer in terms of loading more casks than a plant would normally - we didn't include those risks in our study to try to maximize the benefits that you would get out of the expedited transfer.

And so, for something like the security associated with expedited transfer, I think it was a conscious decision to not add that in to try to see whether the benefits would still surpass the costs.

MEMBER REMPE: Okay.

CHAIRMAN ARMIJO: You're going to talk about your cost-benefit analysis later, right?

MR. WITT: Correct.

CHAIRMAN ARMIJO: Okay. I'll wait for it.

MR. WITT: Okay. This slide talks about the process that we followed to evaluate this Tier 3 issue. And this was a direct result of the ACRS meeting last time where we kind of didn't really clearly lay out the process that we followed.

So, what we did was we reformatted the enclosure to the paper that we sent up to the Commission to talk some more about the steps that we went through. And this slide goes through those steps that we followed.

The first step, and that's in Chapter 3 of the enclosures, the safety goal screening evaluation,

and this is the step that we took a look at what the safety benefit would be in terms of - what the difference in safety would be from an expedited transfer to the regulatory baseline.

And we compared that to the safety goal policy statement or really the quantitative health objectives, which are a surrogate of the safety goal policy statement, to see whether it would pass the threshold for pursuing or for getting to additional analysis for potential regulatory action.

So, following that evaluation we did the cost-benefit analysis. And really, the point here was that even though the normal process would tell us to stop if it doesn't pass the safety goal screening criteria, we went ahead and did the cost-benefit to provide that information to the Commission for their consideration.

So, it's really additional information for the Commission to consider in their discussions on this issue.

MEMBER RAY: Well, let me just say at that point, as John knows a great deal about seismic analysis, I know a little bit about cost-benefit, and I don't understand how - although I understand your goal was to maximize the benefits as compared to the cost, on

the other hand, some reasonable estimate of the cost needs to be assumed.

And how you did that for an environment that's quite changed with regard to cost such as acquisition of a lot of additional casks, how on earth you could do that is beyond me.

So, if the answer simply is, well, we assumed the existing cost of casks would continue even though we doubled the - tripled, quadrupled the demand for casks, if that's as far as it goes, then say that at the appropriate point.

If on the other hand you made some assumption about how the increased demand would affect cost, make that more clear, because I can't find it.

MR. WITT: Well, we really didn't do a lot of analysis - a new analysis in this study. We tried to grab information from whatever sources we could find.

And for the costs that you're talking about in terms of the casks and that type of thing, most of that came from an EPRI report that was completed just recently on expedited transfer.

MEMBER RAY: And it did assume a higher cost as a result of increased demand?

MR. WITT: No.

MEMBER RAY: Okay. Because that - it may

be incidental to what you're doing, but it is an area that I presume there will be some debate about subsequently. And I just want to get clear as I can whether or not there was any assumption made about how costs would be affected by the change in the rate at which this transfer would have to occur.

CHAIRMAN ARMIJO: Harold, Fred is - I haven't read the EPRI report. Fred, I'm sure, has. Maybe you could when we get to that point, you can just tell us, you know, what assumptions they made and how the, you know, cask cost goes up if you have to buy -

MEMBER RAY: Well, he just gave a very good answer. The kind that we like. The answer was no.

(Laughter.)

CHAIRMAN ARMIJO: I thought he said yes, that EPRI did take that into account.

MR. SCHOFER: No.

MEMBER RAY: No.

CHAIRMAN ARMIJO: He did not, okay. Well, that clears that up.

(Laughter.)

CHAIRMAN ARMIJO: Then they should. Somebody should.

MEMBER RAY: Well, I just - I don't want to get into a debate about it now. I just want to get

clarity around it, because I couldn't figure it out.

CHAIRMAN ARMIJO: Okay. I'm glad that in the rewrite that you put more emphasis on the safety goal screening, because that's the fundamental reason we're here and that's safety.

And I would appreciate if you would expand on that. And once we're satisfied that we've really got that nailed down, then we can go into the regulatory analysis.

MR. WITT: Sure, yeah. Well, that's on the next slide, but, I mean, really when it comes down to it, the safety goal, the chapter that we talked about, the screening, is only a few pages. And the cost-benefit is -

CHAIRMAN ARMIJO: Well, that was a problem. That was a problem. And if the decision basis really is safety, then we needed to expand on that and understand that.

And certainly the public needs to understand that because, you know, it's easy to say, gee, the problem is where they put all these hundreds and hundreds of pages and that's where we should concentrate our concerns.

Whereas the thing that really is - if you have a safety goal and you meet that goal with margin,

then that should be clear to everyone.

And that's not saying that the regulatory analysis is meaningless, but it just says it's supplemental as opposed to the primary basis for the decision.

MR. WITT: Exactly.

CHAIRMAN ARMIJO: Okay.

MEMBER RAY: But we can't tell how the Commission will make a decision.

CHAIRMAN ARMIJO: No, no.

MEMBER RAY: They may rely on the supplemental information and -

CHAIRMAN ARMIJO: They may, or they may not, Harold. But I think that the main thing is that safety goal screening isn't just a given. It was work and it's quantitative rather than purely qualitative. And there's an awful lot of qualitative stuff in the regulatory analysis that concerns me.

MR. WITT: Yeah. And in addition to that cost-benefit analysis we did add in - well, it was in there previously, but there were sensitivity studies done on that analysis.

And some of those factors include the dollar per person-rem conversion factor and consequences beyond 50 miles. So, there's a whole section on those

sensitivities that we can go through.

CHAIRMAN ARMIJO: Yeah.

MEMBER SKILLMAN: Let me ask a question here. It's kind of - it's not seismic like John or finances like Harold.

Have you guys ever handled fuel? Have you ever picked them up, fuel assemblies, put them down, tried to put them in a cask, move the cask around and messed with an upender or moving the racks around to make sure that the trolley and bridge are functioning accurately?

Have you ever done that work?

MR. WITT: I've observed it through inspections.

MEMBER SKILLMAN: Here's why I ask. I recognize Phase 2 is the piece of this work that would draw in that activity, but what I got a feel is absent here is the recognition of what the plant staff needs to do to achieve a different loading pattern and particularly offload to a lighter thermal hydraulic pattern, however you define that. Those activities are not without physical risk, radiological risk.

I'm sure the operators would say, we're macho, we can handle this. And they do a very good job, but occasionally something goes wrong and those risks

can be significant. You can droop a fuel assembly, ruin your handling equipment.

It just seems to me that what's missing in this is, if you will, that practical understanding that this is not free to the industry.

If one were to say, you know, it's a really good idea to go ahead and lighten the footprint of the fuel in addition to the cask issue that Harold appropriately raises, there is a lot of work that these plant operators have to do and it's masked here.

So, I'm just wondering is there a way to embed at least a token flag that says we recognize that this is not free? This is going to cost big time.

And if you've been near those pools, if you've watched that activity, if you've done it yourself, there's a recognition. This is hard work, and it's work that takes a huge amount of safety focus. And it takes an army of people to do it.

It takes your operators, RADCON, security. If you're going to put this stuff in the cask or a truck, you have another vary of security that now comes into play.

It just seems that that piece that recognizes the industry burden isn't fully recognized. And if we say, well, we'll just do that in Phase 2,

we might not serve the Commissioners best by advising them, hey, this is not free.

The people that run these plants are really going to be put to the test if we go this way. So, I'd ask you to consider that.

MEMBER SCHULTZ: And, Dick, I know that you're considering this in that comment, but what we are asking for expedited transfer, we would be asking staff at the plants to be performing this task over a concentrated period of time, but that concentrated period of time is a long time.

In other words, this is not happening overnight. It's going to be happening over --

MEMBER SKILLMAN: Five years.

MEMBER SCHULTZ: A few years. Five year's assumption. This is diverting the attention of the operations, the maintenance, the engineering staff of the plant away from other things that they would normally be doing.

That's also an impact on plant safety and it can be evaluated directly with processes that we have for looking at the way plants operate and the way diversion of activity to a project like this could affect overall plant safety.

So, I know there's an argument that says,

well, we're going to have to do this anyway over time. You're going to have to unload the fuel, the casks or to ultimate storage over time.

We're not asking you to do anything differently. It's just unloading the pool, but doing it in a concentrated fashion over a three to five-year period is going to divert that attention and it will have an impact on plant safety. It can be quantified.

MR. WITT: Yeah, another thing that I would add in, too, and I was just looking to see where we talk about this, I'm not sure if - we did indeed include a discussion about the additional risks associated with the movement - or more movement of the fuel in the spent fuel pools.

Another thing we added in there was the uncertainty of the final disposal of these canisters, these casks. There's really no guarantee that if the licensees put the fuel into these casks at this point, that they won't have to repackage them at a later date.

MEMBER SKILLMAN: Again, yeah.

MR. WITT: And so, that's another factor that I think the Commission has to consider on this issue is do you want licensees to start doing this right now when they may have - when a final disposal strategy hasn't been set yet?

And so, we do have a little bit of a discussion. The only problem was that you can't really quantify this stuff. We didn't have the information available to add in those risks associated with the expedited movement.

So, I think if we were to do more work, if this did show indeed that there may be a potential benefit to doing this, then we would look at those additional risks and uncertainty associated with the final disposal.

MEMBER SKILLMAN: My point is that the burden is placed on the operators and may not be fully appreciated unless it's flagged so the Commissioners say, hey, this is not free. If we move in this direction, we're really relying heavy on the people that operate these plants.

And like Dr. Schultz says, it's a diversion of other - of resources to what could be a very slim increase in safety, very huge risk in moving all of this equipment, because it's complicated. Thank you.

MR. SCHOFER: This is Fred Schofer.

With regard to your comments, there was a recognition that having these huge loading campaigns would be a diversion and is complicated and does take a lot of focus.

That was one of the reasons that in the paper it was a five-year campaign to achieve that lower density configuration recognizing that you can't do it much faster than that.

With regard to, you know, there is no - I would say there may not be, you know, sufficient other consideration-type comments that would qualitatively indicate, you know, the points that you're making, but they were thought about when I redid the analysis.

CHAIRMAN ARMIJO: Well, you know, I would expect a lot of these points should have been brought up by the industry comments. And if they weren't brought up, shame on them for missing the opportunity, because they're the ones who know very well what they would have to do.

And we, you know, our members have a lot of experience as well and we're pointing out some of the things we thought about. But as far as a systematic compilation of all the qualitative as well as quantitative concerns, should be put in some slides or package or something so they aren't just buried here and there throughout the report, because it's a non-trivial exercise.

I've worried, you know, I'm just - don't want to spend too much time on this. How many casks

would you need if you went through such an operation on an annual basis, and is there a capacity with licensed US casks to meet that need?

Would you have to be licensing casks from, let's say, France or Germany or Japan or Korea? And that would add cost and that would add uncertainty in order to meet our arbitrary goal.

You said five years. Maybe it would turn out to be ten years. Who knows? But somewhere in there has got to be some little package that says, okay, here's the alternative. It isn't perfect either. It's got some real problems and it better have some really big benefits before we enter into this exercise.

And I know you've got it throughout your report, but I just have a hard time getting it all put together.

MR. WITT: Okay. I'll go ahead and turn it over to Steve now to talk about the safety goal screening.

MR. JONES: Good morning, I'm Steve Jones in the Office of Nuclear Reactor Regulation, Division of Safety Systems. I'd just like to go over the safety goal screening, and also the regulatory analysis.

To start with the safety goal screening, we looked at the highest frequency derived from all the

different plant groupings and considering, in this case, the highest estimate for the frequency which turned out to be 3.46 times ten to the minus five per year. And that considers, really, the Sequoyah site seismic hazard curve and all the other contributing events.

Then we relied largely on the spent fuel pool study which evaluated several bins of releases consisting of cesium and short-term isotopes such as iodine.

For the large release - excuse me - that study determined a condition of probability of 4.4 times 10 to the minus four per release of a latent cancer fatality risk to an individual within ten miles of the plant site.

That number was relatively insensitive to the magnitude of the release, however. So, because the linear no-threshold model was used and protective actions were assumed to be implemented. So, any release that caused the type of actions to be implemented would, you know, result in people being relocated and, therefore, avoid additional dose.

Okay. With those considerations, determined a calculated latent cancer fatality risk of one in 66 million per year. And that's less than one percent of the individual risk goal, which is based on

a routine probability of two cases per thousand people per year. And then taking one-tenth of one percent of that gives you two in a million per year.

MEMBER RYAN: Steve, are you going to do any comparisons to actual cancer incidents? The average latent cancer in the US is one in four to one in three. So, it's striking against one in 66 million.

MR. JONES: Well -

MEMBER RYAN: It's an extremely low cancer rate for -

MR. JONES: Right.

MEMBER RYAN: - an activity compared to the background cancer rate. So, I don't know how you make any sense out of that.

MR. JONES: Well, this is just not cancer, but cancer progressing to a fatality within one year. And then we're comparing it against the two in a million or - which I guess in the same terms would be one in 500,000 per year is the goal. And that's one-tenth of one percent of -

MEMBER RYAN: I guess what I'm trying to address is that if you have very low cancer incidents from something related to the activities we're talking about, it would be impossible to distinguish that as being caused by the activity of being a normal cancer

in the population from all other causes.

MR. JONES: That's true.

MEMBER RYAN: So, I'm kind of wondering what we do with this. Do we interpret it to make judgments and decisions on right and wrong?

MR. JONES: I guess we use this predominantly as a screening. Right now the regulatory analysis guidelines are more formatted to address reactor accidents and focus on core damage frequency and large early release frequencies.

MEMBER RYAN: Right.

MR. JONES: This, because it's a spent fuel pool, the release is a different character, different isotopes.

MEMBER RYAN: It relies not broadly on the wind and all that.

MR. JONES: Well, there's certainly a potential for it to go over long distances and affect large areas, but it does not have the same risk of immediate health effects on a population.

So, we're looking basically in a sense of magnitude. If it was like ten percent of our goal, that would definitely lead us to look closely at a cost-benefit analysis.

When we're far less than one percent, it's

hard to make an argument that any action will be substantially impact the health and safety of the public.

MEMBER RYAN: To stress that kind of, you know, decision-making as what you're driving at as opposed to managing, you know, the risk of cancer from some exposure, because you're using that as a metric to something else.

MR. JONES: Right.

MEMBER RYAN: I'd put that in bold letters somewhere so it doesn't get confused with the other kinds of uses of that sort of parameter.

Does that make sense to you?

MR. JONES: I understand.

MEMBER RYAN: I can just see an awful lot of confusion in trying to explain this versus that kind of discussions with lots of different constituencies. It might be hard to get it across. So, it's probably best to try and get it explained right up front.

MR. JONES: Right. Okay.

CHAIRMAN ARMIJO: Kathy.

MS. GIBSON: Kathy Gibson. This is on research. I just wanted to remind you that in the spent fuel study we did look at some thresholds in addition to the linear no-threshold. And one of those was

subtracting out the background radiation.

And of course any of the thresholds that you use other than linear no-threshold just makes your numbers lower. So, that is in the spent fuel study if you wanted to see the difference that it makes if you take the background into consideration.

MEMBER RYAN: No, I appreciate that, but I think it very quickly dwarfs - background dwarfs any of the 9:21:43 probability low-dose events.

So, there are different ways to handle it, treat it, discus it. And I just think we ought to think about the audience looking at different kinds of risks from, you know, radiation exposures like releases, like, you know, spent fuel accidents and make sure that we don't confuse it more than we do help explain it.

CHAIRMAN ARMIJO: Well, most of this, the dose that you're talking about, is from people returning to a contaminated -

MEMBER RYAN: That's correct.

CHAIRMAN ARMIJO: - property.

MEMBER RYAN: Right.

CHAIRMAN ARMIJO: And so, yet, they can't return unless they meet the habitability criteria.

MEMBER RYAN: Correct.

CHAIRMAN ARMIJO: Which from my point of

view, is effectively a threshold that says as far as the NRC and EPA or whoever else controls this, they have set a threshold, come back and live here indefinitely.

Yet, then we turn that okay situation into a cancer risk calculation, which I would say, you know, doesn't make a lot of sense.

If you think there's a real cancer risk and you believe those numbers, you'd never - you might say don't come back.

MEMBER RYAN: I think -

CHAIRMAN ARMIJO: Yeah, I think there is a threshold built into the habitability criteria that is not recognized in these calculations.

MEMBER RYAN: But the cancer risks that the US population faces, a broad scope, to me, the way to address it is to put that risk in context with other risks, which are people are immune to any kind of consideration that those risks are unacceptable such as smoking.

CHAIRMAN ARMIJO: Well, I think we're in agreement. I just have, you know, there's a lot of - lot of concern when anybody challenges LNT as being meaningful at very low doses.

And yet, regulators set habitability criteria that, in fact, recognize there are safe levels

of radiation over and above background. Tiny, tiny amounts are okay.

And yet, we don't - we still calculate latent cancer fatalities based on doses that we allow people to take.

Okay. That's confusing to me and it just seems like - and it confuses the public, I'd like to tell you. They say, well, is it safe, or isn't it safe?

You guys are saying there's this much cancer fatality risk if we come back. Gee, that's terrible. Why - so, you've got a communication problem.

MEMBER RYAN: You know, years ago I remember a paper. I guess it was Bernie Cohen that wrote the Catalog of Risks. Something along those lines. That wasn't a bad attempt at the kind of structural, you know, how risk plays out that might be helpful, but I think that's what I'm struggling to understand.

And I think, Sam, that's kind of the same thing you're looking at.

CHAIRMAN ARMIJO: Yeah.

MEMBER RYAN: How do you take one risk in one situation and compare it to another risk in a completely different situation? It's tough.

MEMBER STETKAR: You may want to be careful. Watch 1400 tried to do this comparison of imposed risk,

if you will, with routinely accepted risk and daily - and they were criticized wildly for that comparison, you know.

So, the notion of comparing the cancer risk from a fuel pool accident to the normal incidence of cancer in the American population, one to three, one to four or whatever it is, has not gone over very well.

On the other hand, the types of arguments that Dr. Armijo is making, which is strictly limited to this particular issue, repopulating an area under acceptance criteria that are imposed for repopulating for this, is, I think, a very useful type of discussion. Because that divorces it from, you know, is it one in three, one in four from all sources, you know, automobile accidents and all that kind of stuff in terms of plant fatalities.

And it really does focus on this notion of what is a regulatory, whether it's state, federal, acceptable level of risk from inhabiting - permanently reinhabiting that area.

MS. GIBSON: Well, we have to be a little careful, because it's actually the individual states and local governments that make the decision on the return criteria. And it's different from state to state.

MEMBER STETKAR: Obviously there's uncertainty.

MS. GIBSON: It's tough to get out and say below this, you're safe, and above that it's -

MEMBER RYAN: And that compounds the problem trying to explain it. Well, why across the state line is it higher or lower?

MS. GIBSON: Which is why the linear no-threshold serves our regulatory purpose, because it's conservative.

CHAIRMAN ARMIJO: But it frightens the hell out of people, I'll tell you. I talk to lots of groups of people and there is a belief as long as the NRC says, that no level of radiation exposure is safe no matter how small.

By using the LNT, you voice that thought in the mind of people and, in fact, it is not correct, you know.

There is a safe level of radiation. What it is, people can argue about, but, you know, there's no such thing that, you know, so, there is a real problem here and we keep telling people it's safe, but it's not safe by the rules we use.

And we're going to be arguing about this forever if we don't - if somebody doesn't step up to

the bar and says, hey, there's a level from a regulatory point of view that it is safe to come back, here's the numbers. At this level, we believe there's plenty of margin there. And other states may have for political reasons different thresholds, but there is a threshold.

And somewhere along the line that - because, you know, otherwise you're left with this thing saying, you know, we're letting you come back to an unsafe region and - but it's safe, or it's not safe, you know. It's very confusing.

MR. RECKLEY: This is Bill Reckley with NRR.

And just to acknowledge that that might be a policy issue, but giving it back to what we were tasked to do in this particular thing, you know, we're really asking the Commission to make a decision if this issue warrants additional study and we'll do research to do investigations of added costs, added risks. And if we were tasked to do, incorporate other policy issues within that like LNT, but the bottom line when you look through what we've done to date would be with those conservatives in place using LNT without revisiting the conservatisms in here and in that, ignoring the risks of the transfer, ignoring any additional cost.

The staff's conclusion is, we don't need to study this anymore. If we were to do more studies,

spend more time, more FTE, more dollars to get better data, it is only going to reach the same conclusion, in our view, that in the end we would be saying we don't need a rule to require expedited transfer of spent fuel.

So, we acknowledge all of these discussions, but we were really tasked to ask the Commission to make a simple decision, A or B. And we only went as far as we thought we needed to go in order to support that decision.

And as Steve's going to get into as he starts going through the assumptions, we made some conservative, we made some out of convenience, but the bottom line is in the end, in total, they're going to support the recommendation we made to the Commission.

CHAIRMAN ARMIJO: Well, you know, we're all engineers and we understand that, Bill. And we understand these charts and we can interpret them in a way that among ourselves we understand them. But, you know, there's also the general public out there who doesn't understand this thinking and the number can really be misused.

So, even though your conclusion may be right, the degree of conservatism that is in this analysis, I see it, and maybe other people see it, but I'm not sure that the general public sees it. So,

that's really our - part of our point.

The other thing is, and we'll get later in this when we get into the assumptions, some of the assumptions are trying to maximize a benefit of the alternative, our really extremes. And we want - at least a couple of us want to challenge the models. So, let's move on.

MR. JONES: Okay. This slide goes over the safety goal screening results. I've talked a lot about this already, but no risk of fatalities due to the nature of release. And the potential benefit is a very small fraction of the latent cancer fatality goal.

Also, the risk was in, like I said, insensitive to the magnitude of release. Events in the spent fuel pool evolve relatively slowly and protective actions would be effective.

We decided to proceed to the cost-benefit analysis even though the process allows us to stop here due to the margin from the quantitative health objectives.

And next slide, please. Okay. Just real quickly we talked about the cost-benefit analysis before, but the other thing I have just one alternative-expedited transfer. And we wrote looking at that, basically to provide a maximum measure of the

benefit that could result from this action and transferring - the alternative would involve transfer of fuel with more than five years decay to dry casks and store the remaining fuel in a low-density configuration in existing racks. That would be the hottest assemblies would be surrounded by four empty slots on each face.

And then the baseline would be having fuel, hot fuel surrounded by four colder assemblies on each face.

The analysis is conducted for four groups although seven groups were initially, you know, determined. Seven groups based on the risk. Three groups were not evaluated. And the four groups representing the operating plants and one group for new plants were evaluated.

Major assumptions, we separated it out in a new table in Regulatory Analysis Table 2. And it discusses, I believe, the assumptions and basis for those assumptions.

The initiating event frequencies and accident progressions is one section of that. And then economic modeling, the costs and the benefits of reverted dose, and also the timing of the cask transfer or fuel transfer to dry casks.

MEMBER REMPE: I was afraid you were going to leave the slide. On Table 2, I really liked it because it does lay out the assumptions. But when I started going from the assumptions listed in Table 2, for example, like the liner fragility and then I started looking other places in the report like Table 39, I found it inconsistent.

Was that a typo, for example? Because when I look at that like Group 1 -

MEMBER STETKAR: Go to your backup slide number - Page 35 in the backup slides. That will highlight - I'm sorry, 34.

MEMBER REMPE: Yeah.

MEMBER STETKAR: Those liner fragilities are not what we used in the study.

MEMBER REMPE: Please say that, because it's not listed in -

MEMBER STETKAR: I understand that. They are not what they -

(Simultaneous speaking.)

MEMBER REMPE: And so, yeah, that was one thing. And then this factor of 19 and when I compared it, you only invoked it, I guess, for the low-density cases and not the high-density.

MR. JONES: That's right.

MEMBER REMPE: So, there are a lot of assumptions you made that either might be inconsistent with things later in the report or the logic for invoking those assumptions didn't seem very clear to me. And I don't know when the best time to discuss this is.

MEMBER STETKAR: I think this is our only shot at it.

(Simultaneous speaking.)

CHAIRMAN ARMIJO: Well, you know, I think it would be very good to go through the assumptions by table.

MEMBER REMPE: Uh-huh.

CHAIRMAN ARMIJO: Because, you know, I think they are really important. And I know I had a lot of questions that I was going to raise as we went along, but it might be useful for the staff to go through the assumptions one by one and give us the opportunity to raise our concerns in one shot rather than -

MR. JONES: Okay.

CHAIRMAN ARMIJO: And I'll look up all my comments and - but just go ahead, Steve, and we'll just -

MR. JONES: I think on Slide 15 we'll get to the - well, I'd like to progress through them, I guess, until we get there.

CHAIRMAN ARMIJO: When we get there.

MR. JONES: We did establish a base case and perform sensitivity studies around that. Then Slide 11. Okay. Thank you.

Okay. So, what we did at first to establish the maximum benefit is really look at how we could separate the low-density and high-density cases. And that centered really on the release fractions we assumed which came, to a large extent, from the spent fuel pool study and the previous studies and the effectiveness of mitigation.

And there are some issues, really, frankly, with the implementation of mitigation and the uncertainty that's involved in determining that likelihood.

So, for the regulatory baseline we used high cesium release fractions for this. For the BWRs for the elevated pools, we relied on the spent fuel pool study which had values of approximately 40 percent for those releases in the high-density cases on mitigated.

And then for the remainder or the balance of the plants where the pool is at-grade and we're less certain of leak locations and things like that, we used the value from NUREG-1738, a 75 percent release fraction, and assumed ineffective mitigation.

That means that once the fuel heats up, we're assuming that it progresses to a large release. And that basically results in what we considered a conservative outcome for the high-density case.

If we assumed full effective mitigation in this case, you would result with a very small delta between the two events, because essentially all cases would be mitigated. There would be no release.

CHAIRMAN ARMIJO: But cesium isn't that really, you know, this came up, you know, Bill Shack isn't here, but he's our consultant on this thing. He couldn't attend the meeting or be on the bridge line, but he did send me his notes.

And the issue of assigning effective mitigation only to the alternative and not to the base case is, you know, his words were just plain wrong. It's not conservative.

And his argument was that as for the pumps whether it has a light loading or heavy loading in the pool, the pumps still work.

Access to the ability to - to the equipment to cool and measure and things like that is not affected by the loading particularly with the new equipment that the orders have imposed.

So, you know, it's either both of them have

effective mitigation, or both of them don't have effective mitigation to present really a fair picture of the benefits.

So, you know, either we're both - so, that's, I think, a major, major thing. You can't just - in trying to not undervalue the Alternative 2, you can go overboard by giving it so many advantages and depriving the base case advantage that are really there that you just - you wind up creating a false impression that the Alternative 2 is such a good thing at least in some of the cases you analyze.

And then you say, well, that being the - despite that, we don't think it's a good idea. So, you know, somewhere along the line you've got to bring it into - a little bit into balance especially in the high cases and in the sensitivity studies.

If you look at the sensitivity studies and the high cases, it seems like a slam dunk. You ought to go and expedite fuel transfer. And yet, and I know that's not what you believe is the right thing to do, but somewhere along the line - I won't use the words "painted yourself into a corner," but something like that has happened that you've got a very difficult explanation to make of how can you calculate these very large benefits granted for sensitivity studies, but

still then say we don't want to do it. We don't think it's worth pursuing.

So, that's kind of the heart of many of my concerns. And this mitigation has got to be treated a little equitably between the two cases. I just don't think there's any justification, and I share Bill's views on that, that there's any justification for having just one alternative get the effective mitigation and not the other.

MEMBER SCHULTZ: So, you're nodding your heads as Sam went through his discussion related to mitigation.

I just wanted to get on the record were there engineering or analysis or operational rationale that were identified that would have differentiated the alternatives with regard to mitigation?

Because all I saw in the documentation both now and what we have seen over the last several months is that it, in fact, is an assumption in order to maximize the benefit of going to the alternative of low-density loading.

MR. JONES: Right.

MEMBER SCHULTZ: And I just want to emphasize what Sam has said. To put that into a document and say we are not going to credit mitigation for a case, for

the current case, but we will credit mitigation for the case where we have low loading, can be read by anyone to say that we believe that mitigation is not possible for the current case, but it is possible for a low-density loading, which is not the intent.

The intent is to say we have some uncertainties in the evaluation where we have - we want to credit the case of low-density loading in a fashion to maximize the benefit. And, therefore, we're going to incorporate a factor of 20 and see what happens.

But to attach it to an engineering rationale that says, okay, say there is no mitigation possible, for Case B there is large mitigation possible, it presents the wrong impression, the wrong rationale, the wrong reason for the difference.

And I think the same is somewhat true, at least, for the assumptions that were used with regard to the cesium release fractions.

Because you use - we'll use the high one for this, we'll use the low one for this and there was a difference, but one does not - one cannot attach that, really, to high-density loading and low-density loading in such a direct way as was done here.

There's also the rationale that we're doing it because we want to maximize the difference that we

will see. And when we're all said and done, we will see that we still don't justify doing further study.

That's all well and good, but to present it as an engineering rationale that Case 1 is different than Case 2 in a real practical engineering and analysis way, presents the wrong information to scientists, as well as the public.

Some scientists are in the public. I don't -

(Laughter.)

MEMBER STETKAR: I'm just saying that the reader of the document can be misled.

MR. JONES: I guess I'd have to say that the mitigation is really turning out to be somewhat more of a distraction than a help, because really the dominant impact is the release fractions. That the assumptions that go into driving those using the highest case from the spent fuel pool study and using 75 percent from NUREG-1738 give you, you know, 30 to 50 times more, I guess, consequences, greater consequences from the baseline or high-density case than from the low-density case.

The additional factor of including mitigation is relatively small. It's just -

CHAIRMAN ARMIJO: Are you saying, Steve,

that even if you'd applied effective mitigation to the high-density case, the release fraction would not have been -

MR. JONES: Well, that would go the other way. That would just result in basically no difference or -

(Simultaneous speaking.)

CHAIRMAN ARMIJO: I think that should be made clear that when you have effective mitigation in either case, there's not much going on.

If you have ineffective mitigation in both cases, low-density has an advantage.

MEMBER STETKAR: Let me try something. And this follows up on a little bit of what Steve was saying.

We tend to talk about effective and ineffective mitigation. And for whatever reason, effective mitigation for this particular study is assigned a 95 percent chance of being perfectly good, and a five percent change of being perfectly bad.

CHAIRMAN ARMIJO: Okay.

MEMBER STETKAR: And I'm not going to argue about 95 and five percents. What I heard Steve asking and what I think would be very useful rather than saying, well, suppose we assume 95 percent effectiveness for the high-density loading case, you know, you're saying,

well, if we did that, there wouldn't be any difference. At least that's what I'm hearing you say.

Is there anything, because you guys have studied this a lot and understand it a lot better than I did, that would say, well, there's a rationale to say that we believe that the effectiveness for the high-density case might be less than 95 percent, may be 80 percent because the timing is a lot faster, because I don't know, you know?

Don't focus on pumps, because the pump doesn't care. The hose doesn't care. People do care, you know, and that's this whole notion.

Is there - if there's no engineering rationale to say that we don't believe high-density loading versus low-density loading would result in a difference, I don't care whether it's 95 percent effectiveness or 50 percent effectiveness, if there's no rationale to say that there would be any difference, then it ought not to be included as a variable parameter.

If there is a rationale to say that there would be a difference, that rationale ought to be presented and perhaps you ought to take a shot at what the difference might be.

CHAIRMAN ARMIJO: Okay.

MEMBER REMPE: Also -

MEMBER STETKAR: And if it's true that the mitigation doesn't make a difference to the overall conclusion, why are we having this discussion, you know? Why is it given the prominence in the report that it is getting?

CHAIRMAN ARMIJO: Well, I think it's given prominence, because that's exactly what we want to do. We want to mitigate.

MR. JONES: I think at the end of the cesium case, we do rely on that somewhat for defense-in-depth purposes, but we're not using it for the - to evaluate whether we need to refine the cost-benefit analysis, I guess, is the point.

CHAIRMAN ARMIJO: You know, I think - I just want to read what Bill sent me, and I think he sent copies to all the members, and relate it to the mitigation.

And it basically says it's technically indefensible to just assign zero to one and a hundred percent to the other.

So, his arguments are, you know, the pumps either survive the event and are in place and operate, or they don't. None of this is affected by loading density in the spent fuel pool.

There may be, John, small differences in time available, but the overall accident sequence is

long enough that this would have little affect.

And so, you know, basically you're saying, well, if we want to compare the two alternatives, let's do it apples and apples. Both of them get full mitigation whether it's 95 percent or 50 percent, but they both get the same, or they both get zero. And then you can just compare them, but you can't just say, well, we'll cripple this guy, and this guy who's not even wounded, we'll give him help, you know. Something is wrong here.

So, I guess maybe we've beat that to death.

MEMBER REMPE: Before you leave this slide, though, on the cesium release fractions just is it because you - where there's more certainty is why we used higher values for Groups 2 through whatever, but is there really a physical reason to say that we think that Groups 2 through 4 have a higher release?

What is the physical reason? Is it because you just don't have a MELCOR analysis you're not spending a lot of time on it or -

MR. JONES: Well, it's predominantly because we're talking about largely PWR fuel and it has higher -

MEMBER REMPE: Mark III though.

MR. JONES: Yeah, that's true. The Mark III

BWRs are in there.

MEMBER REMPE: Uh-huh.

MR. JONES: They still have a little bit higher power density, The PWRs have a lot higher power density for the fuel.

MEMBER REMPE: Right.

MR. JONES: The past studies like, for example, NUREG-1353 assumed a factor of four difference between the probability of reaching a high enough temperature in BWR fuel versus PWR fuel to ignite and have a large release.

MEMBER REMPE: So, some of that logic it would be helpful if it were included. You don't even have NUREG-1738, I think, included in Table 2 in the comments.

And the factor of 19 even is - you've got to dig around in that table and it just seems like this document if it's standing alone, would be helpful if you put a little bit more beef and why you make certain assumptions.

MEMBER SCHULTZ: So, I mean, the logic that you just described, though, is it - in the first case you have that the value of 75 percent was used for other groups in the base case. And with the low-density loading, the assumption has been three percent for all

groups in the base case.

So, it seems as if there are conservative assumptions or biases, I would call them, associated with the assumptions here as well.

In other words, the title of the slide is completely right. They are assumptions to maximize the calculated benefit. But, again, I'm concerned that they also lead into conclusions - or could lead to conclusions that there, in fact, is a real difference between having a low-density loading and a high-density loading. It is -

MR. JONES: We did have some problems there resolving the release fraction for the low-density case, because there is, I mean, the spent fuel pool studies the first time that that's where we've been examining in detail. So, we only have that three percent data for BWR fuel. We don't have information like how a PWR assembly might perform in the similar low-density configurations.

So, and the previous studies are no help at all, really, with respect to that.

MEMBER SCHULTZ: Right. But, again, what has been done is to maximize the difference between the two cases.

MR. JONES: Right.

MEMBER SCHULTZ: And there are rationales going into the documentation both previous and here that we're trying to account for uncertainties and we're trying to maximize the difference.

But, again, my concern is that the conclusion will be that there is a real difference between this and, therefore, why are we saying we don't need to do this?

It can cause confusion where it seems as if it's based upon real scientific evaluation and analysis, and it's really a result of some assumptions to see whether we should go forward.

MR. ESMAILI: This is Hossein Esmaili.

There is some rationale to what for the low-density cases we assumed lower release fractions. It's the insight we got from the SFPS.

And in the low-density cases, generally we didn't see any hydrogen combustion. So, the building remained intact. And the same thing can be applied, you know, sort of to the PWR that, you know, if you have low-density cases, you are not going to produce a lot of hydrogen, you probably are going to maintain.

And if you remember in the NUREG-1353, they assumed a range of release fractions going from 10 percent to a hundred percent. At hundred percent meant

that it was a high-density case, you know, you have a large release to the environment.

A ten percent case was for the case where the building remained, you know, you have a decontamination factor of ten. That means that the building was retaining some of the fission product.

So, we are using that, you know, like about ten percent for the case where it's a low-density case that, you know, the building remains intact. So, even though you get releases from the fuel, it's not all going up.

These rationale have happened, you know, kind of explained in the report, but, you know, there is a rationale behind, you know, why we - and even the low-density cases we don't see a large variation.

And if you remember from the SFPS when we do high-density cases, you have large variations. You can have, you know, a few percentage going all the way to 60, 70 percent.

We didn't see these in the low-density cases. So, we are a little bit more comfortable with the type of releases that we are getting from the low-density situation.

So, it is consistent with past studies and, you know, insight.

MEMBER SCHULTZ: I understand that, but it also, I mean, if you look at the numbers here, what has been selected in order to maximize the calculated benefit for the case of current spent fuel pool loading, or high-density loading, some are not - some reactors are not - fuel pools are not at a higher density loading, but for those - for this study, the assumption is we're going to use to maximize the benefit, the upper range of what has been calculated in the past for cesium release fractions. And then we're going to use the better values that we have calculated for the low-density case.

And I just think it can be taken in a wrong fashion if one is reading this study to try to maximize the difference and interpret it that it's a result of, if you will, equivalent engineering analysis and evaluation where, in fact, we are trying to maximize to calculate that. It just needs to be presented very clearly.

CHAIRMAN ARMIJO: Okay. Let's move on.

MR. JONES: Okay. For the base case analysis, this discusses some of the assumptions we have. And we considered the base cases appropriate for the decision whether to conduct additional studies to refine these numbers, or in some cases that would

involve, for example, studies on - in the scope of the spent fuel pool study for other plant types to get a better understanding of, for example, the structural integrity of the pools or to refine the thermal hydraulic response of PWR versus BWR assembly, things like that.

But to run through these real quick, the initiating events we have used the USGS 2008 information for the seismic hazard curves.

They're not refined. I guess the completion of Generic Issue 199 will probably result in better information for seismic hazard for the central and eastern plants.

But we used the Peach Bottom site which was selected, because that is among the highest seismic hazard sites among the central and eastern US sites.

And then for other initiators such as station blackout or conditions that lead to a partial loss of cooling and then boiling of the pool, we've used initiating frequencies from NUREG-1738 and NUREG-1353.

MEMBER STETKAR: Steve, be really careful about your use of terminology and your sweeping statements in this report.

The precision of the sum total of all other initiating event frequencies that can - I've forgotten the words and I won't take the time to look them up -

threaten or disrupt - I think the word is "disrupt" fuel pool cooling of 2.37 times 10 to the minus seven event per year, that's a very precise and very tiny number.

If I look at the frequency for a two-train plant of a complete station blackout, meaning loss of offsite power and destructive failure of any emergency power supply, it is considerably higher than 2.37 times 10 to the minus seven.

So, it's pretty doggone clear to me that that 2.37 times 10 to the minus seven is neither an initiating event frequency -

MR. JONES: Right.

MEMBER STETKAR: - and it is certainly not the cumulative initiating frequency of all initiators that can disrupt spent fuel pool cooling.

MR. JONES: Right.

MEMBER STETKAR: It must include some other assumptions and failures, et cetera. So, it's some surrogate for a large number of other event sequences that you feel have been adequately quantified by some other studies, I think.

MR. JONES: That's correct.

MEMBER STETKAR: Okay.

MR. JONES: It does -

MEMBER STETKAR: It's not an initiating

event frequency.

MR. JONES: Right.

MEMBER STETKAR: Okay.

CHAIRMAN ARMIJO: Steve, I have to go back to release fractions again. I want to make sure I understand it.

In your assumptions, you've summarized that for spent fuel pool Groups 1, 2, 3 and 4 low-density loading release fractions are 0.5 percent for the low estimate case, three percent for the base case, and five percent for the high estimate. And that is based on a calculation that - or is it based on the assumption that it is 95 percent mitigated or not?

So, this says if even unmitigated, release fractions for the low-density case would be this low.

MR. JONES: That's correct.

CHAIRMAN ARMIJO: Okay. I just wanted to -

MR. JONES: And the same for part of the detail about that, but the building integrity plays a large role in the assumed release fraction.

MR. ESMAILI: Yeah, this is Hossein Esmaili again.

That's what I was saying before that for the low-density cases even unmitigated you have very,

very - even though you have some releases, but because the - this is we are talking about environmental. Because the building for the most part remains intact, you have no releases to the environment.

CHAIRMAN ARMIJO: Right.

MR. ESMAILI: The cases in the high-density case that led to, you know, very high releases was because you had large releases, you have hydrogen explosions, you have - you brought in - if you remember, you brought in air and you lost the building. So, those were the cases that led to about 40 percent, 50 percent releases.

CHAIRMAN ARMIJO: But if you had allowed mitigation to be effective at some level, not zero, that 40 percent would be lower because the probability of getting to a hydrogen --

MR. ESMAILI: Right.

CHAIRMAN ARMIJO: - situation and big fire would be much lower. And the question is, you know, and that isn't even shown in the analysis, right?

You don't show the effect of what mitigation would do. You do in the - probably in the pool study, but you don't here for the high-density case.

MR. JONES: That's correct. We don't have the details to cover the variety of plants.

CHAIRMAN ARMIJO: So, the low-density case, the big advantage of that is it doesn't require effective mitigation. That's a legitimate conclusion that I think you can draw to have relatively low release fractions.

MR. JONES: Well, one of the big take-aways from the spent fuel pool study is that the frequency of release is essentially identical between low-density and high-density cases.

The hot assemblies are - require essentially the same amount of cooling. Very nearly the same amount of cooling. And they will proceed to an oxidation state with - at about the same frequency under the same conditions. So, and mitigation for the same reason, mitigation would be essentially equally effective.

Any time mitigation would be deployed if it's effective at deploying spray when spray is required, then you would have no release for the majority of the cases. That's the 19 out of 20.

CHAIRMAN ARMIJO: Okay. I just wanted to make sure I understood that. Thank you.

MEMBER REMPE: Before you leave Slide 12, a while ago we had a meeting and Dick brought up about the fact that you were assuming Peach Bottom weather

for all of these cases, right?

MR. JONES: That's right.

MEMBER REMPE: And there was a question raised about, well, what would happen if you - I know it's too much work, but is there some weather for particular plants where you actually can see some difference if you went site-specific?

And I thought the answer we got from the staff at that time is, we'll get back to you on it. And I don't think we brought it up when we had the last full committee meeting and have you looked at that at all?

MR. JONES: Well, it does affect like what populations - particularly when you're looking within 50 miles, what population groups might be affected, but there is a sensitivity that addresses changes in population density and the effects. And we'll talk about that a little bit later.

When you go beyond 50 miles it's really not so much of an affect, because eventually, you know, you generally will get to a population center that will be impacted by that weather.

MEMBER REMPE: Okay.

MR. JONES: You're looking at a long-term release. So, there's a lot of wind shifts and things

that are -

MEMBER SKILLMAN: Well, clearly you used the wind rose for Peach.

MR. JONES: Correct.

MEMBER SKILLMAN: And you had a prevailing northwesterly that pushes all the isotopes down into Baltimore.

If you take the wind rose at Cooper, you probably would have said, golly, there are a lot of cattle that are affected, but not many people.

So, the plant and the wind rose are important to the conclusion particularly for the downstream effect for a major event.

And so, to hang the conclusion on one plant without wind rose really does maximize the benefit.

And as my colleagues have pointed out, you need to be careful how to interpret that, because that wind rose in that particular plant gives a stunning benefit particularly at the increase to the alternative, the dollars per man-rem.

So, there's a need just to toggle some of these issues that communicates caution. One can't be too accepting of the conclusions without understanding what they really mean. Thank you.

MEMBER STETKAR: Steve, you also used - and

I don't know whether we raised this question earlier, because I wasn't at one of the Subcommittee meetings, but you also use the evacuation plans and evacuation time estimates for Peach Bottom in your MACCS 2, right?

How would those be - how important are those to the overall results? Let me ask you that. What I'm concerned about is, I know the spent fuel pool scoping study looked a lot - or SOARCA or somebody looked at - everybody has looked at Peach Bottom. Everybody has looked at bridges. Everybody has looked at roadways. Everybody has looked at pathways and things like that for that particular site.

We're talking about a really big seismic event here and we're curious about how representative the Peach Bottom evacuation time estimates and evacuation plan is for the infrastructure surrounding all the other sites in the country under this type of very severe seismic event.

I know some sites, for example, that have only two directions that you can leave, and one direction might be throwing you over a bridge, for example.

But I don't have a sense - I didn't run MACCS. So, I don't have a sense - and I don't have a sense of the timing here of how important that might be, but it definitely correlates with some of the other

comments that we've had on meteorology.

MR. JONES: Certainly I think the evacuation assumptions would be more important if we were looking at a reactor event like SOARCA was.

MEMBER STETKAR: Uh-huh.

MR. JONES: For the spent fuel pool study, you do have a very long period of time -

MEMBER STETKAR: You're also looking at a doggone big earthquake and something fell down. You're not going to, you know, the Corps of Engineers isn't going to come in and build a pontoon bridge in, you know, a couple of days.

MR. JONES: That is another issue that we would have to refine in more detail to proceed with, you know, the next step analysis of this event.

But we thought that given the long time for this scenario that using the Peach Bottom information as readily available and thoroughly researched would be a good approach to this screening.

MEMBER STETKAR: Well, but "long time" is long compared to a power reactor core damage event, but it's not long in terms of, you know, calendar time.

MR. JONES: Right. We're talking on the order of one to two days.

MEMBER STETKAR: A couple of - yeah, a couple

of days. That's why I used a couple of days for building a pontoon bridge. It's not like months.

MR. JONES: Right.

MEMBER STETKAR: Okay.

MR. ESMAILI: This is Hossein Esmaili again.

I'm not a MACCS expert. But if you remember from SFPS, we did consider three EP models. And, you know, getting back to your question of how important it is, you know, they did model it and it was not that important precisely because of what Steve was saying that, you know, this is a very, very small event and sensitive to different EP models.

MEMBER STETKAR: But that was still for the - that site.

MR. ESMAILI: That's right. But we did look at, you know, the - yes. So, there is some sensitivity that we have considered.

CHAIRMAN ARMIJO: Okay. Just to let everybody know that I'm going to - somehow we didn't have time for a break in the agenda, but I'm going to shoot for somewhere around 10:30 for a 15-minute break. So, Steve, we'll try and not mess up your presentation.

(Laughter.)

(Discussion off the record.)

MEMBER BALLINGER: I have a question about

the line of fragility issue. You're talking about assumptions.

MR. JONES: Okay, yes. We do have another slide that gets a little bit more into that detail.

MR. WITT: Slide 14.

MR. JONES: Yeah, Slides 14 and 15.

MR. WITT: If we ever get there.

CHAIRMAN ARMIJO: We're on 13?

MR. JONES: No, we're not there yet.

CHAIRMAN ARMIJO: Okay, sorry.

MEMBER BALLINGER: Okay. Neither one of those slides addressed my concerns.

MR. JONES: Okay.

MEMBER BALLINGER: I was onboarding at the time of the spent fuel pool study. So, I'm sort of in between, but have you guys gone back and looked at the basis for the event that causes a ripping of the liner?

Because NUREG-6706, which is the basis - is it - 6706? I actually remembered it. That's not for stainless steel. That whole study was based on carbon steel.

The properties of carbon steel are way different than stainless steel. The toughness of stainless steel is much higher than carbon steel.

So, we've been talking about releases and

all that kind of stuff, but I'm a little bit worried that we're not adequately treating the thing that allows for the release to start with and that a better treatment of that - or not a better, but a more thorough treatment of that might cut the head off of a snake, so to speak, because the liner is so tough compared to the properties, maybe a factor of two, that you were using.

So, anyway, it's just a, you know, I don't know because I was in between, out and in on the ACRS, I don't know how that was treated.

MR. JONES: Okay. I guess going back to the spent fuel pool study, that event, there was pretty minor, really, relative motion of the wall relative to the floor, but there's enough to cause like I think it was 20 percent of the strain that might normally be associated with failure.

And for that reason, the spent fuel pool study used a ten percent overall probability of liner failure. And that was based on using those stainless steel properties in that case.

MEMBER BALLINGER: I don't think so.

MR. PIRES: This is Jose Pires.

We were conservative on the failure strains for the stainless steel, but also at the - when you get to the very large crackings of the wall and you start

increasing the flow, you start seeing a very large increase on the displacement.

If you keep increasing the load at that stage, the displacements that open the crack would start increasing in a very nonlinear manner.

So, if you - yes, it was a conservative assumption on the various strains for the liner, but you was changing those strains very rapidly at those load levels.

MEMBER BALLINGER: Yeah, I understand that, but the ductility of stainless steel is twice -- it's -- 40 percent compared to carbon steel of 25 percent or something like that.

And so, the whole basis for determining how big this rip is, I just worry that we're not -

CHAIRMAN ARMIJO: Well, if, in fact, Ron has it right that the mechanical properties of carbon steel were used to determine the amount of strain, then that's incorrect. It should have been the mechanical properties of stainless steel including the ductility.

I missed that point in my view whether it was - because it should have used the mechanical properties of stainless steel.

And, Jose, do you know for sure that the liner properties used in your analysis were for

stainless steel?

MR. PIRES: The greater strains we used on carbon steel versus stainless steel. So, there is a conservative assumption there, but we also have some lack of knowledge on, for instance, not in this particular pool, but in other pools what is the welding.

Is there the transition between the liner of the wall to the floor? If there is a welding joint there, that might have been degradation on the welding.

Not in the case of this pool. In this pool, the detail was that in a different manner that was better. So, also as I mentioned, you have displacements at those load levels increasing very rapidly.

So, even if it is not safe that you get - just a hypothetical number. If you don't get the very large strains, let's say, at the 0.8 g or 0.9 g, you will get that at probably 0.1 -

MEMBER BALLINGER: I'll grant you that. It's just a matter of scale though. The same material of steel will perform differently for the same set of displacements than stainless steel.

CHAIRMAN ARMIJO: Right.

MEMBER STETKAR: The only comment I make is that's certainly a valid concern. It certainly would affect the absolute frequency of failure.

It would not affect a comparative analysis here. Because if it fails under the earthquake for high density, it's going to fail under the earthquake for low density.

So, although the absolute magnitude of the frequency of failure can be affected by the properties, the difference between high density versus low density for the purpose of this regulatory analysis wouldn't be affected.

MEMBER BALLINGER: But I worry about a failure at all.

(Simultaneous speaking.)

MEMBER STETKAR: At some loading, it will fail.

MEMBER BALLINGER: Well, okay. If you explode a nuclear device on the site, you will get failure. You're right.

MR. PIRES: Well, as I said - I keep saying that at the load levels where we used that to get the large strains in the liner, those strains change very rapidly with the load level.

You want to - at the very - at the region where the stiffness of the bolt has degraded and you have a rapid change on the strain. So, if you did increase the load, but at larger loads, but will still

happen.

CHAIRMAN ARMIJO: But there is a profound difference between the properties of stainless steel and plain carbon steel. And the stainless steel is tougher. It will work harden more. Even under greater loads it's much more resistant. So, it will take an awful lot more seismic loading before -

(Simultaneous speaking.)

CHAIRMAN ARMIJO: But the point is I think maybe what Ron is saying, you say, look, these stainless steel liners are incredibly tough. And if we use the wrong properties in the spent fuel pool study, we ought to correct it and just say, hey, look, there's much more margin here.

MEMBER BALLINGER: NUREG-6706 is capacity of steel and concrete containment vessels with corrosion damage steel.

We have lots of data on casks that have been dropped on, what do they call it, immovable objects from a height of whatever it is, stainless steel casks where they've undergone enormous amounts of deformation and still not failed.

Okay. I just - my concern is -

MR. PIRES: The other thing -

MEMBER BALLINGER: That's it.

MR. PIRES: The other thing that you have is also you may have different failure modes that can - and I go back to the same thing is you might have some failure modes where you could have the large strains near the welding. Those are for the welding areas. We were told that there may be degradation on those, because there had been some cold forming of the steel that could have - that also might reduce somewhat the various strains of the liner.

In addition to that, as I said, the strains when you start getting the very large failures, you have very large increases in strains, but smaller increases on the load.

MEMBER RICCARDELLA: That's assuming the concrete is cracked?

CHAIRMAN ARMIJO: Oh, yeah. The concrete always cracks.

(Simultaneous speaking.)

MEMBER RICCARDELLA: I mean, that's what causes the large increases in strain?

CHAIRMAN ARMIJO: Yes.

MR. PIRES: I agree with you. It is a conservative assumption on that.

CHAIRMAN ARMIJO: Well, it's kind of a hidden conservative -

MEMBER STETKAR: By the way, my earlier comment, this discussion doesn't affect the conclusion in the regulatory analysis on high density versus low density at least as far as I - it could be important, though, in terms of presenting the results to the public, because the absolute frequency, you know, of both could be substantially reduced.

Substantially, factors of two are not substantial to PRA people, but -

(Laughter.)

(Discussion off the record.)

MEMBER STETKAR: At some seismic acceleration it will -

CHAIRMAN ARMIJO: Of course, John.

MEMBER STETKAR: That's the whole point of looking at the frequency and the consequences is to understand if what we're talking about potentially large consequences and it's important to understand what the frequency of those potentially large consequences may be.

It's not just one or the other. I mean, that's why the absolute frequency can make a difference when you're presenting the results, because it will scale both of them down.

MEMBER RICCARDELLA: Would the probability

of leakage under this 0.7 g earthquake change significantly if you use stainless steel -- in the stainless steel liner?

CHAIRMAN ARMIJO: Oh, yes.

(Simultaneous speaking.)

MR. PIRES: It's also in the base case analysis I understand that the more controlled results is what we call the Bin 4 - are the loads on the Bin 4, not the loads on the Bin 3.

In the Bin 4, you have much higher levels. So, it is - it's also - and that's back to the fact that the strains will decrease rapidly as the loads increase when you get to the Bin 4 pack acceleration.

MEMBER BALLINGER: And I'll just say it one more time. As a matter of scale, there's a point at which you get failure for carbon steel. There's a point at which you get failure for stainless steel. They're very different.

And so, the initial - the initial starting event is affected by those properties. And it is on a good nonlinear that stainless steel is very tough material.

MR. PIRES: And we have concerns. I mean, materials - people I talk to, they have concerns about - mostly about degradation of welds under water for 30

years, 40 years.

MEMBER BALLINGER: I'd be more worried about carbon steel welds than I would be about stainless steel welds.

MR. PIRES: I understand.

MEMBER BALLINGER: Okay, enough.

MEMBER RICCARDELLA: But neither addresses the welds.

MR. PIRES: No.

CHAIRMAN ARMIJO: Okay. So, we've got a materials issue on the table, but I share Ron's - both of us being materials guys, you know, you're really - and shame on me for missing the fact that it was carbon steel properties used in the analysis as opposed to stainless steel, but it's a hidden conservatism. Probably not intended to be hidden, but it -

MEMBER BALLINGER: The report that they used contained the methodology for determining the fragility numbers.

The materials that they used were carbon steel. So, the methodology -

CHAIRMAN ARMIJO: No, I have no problem with the methodology. I thought that was good. In fact, we said so in our letter, but it's the properties, the stress strain curves of carbon steel and stainless steel

are very, very different. And at these temperatures, the stuff is really ductile and tough.

So, and that explains one of the reasons why the Fukushima plants and the Kashiwazaki plants subjected to these earthquakes that you point out in your study performed so well.

Okay. Go ahead, you know. You take a stopping point, Steve, whether it's this chart or the -

MR. JONES: Okay. We'll finish this slide and the next slide, I think, and then we'll -

CHAIRMAN ARMIJO: Okay.

MR. JONES: Okay. Just the last couple items here. Population density and economic activity were based on the Surry site as a mean.

It's higher than the median levels for economic costs, but lower than the upper bound sites. The high case, for instance, used Peach Bottom representative of the 90th percentile.

And then the industry implementation costs were just derived from the EPRI information. I forgot that before.

Next. The one assumption that we were, I guess, very constrained with was what we were just talking about. Really is to some extent, is liner

fragility where the liner might fail and the resultant pool level and also what fuel distribution may exist in the pool at the time of the event.

All those can affect the ability of the pool to - or the ability of air cooling to provide adequate heat removal.

Okay. That results in the dominant initiating events generally - the assumptions we make here is, for the most part, air cooling would be insufficient. And that results in the dominant initiating events progressing the fuel heat-up and if there's no mitigation like for the high-density case to a release.

This is conservative, because the spent fuel pool study and other studies have identified substantial potential for air cooling when the pool is either fully drained or when the fuel is particularly - has a particularly long decay time and is not generating much heat.

We did make an exception for the Mark I and II BWRs that were the focus of the spent fuel pool study.

In that case, we have a lot less uncertainty and we used the eight percent value for just covering the first part of the operating cycle where the fuel is particularly hot and, therefore, would heat up to

a release potentially.

Okay. Next slide. I'll do this one and then we'll stop.

CHAIRMAN ARMIJO: Okay.

MR. JONES: Okay. For seismic event frequencies we used the Peach Bottom seismic hazard which falls near the upper end for all the sites considered in the central and eastern United States.

It's lower than the bounding site, Sequoyah, by a factor of a little over three. And Sequoyah is a Group 4 plant, which is the shared pool.

Okay. For population demographics, I talked a little bit about this in the last slide. The Surry population was used and it's above the median for all sites.

There's a sensitivity evaluation in the regulatory analysis that addresses the effect of looking at higher population density sites and using Peach Bottom within 50 miles would have increased the benefits by about 28 percent compared to the Surry demographics.

Most of the other - the other assumptions all have generally smaller affects. But added up I guess when you look at the highest in the cases, a lot of little factors adding up to a very large increase in the potential benefits from the - in the alternative.

I guess before we get into the table, we could -

CHAIRMAN ARMIJO: Yeah, I think --

MR. JONES: - take a break.

CHAIRMAN ARMIJO: I think it's a good time for it. Why don't we take 15 minutes. So, let's be back at 10:40.

(Whereupon, the proceedings went off the record at 10:26 a.m. for a brief recess and went back on the record at 10:43 a.m.)

CHAIRMAN ARMIJO: Okay. Go ahead, Steve.

MR. JONES: All right. Together this slide is to basically demonstrate how the base case frequencies for heat-up of the fuel and release might occur for these - the different initiating events that were considered here.

Okay. For Seismic Bin 3 we're looking at a 0.7 PGA earthquake. That's somewhat higher than the 1.2 g, I mean, there's different measures for the seismic acceleration. So, I do want to make clear that that's different than the 1.2 g fragility that was assumed in NUREG-1738, because that corresponds to 0.5 g PGA.

So, it's slightly less than the Bin 3 earthquake. So -

(Simultaneous speaking.)

MR. JONES: Anyway, Bin 3 earthquakes, what we're looking at here at 0.7 peak ground acceleration is more severe than what was generally considered in the NUREG-1738 as a fragility point. A point at which liner fragility would be a concern.

And the numbers are a little bit confusing, because they kind of overlap. They have 1.2 showing up in two different contexts if you look between the two studies.

But anyway, for this case we looked at Peach Bottom for the base case frequency. And taking basically an average of the seismic hazard during - from the 0.5 to 1 g realm you end up with a 0.7 peak ground acceleration being the average. And the frequency drawn off for that seismic hazard was 1.65 times ten to the minus five.

For the liner fragilities for the elevated pools, we're using what was assumed or essentially based on current calculation from the spent fuel pool study of ten percent for the liner fragility.

For the at-grade pools representing Groups 2 through 4 we used five percent. And then for the inadequate cooling, and I mentioned this previously, for the elevated pool again we're looking at eight percent of the operating cycle where air cooling would

be ineffective.

CHAIRMAN ARMIJO: I just don't understand the hundred percent inadequate cooling for the at-grade pools when the liner fragility is half of the elevated pools.

Could you explain why that -

MR. JONES: Okay. We're really trying to encompass a lot of conditions that could affect the adequacy of cooling.

One of the principal ones is the location of the liner tear, because the pool is at or near grade and you have different supporting structures around the pool.

The potential for there being a shear condition in the pool structure might be somewhere other than at the bottom of the pool. And, therefore, you have greater potential of blocking the natural circulation air cooling.

And also, we don't have full publicly available MELCOR analyses of the PWR assembly performance under low decay heat cases with the partial joint conditions.

CHAIRMAN ARMIJO: So, you just picked it as a bounding situation to cover all of those uncertainties.

MR. JONES: Right.

CHAIRMAN ARMIJO: Okay.

MEMBER RICCARDELLA: And why, again, is that so much higher than the elevated pool?

CHAIRMAN ARMIJO: The elevated pool -

MR. JONES: The principal difference is the - for elevated pools, the leakage location was determined based on structural analysis for the Peach Bottom plant to be at the bottom of the pool.

So, you have a full drainage of the pool for most of the conditions that allows air circulation to, you know, cool and to be drawn underneath the racks and to go through the assemblies and provide adequate cooling after a certain number of days have passed into the operating cycle since the fuel was last used in the reactor.

For the PWRs, we're saying we don't really know where the most likely leak location would be. And if you have a partially exposed fuel, there is potential for the upper part of the fuel to heat up to the point of release.

MEMBER RICCARDELLA: So, a partially drained pool in some cases could be worse than a fully drained pool.

MR. JONES: Right.

MEMBER RICCARDELA: I think that was one of the comments from one of the other outside comments.

MR. JONES: That's a principal concern. There's also issues with distribution of the fuel. There may not be fully distributed in the assumed configuration at all times during the operating cycle. That hundred percent covers that as well.

MEMBER RICCARDELLA: So, these cases, the 0.7 g and the 1.2 g, are they combined into your overall, you know, with the different probabilities that are combined?

MR. JONES: We're adding the heat-up frequency at the end to total an initiating event frequency or for a release frequency, basically, for the case of the high-density fuel storage.

MEMBER RICCARDELLA: But was there any look at smaller, like, Bin 2 earthquakes or, you know, combining those which might be higher frequency recurrence, but lower fragility?

MR. JONES: No, we didn't - we based, in part, on -

MEMBER RICCARDELLA: So, anything less than the 0.7 you're saying you have a hundred percent -

MR. JONES: Well, the Bin 3 is meant to cover 0.5 g to 0.7 g.

MEMBER RICCARDELLA: Yeah.

MR. JONES: I mean, sorry, 1 g in the peak ground acceleration and that goes above the fragility limit.

What I was mentioning was for NUREG-1738 these was assumption of at 0.5 peak ground acceleration - 0.5 g peak ground acceleration the pool would maintain its integrity.

MEMBER RICCARDELLA: Zero probability.

MR. JONES: Zero probability of leakage.

MEMBER RICCARDELLA: For leakage, okay.

MR. JONES: And then going up to Bin 4, you know, we really do have a pretty rough seismic input coming from USGS studies for the 2008 values.

This is an estimate of the average frequency for a very severe earthquake over 1 g. And from that, we used a pool liner fragility of a hundred percent for the elevated pools, and 50 percent for the at-grade pools.

This predominantly comes from actually earlier studies that looked at Vermont Yankee and Robinson spent fuel pools and the relative fragilities of those two pools.

We wanted to give some benefit for the PWR pools that don't have the same level of amplification

due to higher elevation of the PWR pool.

MEMBER RICCARDELLA: Do you know if those earlier studies you referred to assumed carbon steel or stainless steel liner?

MR. SCHOFER: 1738 does.

MR. PIRES: This is Jose Pires.

Those studies did not really calculate liner strains. They assumed that there will be a crack on the concrete and that the crack on the concrete will grow sufficiently large to cause failure of the liners.

They didn't go into detail on even trying to estimate strains in the liners, but they considered a crack that would be susceptible to growths. Very large growths.

CHAIRMAN ARMIJO: So, the liner really didn't play a role in the analysis. It was just assumed to fail?

MR. PIRES: Yes, but they had a failure mode on the concrete that was somewhat brittle. So, their assumption was that the crack would grow quickly and it would drag the liner with it.

CHAIRMAN ARMIJO: Okay.

MEMBER RICCARDELLA: Yeah, I mean, if you get a significant amount of concrete cracking, it seems to me that a factor of two difference in liner ductility

wouldn't make that much difference.

CHAIRMAN ARMIJO: You know, there's not much strain in the concrete.

MEMBER RICCARDELLA: Well, once it cracks, though, and if the liner is going -

CHAIRMAN ARMIJO: That opening is pretty tiny.

MR. PIRES: Once the concrete cracks, it depends on the characteristics of the load. Depends on whether there is some ductile behavior or if the crack spreads faster.

So, at that time the assumptions were made in a very simple way that they assumed the crack would grow large enough to strain the liner beyond this -

MEMBER RICCARDELLA: To strain regardless of the -

MR. PIRES: Right.

MEMBER RICCARDELLA: - material ductility.

MR. PIRES: Those were the assumptions that were made.

MEMBER RICCARDELLA: Okay. Thank you.

MEMBER REMPE: So, the values you have on this slide are consistent with Table 39 of your report, which is way back in an appendix, but Table 2 has incorrect values.

Is there going to be an update or something or revisions to what's been published or -

MR. JONES: I guess - we didn't recognize this in advance. I think pointing it out, I think we'll assess whether or not it's incurred enough to find a route to the Commission, because it's already been issued.

MEMBER STETKAR: Table 2 is wrong. These are actually -

MR. JONES: Table 2 is wrong. That's correct. For the liner fragilities, we have the wrong values.

MR. WITT: Yeah, we'll see if we can get a correction.

MR. JONES: Then for the cask drop, the two to the minus seven value comes from NUREG-1738. And in that case, that actually essentially considered liner fragility in that analysis.

And there is - well, they're assuming an inadequate cooling for that case also. And but I do want to point out that cask drop is not really a credible failure for all plants.

In many cases there are - the crane that handles the cask is configured in such a way that it can't pass over the spent fuel pool with a load or it

has some type of operating limits that prevent it from going over the pool. And the cask loading area is separate from the spent fuel pool structurally.

Okay. And then all the other initiators, there's a wide variety that were considered in NUREG-1738. Again, they include additional factors beyond the initiating event. Like I mentioned, initiators as station blackout is one. A pipe break in the cooling system might be another.

And these would - then you'd have mitigative activities that are simplified human error probability analyses listed in NUREG-1738 that results in -

MEMBER STETKAR: So, for those other initiators, both the high density and the low density take full credit for all of those other mitigation for -

MR. JONES: Right, but those are not - that's not using the 50.54(hh) mitigation. That's strictly existing firewater systems or maybe servicewater or other makeup means that are available onsite.

It doesn't consider spray, which is the, you know, predominant benefit of the B.5.B or the post 9/11 actions.

CHAIRMAN ARMIJO: So, none of these other things, the cask drops, other initiators, they're not

station blackout conditions. They're just normal operation.

MR. JONES: No, no, no. They are. They are.

MEMBER STETKAR: That 2.37 times 10 to the minus seven ostensibly includes the frequency of every other possible initiating event other than the seismic event and cask drop that could possibly happen at the site with every possible event sequence that could be developed that could result in loss of fuel pool cooling. That's what they're claiming.

MR. JONES: A lot of these are very -

MEMBER STETKAR: And, again, those are really small numbers and they're certainly really precise.

I'm not at all clear that they're very accurate. They're certainly very precise.

CHAIRMAN ARMIJO: Okay. Okay, good. Thank you.

MR. JONES: Okay. And then from those results we get the numbers at the bottom that were used for the base case event frequencies.

CHAIRMAN ARMIJO: This is what you worry about, this Bin 4.

MR. JONES: And about 90 percent of that is

a seismic contribution. And if you go looking up further back up at Bin 4, Bin 4 is the dominant contributor.

MEMBER RICCARDELLA: Yes. 88 percent Bin 4.

MR. JONES: Okay.

MEMBER RICCARDELLA: And these were applied equally to both options, the low density and high density?

MR. JONES: Right. Both low density and high density. The only difference here, again, is the application of mitigation would reduce the frequencies for - of actually going to a release for the low-density cases.

MEMBER RICCARDELLA: Okay.

CHAIRMAN ARMIJO: But would mitigation affect the heat-up frequency?

MR. JONES: I guess it depends on when you consider the - I was looking for a good word to describe - we're in this intermediate state where the fuel is heating up. And if you don't do anything else, it will lead to a release.

CHAIRMAN ARMIJO: To me -

MR. JONES: For mitigation to be effective, it does need to be deployed relatively early probably

before the floor is exposed unless you have a separate area.

CHAIRMAN ARMIJO: Yeah, I see mitigation as preventing fuel heat-up as opposed to -

MR. JONES: Bad word selection, yeah.

MR. WITT: This is really initiating event frequency. I mean, all these things considered multiplied by the consequences gets you the risk or the consequences.

MEMBER SCHULTZ: This would be it without mitigation. So, the low-density case gets mitigation.

CHAIRMAN ARMIJO: Yeah. So, this is without.

MEMBER SCHULTZ: The other case does not.

MEMBER RICCARDELLA: Yeah. And there's no difference in the heat-up - in the fuel heat-up rates for low density versus high density.

CHAIRMAN ARMIJO: Well, yeah.

MR. JONES: That's correct. The event progressions are basically the same given all the other external conditions are the same, because the - it's driven by the very hot assemblies which are present in both the low density and high density cases.

(Discussion off the record.)

MR. JONES: I have a results slide here, but

I'm not certain if we -

CHAIRMAN ARMIJO: We're not quite ready for that results slide.

MR. JONES: Okay. So, we have backup slides available that we can talk about Table 2 or look at some of the other progressions.

CHAIRMAN ARMIJO: Yeah, why don't we, you know, I have a couple of questions on Table 2 other people may have.

MEMBER STETKAR: Let me ask you before we get to that, Sam, I need some clarification. We don't have backup slides for this. So, you mentioned this in October offline, but I'm going to put it on the record now.

If I look at the report and I compare the base case results, and write these down, in Tables 4, 44, 54, 56, 60 and 64, okay, that will give you the scope of the things that I looked at and I'm only looking at base case now, I notice that there are distinct differences in the sense that those tables align in two, what I'll call, collections, because I want to avoid the word "groups."

Collection Number 1, and those are Tables 4, 56 and 64, gives me base case dose averted values for spent fuel pool Group 1 of 1740 person-rem. For

Group 2, 1630. Group 3, 3,020. And Group 4 is 1690.

Now, if I look at those qualitatively, I don't care about the absolute values. It's - and if I normalize them to Group 1, it says Group 2 is a little bit lower, Group 3 is about 75 percent higher and Group 4 is about the same as Group 1. Okay.

Now, if I look at Tables 44, 54 and 60, Group 1 got corrected between October and today. So, it now has a dose averted of 1739 person-rem. Group 2 has 2109. Group 3 has 3616. And Group 4 has 2284.

Groups 2, 3 and 4 in that second collection are much, much different than Groups 2, 3 and 4 in the first collection as are the relative fractions when I normalize it to Group 1.

Group 2 in the second collection is now higher than Group 1. Group 3 is a factor of two higher. And Group 4 is somewhere in between.

I did my own back-of-the-envelope calculations and I don't do MACCS runs. I don't have all of these sophisticated computer tools. All I have is a spreadsheet.

The qualitative behavior of Groups 1, 2, 3 and 4 in my little calculation seem to behave more like the second collection than the first collection.

In other words, if I normalize to Group 1,

Group 3 is the highest, Group 2 is a little - is between Group 1 and 3, and Group 4 is between Groups 2 and Group 3.

So, I'm curious now if I look at those table of results, why are they different? Why do we have these two different collections for the base case?

Now, it used to be in October that the Group 1 was different between the two collections, but somehow that got corrected. So, Group 1 is now consistent, but Groups 2, 3 and 4 are different.

You probably can't do this realtime, but there are differences. And if those differences are used in the overall results of the study, and I maintain that they are, I'm not clear now what the sensitivity studies are telling me and which set is correct, if either.

(Discussion off the record.)

CHAIRMAN ARMIJO: Well, these are all on the same order of magnitude, right?

MEMBER STETKAR: They're on the same order of magnitude, but the important thing is - I don't care about the absolute values. The behavior is different also.

CHAIRMAN ARMIJO: Right. Yeah, yeah.

MEMBER STETKAR: Whereas if I just compare

One and Two in Collection 1, Two is - Two gives me lower releases than one. Whereas in the second collection, Two gives me more releases than Group 1.

CHAIRMAN ARMIJO: Yeah, yeah. Same with Three.

MEMBER STETKAR: You know, so something fundamentally is different in those two collections the way the model seems to be developed.

And it's not just a simple typo, I don't think, because it propagates through all the costs. I mean, it isn't one column and one table in the report, because it propagates consistently through the cost estimates.

MR. WITT: It seems like something we'd have to go back and -

MEMBER STETKAR: You can't do it in real -

MR. WITT: - determine where these numbers came from.

MEMBER STETKAR: Right.

CHAIRMAN ARMIJO: But, you know, whichever is the right set of numbers, they should be throughout the report and some explanation of -

MEMBER STETKAR: Anyway, I'll just -

CHAIRMAN ARMIJO: But for the -

MEMBER STETKAR: It's an observation. As

I said, I - my little back-of-the-envelope calculation seems to be behaving more like what I call the second collection, which is Tables 44, 54 and 60, than the first collection, which is Four, 56 and 64.

The problem is that Table 4 is your base case results that you highlight up front in the study.

CHAIRMAN ARMIJO: Yeah, yeah.

MEMBER STETKAR: And if that's wrong for some reason, there could be a concern.

CHAIRMAN ARMIJO: So, the benefits or the averted person-rems are greater for the plants in Groups 2, 3 and 4 than for Group 1? Is that -

MEMBER STETKAR: No, you can't -

CHAIRMAN ARMIJO: You can't come to that?

MEMBER STETKAR: Yeah, don't look at - don't look at - these are just numbers getting out to -

MR. WITT: I see what you're saying in terms of the differences in the tables, the -

MEMBER STETKAR: Well, the absolute values are different. So, if I look at - if I just look at averted person-rem in Group 2 in Collection 1, if you're following me, the averted person-rem in Group 2 in Collection 1 is 1630. In Collection 2 it's 2109.

MR. WITT: Which is higher than -

MEMBER STETKAR: Which is 470 some odd, you

know. And I don't know why that is different, because I'm just comparing so-called base case values.

MR. WITT: Right.

MEMBER STETKAR: So, not only the absolute value is different. But because the absolute values are different, the relative ranking, if you will, of those groups when I - if I normalize it to Group 1, if I called Group 1 my normative condition, the relative rankings of the groups become different also in terms of what is - what gives me more releases versus less releases than Group 1.

MR. WITT: Yeah, I think this is something that we'll definitely have to investigate to see where those numbers came from.

MEMBER STETKAR: Okay. I wanted to do it here. I was waiting until we got here, because this is the only place where you sort of talk about all of, you know, the sensitivity study/cost-benefit analyses in one place.

MR. JONES: Sure.

MEMBER STETKAR: And I think we've probably sent enough time on that. We can now go to Table 2 where Sam wanted to -

CHAIRMAN ARMIJO: Yeah, I wanted to get into this health consequences part of the table.

You have a statement under - in the table that says the LNT dose response model is used as the base for reporting results. And that statement is the dose truncation methodology introduced in SOARCA analyses documented in 1935 is provided as a sensitivity analysis.

It's not actually - you didn't actually do a sensitivity analysis in this document, did you? Or did you just say SOARCA did the sensitivity -

MR. SCHOFER: SOARCA did the sensitivity analysis.

CHAIRMAN ARMIJO: Okay. Because I was looking all over in the document for this sensitivity analysis. And you're saying that if you really wanted to know what benefit you could get, you'd have to go look at SOARCA.

MR. SCHOFER: Correct.

CHAIRMAN ARMIJO: Okay. I understand what you did then.

MR. SCHOFER: Yeah, you could either do that or -

CHAIRMAN ARMIJO: I was looking for it, because I was saying, great, now we'll have -

MS. GIBSON: For spent fuel pools you need to look in the spent fuel study. For a reactor analysis,

it would be in SOARCA. We use the same truncation in both SOARCA and the spent fuel study.

CHAIRMAN ARMIJO: Okay, but it's not - it's nowhere really visible, but the effects of truncation are in this report. At least I don't see numbers or anything like that.

MEMBER SCHULTZ: It's a remarkable difference or -

CHAIRMAN ARMIJO: It's a huge difference.

MEMBER SCHULTZ: - associated with in the spent fuel study.

MR. SCHOFER: A couple thousand difference. A factor of a thousand.

CHAIRMAN ARMIJO: Yeah, and that's - to me, that's so important. And it's not really - doesn't get much - it doesn't get any visibility particularly when you're extending it to beyond 50 miles and huge populations and habitability for 50 years and on and on and on.

It just multiplies and accumulates and seems to me it should get more visibility and that's just an observation. So, I won't hold you up anymore on that. I just wanted to see if I had any other questions.

Okay. I don't have any other question

myself on that table. Anybody else? Keep going, Steve. You're on a roll.

MR. JONES: Nothing else on Table 2. I guess we can go back to the -

MEMBER REMPE: I think we've covered it at other places, but again I guess I would really if you are going to issue some sort of update with corrections, I sure would like to see more explanation.

This report is getting a lot of visibility and there's a lot of assumptions in here that - it's coming from the spent fuel scoping study, but these values are - the factor of 19 isn't identified in that section or in that table. And, you know, these things - we have these discussions here. I know where it's coming from, but it's not obvious to the reader, I think.

And so, those kind of things, I think, should be documented better.

CHAIRMAN ARMIJO: Well, you know, when the Committee writes a letter, we may point that out, you know, some things that could be improved or that would be helpful.

MEMBER REMPE: Yeah.

MEMBER STETKAR: And a factor of 19 is just the 95 percent mitigation.

CHAIRMAN ARMIJO: Well -

MEMBER REMPE: But isn't it coming from the scoping study and the table in the scoping study where they found that value and it's not clearly stated?

It just seems like, you know, it's something that could be documented better.

CHAIRMAN ARMIJO: Bill calls it the artificial factor of 19. If he was on the line, we could ask him to expound.

MEMBER STETKAR: It's the 95 - I don't know why you use 19, but it's 95 percent gets you 19 out of 20 if you want to think of it that way, of the stuff recovered. Five percent is not recovered, which is like 120th.

MR. JONES: Right.

MEMBER STETKAR: It's not really a factor of 19.

MEMBER REMPE: Isn't it coming from Table 33 of the scoping studies where you got that factor of 19?

MR. WITT: Is that the HRA? Do you remember the -

MEMBER REMPE: The mitigation. And then -

MR. JONES: It's not coming from the HRA. It's coming really from - it looks at the - the scoping study had an assumed response for mitigation and it

relied on the rate of level decrease in the spent fuel determined like what response whether it would be mitigation by makeup or mitigation by spray.

And for one of 20 scenarios looked at for the low density storage configuration, the level drop would be occurring very slowly, because it's actually very early in the outage when the reactors connect with the spent fuel pool.

And that leads based on the methodology in the study, the operators to determine makeup by just additional water to the pool as the appropriate response when spray is really necessary to effectively mitigate the condition.

So, there's a failure and it's modeled in the spent fuel pool study for that one out of 20 evaluated cases. That's really the - did I cover that correctly, Hossein?

MR. ESMAILI: Yes. So, in the spent fuel pool scoping study we looked at medium - moderate leaks and small leaks.

The small leaks were of no concern because as soon as you got the mitigation, you were always recovered. You never got any releases.

The case with the moderate leaks you could not with the 500 gpm, you could not recover. So, there

was a period of time in OCP1 during the first week the fuel is so hot.

So, whether you inject or you spray, you're still going to get a release. And that constituted that five percent of the time for half of the damage state, you know.

So, only for moderate leak cases only during the first week even with mitigation you get a release.

MEMBER REMPE: Okay. I think it ought to be documented.

CHAIRMAN ARMIJO: Yeah, you either have to expand that to make it easier to understand --

MEMBER REMPE: Yea.

CHAIRMAN ARMIJO: - or as, you know, some of us believe, it would be so much better if you just treat both alternatives; mitigated and unmitigated.

And the benefits of the Alternative 2 for the unmitigated situation would be evident, but the effectiveness of mitigation for the base case for the Alternative 1 would be clear, too, so a decision-maker isn't left hanging with just a big advantage on Alternative 2 when it's without realizing the advantage of Alternative 1.

Anyway, that's again consistent with some of the things that Bill would have contributed if he

were here.

Okay. Keep going.

MR. JONES: Okay. I guess we can go back to Slide 16.

(Discussion off the record.)

MR. JONES: So, we've talked about, I guess, basically the overall assumptions. The benefits - or excuse me, the base case costs outweigh the benefits particularly when we're looking at within 50 miles and $2,000 per person-rem, which is the standard regulatory analysis approach. And the changes in discount rate do not affect that result.

Sensitivity analyses for - address commissions involving $4,000 per person-rem and consequences extending beyond 50 miles from the plant. In that case, there is margin benefits in some of the cases.

The costs continue to outweigh the benefits for Groups 1 and 2. And Groups 3 and 4, the benefits marginally outweigh the costs.

The main difference driving that for Group 3 is, you know, there is a longer period of operational life and really the costs are a little bit, I want to say, lower, because the cask purchase is deferred later in the life of the plant relative to the other cases

we're looking at.

For Case 4, that's the case where you have a shared spent fuel pool and a higher inventory of cesium present that really drives - so, there's additional benefits from avoiding that release.

CHAIRMAN ARMIJO: Just to - and I'm asking you to guess. If the - for Groups 3 and 4 where the base case benefits currently marginally outweigh the costs, if the costs for these casks and expedited process were off by a factor of two from what you used, would that still be the situation?

I've never been involved in a procurement that hasn't been off by a factor of two.

(Laughter.)

CHAIRMAN ARMIJO: Even after we scrubbed it.

MR. JONES: Yeah, I think it came out cost beneficial by a very small - by a fraction of the total cost. So, I would expect that - and most of the costs are the procurement of the casks.

MEMBER SCHULTZ: So, if you took Groups 3 and 4 and determined that for those groups it was warranted to go to a further investigation, wouldn't the first thing that you would do be to reevaluate the conservative assumptions that have been used in this analysis for the base case as compared to the low case,

which in many areas of assumption is the low case is more conservative than what was derived in the spent fuel pool study, which was the most - is the most recent evaluation that we have done. And then these conclusions would disappear associated and would marginally outweigh the costs.

It would be clearly demonstrated that there is no benefit for those groups as well doing anything differently than what is currently done in the spent fuel pools we have today.

Is that a fair evaluation?

MR. JONES: One thing -

MEMBER SCHULTZ: In other words, one of the things we tend to do is we tend to make lots of assumptions in order to create an evaluation technique that can differentiate between one option and another. And then we begin to apply it to other cases.

And when we do, we get a - as it's stated, marginally outweigh costs. But, in fact, if one were to go back and do something that was just not best estimate, but just somewhere directed toward a more reasonable evaluation, there would be no difference demonstrated. And we begin to lose that when we draw general conclusions like this.

MEMBER RAY: Well, at the very least a lot

of the comments around here have been to make more clear to what assumptions were made.

I think a factor of two is way low. I had to go off and build casks at one time in my life. I know a little bit about it.

And if everybody is out trying to get casks even over a five-year period or so, their costs are going to be - and that was because at any price you couldn't get them, period.

So, I just think it needs to be highlighted if we're - and it looks to me like we're stuck in the position that we're in just from the standpoint of what's practical to do. I'm talking about the timing of this phase that we're engaged in now.

It just needs to be more clear that, you know, we really have no idea of what the cost of casks will be when everybody is trying to do this. And to just pick a number and say, well, conservative, assume it's twice what it was, I don't even think that's good enough.

CHAIRMAN ARMIJO: Well, it's better than one.

MEMBER RAY: Well, yeah, but there's a sense in which making things better you think you've made them good enough and that's not necessarily true.

I just think we ought to make clear in a footnote or somewhere that this assumes there's no affect on the price of the cask.

MEMBER BALLINGER: In fact, I talked to Transnuclear, Holtec - all three cask manufacturers, because there's an estate program dealing with canister life and NRC is involved in some of that, and asked them the blank question, could you do it? And every one of them said no, not possible to respond in time to build enough casks.

MEMBER RAY: Well, you'd have to have new -

MEMBER BALLINGER: Yeah.

MEMBER RAY: - capacity to build casks that would be amortized over the period that this demand would exist. And then after that you'd have a lesser demand than what was the basis of the existing manufacturing capability and so on.

So, it's a complex analysis and Sam suggested maybe go overseas. Well, maybe you would.

CHAIRMAN ARMIJO: Well, that's what would happen.

MEMBER RAY: There are some complications associated with it. Rather than trying to sort them out at this late date -

CHAIRMAN ARMIJO: Right.

MEMBER RAY: - I think it's best to simply say we don't have any idea.

(Laughter.)

MEMBER RAY: Well, we don't. That's the real, you know, we don't have any idea. And, like I say, I couldn't buy them at any price when I needed them.

CHAIRMAN ARMIJO: Well, you know, what bothers me is the base case would normally be what you would use for a decision. And if your base case analysis shows a benefit, you have a hard time saying we - it's - we don't want to - we don't think it's worth doing or pursuing further. And it just seems like there's - somebody would sharpen their pencil and say, you know, did we overdo our effort to maximize the benefit of the alternative and wind up in a situation that we just don't believe, you know, we just don't support? And but that's where we are.

MEMBER REMPE: And so Table 2, it lists the price per cask assumed and just points as a comment, look at the EPRI study is a good place if you're updating Table 2 to provide this footnote or this comment and to say that there's a lot of uncertainty.

And I know I'm harping on Table 2 updates, but it sure seems like a good place where you're listing

your assumptions to have additional caveats.

CHAIRMAN ARMIJO: Well, you know, the staff has finished their report. And, you know, if the Committee wants to make some points -

MEMBER REMPE: They're going to have to issue a correction for Table 2.

CHAIRMAN ARMIJO: Yeah, certainly corrections. The staff would do that without -

MEMBER RAY: On that point, Sam, I think at another time we're going to have to discuss to what extent we have been going from the cost-benefit. That's normally not something - we engage in the benefit side, but the cost side we normally don't do.

But on the other hand if you're talking about cost-benefit of necessity, one of the factors is cost and I would think we could observe that there's no basis for the assumption that was made here.

CHAIRMAN ARMIJO: Weaknesses, yeah.

MR. WITT: One point on this slide that I think is important to bring up is that the top bullet here which talks about the costs outweighing the benefits, that's for all the cases that we evaluated, base case.

All the groups that we evaluated were the costs outweigh the benefits utilizing the current

regulatory policies and guidance that we have in place.

Now, when we did the sensitivity analyses where we up these factors like the dollar per person-rem and the consequences beyond 50 miles, that's when you start to get into some cases where the benefits may outweigh the costs.

And I would think that if the Commission decides to tell us that based on these sensitivities that they want us to do additional research, I would hope that we get additional guidance on how we should consider these things like consequences beyond 50 miles, because they are not a part of our current regulatory policies where we make regulatory decisions on those.

CHAIRMAN ARMIJO: But if you for that case, a sensitivity analysis where you increase the dollars per person-rem, because that's likely the direction where you're going, everything costs more, so everybody understands that, and beyond 50 miles you create a much larger population, but then you admit that the dose that these guys get is because they return consistent with habitability criteria which are deemed safe enough by whatever regulatory authority exists and you still - but you still put in the cost, dollars per person-rem, for a large number of people that are presumed safe. Safe enough.

So, if you took out all those costs, would you still be in the situation? I don't think so. I think you would have no benefit. So, you can't have it both ways.

I think you can't charge $4,000 per person-rem for a huge population for a situation in which they've been allowed to return based on a judgment that it's safe.

So, somewhere along the line I think that there isn't a base case benefit even for the sensitivity analysis.

MEMBER RICCARDELLA: Could I ask a question?

In your judgement if you credited mitigation equally to both options, either you credit it in both cases, or you don't credit it in both cases, would that conclusion about the marginally outweighing the costs, would that change?

CHAIRMAN ARMIJO: Yeah.

MR. SCHOFER: Well, there's enough of them, in fact, to get - yeah, probably if you don't credit it for either case, then I would guess that that margin - the marginal benefit would go away and it would be non-cost beneficial for all of them.

MR. JONES: If you looked at the case where mitigation was effective in both cases, yeah, definitely

the costs would -

CHAIRMAN ARMIJO: Disappear.

MR. JONES: I mean, the costs would far outweigh any potential kind of -

CHAIRMAN ARMIJO: There would essentially be no -

MEMBER BALLINGER: Is that not a more internally consistent way to do things?

MEMBER RICCARDELLA: I think, you know, a lot of the comments from the Committee have to do with these apples and oranges comparisons.

I mean, the real benefit of these types of probabilistic analysis is not the absolute. It's the relative.

And when you make inconsistent assumptions, you're biasing that relative benefit.

MR. RECKLEY: This is Bill Reckley again.

And we're doing that on purpose.

(Laughter.)

MR. RECKLEY: But, again, I'm going to just ask everyone to come back to what the staff is asking the Commission to decide in this particular case and we look much more targeted in terms of the decision that we're asking them to make, which is whether to direct us to go do more study of this issue, or whether they

agree with us that more study would likely show what we think we're already demonstrating here that the combination of the safety test, the risk test and the cost-benefit would show that we are unlikely to proceed to develop a rule to require this particular action.

So, that it really comes down to all we're asking the Commission to decide is tell us to either do more study, in which case as Steve was mentioning we're going to go in and we're going to revise the conservative assumptions, and if there's only group that might show cost beneficial, we'll focus to make sure we're not being overly conservative for that particular group, to reach that point of whether to go into rulemaking or not.

And so, yes, we were inconsistent. We look at it from a couple points of view, right? Every time we make a conservative assumption, we were also thinking if we don't make a conservative assumption, somebody else will say you didn't make a conservative assumption and, therefore, biased it in the other direction, which was what we primarily wanted to avoid.

And so, yes, we were by and large when we faced the choice of A or B, we would pick the one that biased it towards being beneficial to move.

(Simultaneous speaking.)

CHAIRMAN ARMIJO: That's why we have a probability. If you just say, hey, I'll apply the same assumptions to both alternatives across the board, you wind up with, in certain cases, one alternative on its own will be - will have a benefit that the other one doesn't have even though they've all been -

MR. RECKLEY: I understand.

CHAIRMAN ARMIJO: And so, I think you're making the Commission's job a little bit harder than it needs to be with call it a bias or a tilt to - that exaggerates the benefits of Alternative 2 where at least I don't see them.

MR. RECKLEY: And the unintended consequence of doing it the way we did it.

The other thing I'd point out is whenever we're talking about the cost-benefits here and even in the cases where we would say with the sensitivities some of them are marginally cost effective, is never consider that in isolation from the first test of the actual risk reduction against the QHOs which still would not pass.

And so, the way we do regulations it's not just - it's not one way or the other. It has to be both. And you have to get by the QHO test first.

And so, this is just a little additional information for the decision-makers, but not to focus

too heavily on the fact that there's a marginal cost benefit in some of the calculations.

We can always come up with cases where there's marginal cost benefits. We don't pursue them, because they don't have the corresponding safety benefit to warrant doing the ruling.

MEMBER RAY: But, Bill, you normally aren't dealing with the kind of issue that this -

MR. RECKLEY: No, I understand the politics of it.

MEMBER RAY: All right. Then you should be sympathetic to the comments.

MR. RECKLEY: No, I do. I do. Then, again, I say that might be an unintended consequence of the way we chose to do this, but -

CHAIRMAN ARMIJO: Okay, let's keep going - oh, I did have a question on Table 3, and that is replacement energy cost. And I have to confess I didn't go into the details of how it was calculated, but you project - is this years into the future or - and how do you come up with the replacement cost?

Probably Harold can answer it for me, but seems like there's so much uncertainty on that. That could be a huge variable whether it was now low-cost gas or high-cost windmills or something like that.

MR. SCHOFER: This is Fred Schofer.

We actually had a study performed for us where we looked at all the generating facilities within the US as well as the forecast for new generation and retirement of facilities over ten years to, you know, model each region.

And then looked at economic dispatch for each of the regions within the country to come up with values both at the hourly, daily and annual rates. That's how we came up with that.

That is part of the - as we're revising our reg analysis guidelines, we wanted to update the cost of replacement power, because a lot has changed since the original reports came out particularly with emergent plans and deregulation. So, that's why we went to a dispatch model.

MR. JONES: And I think it's important to note this didn't really have a - play a role in this particular analysis, because we're not talking about -

MR. SCHOFER: And of course included in that is a forecast for natural gas and solar and hydro and so forth.

CHAIRMAN ARMIJO: All right. Thank you.

MR. JONES: Just a few more slides to go

through. I wanted to put this in a safety perspective, I guess.

Basically, confirm the pools provide adequate protection and we feel they have a substantial defense-in-depth.

The overall frequency of damage to spent fuel even with these relatively low or conservative assumptions is just a few times in a million years. And these frequencies exclude the effective deployment of mitigation, which is the subject of past regulatory action and orders that are in place to expand that mitigation capability.

We think the spent fuel pool has defense-in-depth, because there are several layers involved here.

Predominantly the pool itself was so robust that, you know, we're getting into pretty extreme earthquakes to even have any substantial damage. So, variable frequency of an initiator requires any mitigation whatsoever.

The ones that do require mitigation, we have capabilities now especially with these new orders that provides good mitigation capability to address those situations. That's all I have for that.

I want to acknowledge that there is some

issues with the use of quantitative health objectives for screening. As we mentioned earlier, they were developed or intended for reactor accidents. And we recognize that the spent fuel pool accidents could affect large areas and populations.

And while we could develop alternative societal measures, we feel we have appropriate safety levels based on the defense-in-depth and the ability to meet the quantitative health objectives.

CHAIRMAN ARMIJO: What is it that would make you feel that the QHOs aren't suitable for spent fuel pool accidents? I mean, it's -

MR. JONES: Well, they're an individual risk measure. That's the predominant problem. It doesn't integrate, I guess, all the effects that could go into population.

That's one of the main reasons I guess to go onto a cost-benefit analysis is that it does more fully capture all the impacts, because we are, for instance, the evacuations change the health effects, but it costs money to move people around.

So, when you factor that into the cost-benefit analysis, I think you see a little bit - a fuller picture of the societal impacts that are also part of the -

CHAIRMAN ARMIJO: So, that's an added benefit of the regulatory analysis. It fills that gap in some way.

MR. JONES: Somewhat, right.

CHAIRMAN ARMIJO: Yeah.

MEMBER SCHULTZ: At the same time when you did the evaluation for the QHO calculations, and I'm expecting this is done routinely, very conservative assumptions were made associated with the frequency of the event and the fragility of the pool.

It was, you know, essentially we're going to have an event and if the pool fails, then let's see what the consequences would be.

And the highest, not the mean and not the base case, but the highest frequency was chosen to represent the conclusion that the QHO was met.

So, yes, they were used, but - and they may not be particularly applicable in the case of spent fuel pool, but they were evaluated in the high case with conservative assumptions and so forth.

So, you can suggest that in that particular approach, you've taken care of a concern that we shouldn't be using these for the evaluation or some, you know, something else could certainly be better, but -

MR. WITT: Well, what we said is that we think that the QHOs are appropriate, because we feel confident with the measures that we have in place for spent fuel pools, but there may be some issues that people have with the way we did this. And we acknowledge those issues.

MEMBER SCHULTZ: If we were to move forward and look further at approach that would be applicable to spent fuel pool, it seems to open up the book associated with as we've talked about here.

Another level of technical evaluation that I think we would be prepared for, but it would get into issues like application of LNT and - versus thresholds associated with application of dose impact and so forth. And that's just one thing.

There are the other things that would certainly come into evaluation also.

MR. WITT: And another thing with all these issues is that there are much broader policy issues than just this one specific aspect of spent fuel pools.

I think it applies to all the regulatory activities that the NRC conducts.

MR. WITT: And other agencies as well in protecting the public.

CHAIRMAN ARMIJO: I had - are you finished

with your slide, Steve? Other alternatives. Okay, there you are. That's the one I anted to get to.

MR. JONES: All right. There were other alternatives. Alternative loading patterns, for example, going to a less dense configuration than even the current regulations or requirements called for.

However, there are limits to that as far as available pool storage space. Not all plants can get to much lower density configurations or - I'm sorry, fuller distribution of the hot fuel among the colder fuel due to limits on the storage space.

Direct offload of fuel into more - into the required patterns is also an area that might potentially have benefits, but there is effects then on reactor conditions during refueling particularly.

And then, finally, enhancement of mitigation strategies, we think the existing mitigation orders have established quite a robust capability to provide mitigation for spent fuel pool accidents and not much areas for further improvement there.

So, overall we considered these changes, but determined that they wouldn't for the same reasons, really, as the expedited transfer of fuel, wouldn't provide a substantial safety enhancement such that further study and regulatory action would be warranted.

CHAIRMAN ARMIJO: In the spent fuel pool study, the Peach Bottom loading pattern of the one by eight -

MR. JONES: Right.

CHAIRMAN ARMIJO: - seemed to have a really powerful effect in reducing the likelihood of getting into a problem. And, you know, certainly if it could be done, would be very low cost compared to cask loading and expedited and things like that.

It's something that could - and maybe it's only limited to BWR 4's spent fuel pools. I don't know. But in the cost-benefit analysis, that would come out very favorable, I suspect, if it worked for maybe just only one set, one group of pools.

Did you look any further than that? You know, the cost of that can't be very much. Of course nothing is cheap in this business, but -

MR. JONES: We didn't do a strip cost-benefit analysis. However, there are several pools that can't quite reach that due to absolute limits on the storage capacity, as I mentioned.

Other things come into play with regard to how the existing storage locations are used. The technical specifications may include, you know, storage locations that aren't suitable for fuel or that are

needed for other equipment storage that's not really, you know, addressed directly, but, you know, highly activated components are routinely stored in the spent fuel pool and they take up some of the space that might otherwise be available for fuel storage.

CHAIRMAN ARMIJO: But if you weren't limited by those kinds of concerns, wouldn't that be just something of good practice? It's really low probability, but it doesn't cost very much.

MR. JONES: I think it's something that I think we have addressed.

MR. WITT: Yeah, we did have that in the COMSECY. We did identify to the Commission that we would communicate this to the industry and they could pursue it.

There may be benefits, but we just don't feel that this would pass our regulatory thresholds in terms of a substantial safety enhancement to warrant regulatory action.

So, we thought the best step was to identify this to the industry and let them figure out if they want to do it or not.

CHAIRMAN ARMIJO: Okay. Next slide.

MR. JONES: Conclusions. Okay. Just real briefly again the safety goal screening using the

quantitative health objectives determined that the potential benefits were a very small fraction of the safety goals. And, therefore, it wouldn't be justified to continue with the regulatory action.

And the cost-benefit analysis determined that there's only, if any - for the most part, the costs outweigh the benefits. But if there are any cases where the benefits outweigh the costs, they're very marginal and might be overcome by, as we mentioned, changes in the assumptions and other factors not fully evaluated yet, but we think that will change based on further, more detailed analysis of some of our input assumptions.

Based on the generic assessment of the cost-benefit analysis, we don't feel that additional studies are necessary to evaluate spent fuel pool transfer and the potential added risks involved with that storage situation in addition to further refining some of the input assumptions we've already considered here. And, therefore, we also recommend no further regulatory action on this issue.

And that's it.

MR. WITT: We do have this slide on upcoming activities. As I'm sure you are aware, there is an ACRS full committee meeting.

CHAIRMAN ARMIJO: We're aware of that.

(Laughter.)

MR. WITT: And the Commission originally planned to have a meeting on this next week, but that did get postponed to early January. So, that will be happening in early January.

CHAIRMAN ARMIJO: Excellent. With that, if you're finished, let's go to comments from the committee members.

Joy.

MEMBER REMPE: They've done a lot of work, but I have brought up a couple of places where Table 2 needs to be updated, assumptions, I think, should be more clearly documented and John's comments about the tables where values seem to be inconsistent.

And so, clearly by the December meeting I hope some of those issues are identified or clarified for us.

And I don't know how to deal with some of these other issues that it's already gone to the Commission and if there's an errata sheet or I don't know how that can be resolved.

CHAIRMAN ARMIJO: I'm sure the staff has ways to say, hey, look, there's an error here. But if those things were sorted out before our December meeting, which would, you know, they're solved, we don't have

to comment on that.

MR. WITT: Yeah, we'll definitely investigate how to pursue that.

MEMBER REMPE: If we could see the corrected tables before our December meeting, it would be nice so we're not just trying to think on the fly at the meeting. That would be helpful.

MR. WITT: Sure.

MEMBER STETKAR: That would be next week, and Thanksgiving is next Thursday. Just keep that in mind.

(Laughter.)

(Discussion off the record.)

MEMBER STETKAR: Perspective on time is important often.

MEMBER REMPE: Yeah.

CHAIRMAN ARMIJO: Yeah, we can forget.

All right, Charlie.

MEMBER BROWN: Since I'm not an expert on all these fragilities and other goodies that you all were tossing around, I did enjoy and learned a lot from a bunch of the questions you all did ask relative to this on the diversity of assumptions and the bias that we use to come to the conclusions.

And I really rogered up mostly to, I guess,

Pete's comment about seeing a set of the two alternatives, you know, do you do it or do you not do it, evaluated with the same assumptions with no biases in there, the same, you know, relative uncertainties and stuff being the same and then doing your variations.

So, if I was a decision-maker, that's what - the way I thought about it. That's what I would have liked to have seen. In whole, I thought that was an excellent comment relative to it's hard to tell with all these different biases thrown in.

I understand the basis for what you all were doing and what you were aiming at, but I think that, to me, that was the missing piece. That sounds like the train has left the station and the piece of paper is out, so - but that's what I reckon I'll pass on to Mike.

MEMBER RYAN: I don't have anything else to add to the comments made already. Thank you.

CHAIRMAN ARMIJO: John.

MEMBER STETKAR: Couple of things. First of all, I like the fact that you've kind of pulled the QHO stuff and the safety perspective into, you know, a little bit better focus in this version compared to the draft that we saw. I think that helps a lot.

It's short and sweet, but it at least draws

people's attention to it. And I guess I don't have anything else.

I do understand, you know, my kind of engineering hat notwithstanding, I do understand the reasons that the staff made the assumptions to try to bias things toward the alternative. I understand that as an engineer. I don't like to work that way, but I get it.

Whether the report makes it so glaringly obvious that the deck was stacked that way, I think, remains to be seen when you have people who are not involved in this long drawn out process, you know, reading this report.

And I don't think the staff can read it objectively, because you know what you did. And, quite honestly, we've been involved with it long enough that we can't read it with a fresh set of eyes either. And I think I'll just leave my comments at that.

CHAIRMAN ARMIJO: Okay, Steve.

MEMBER STETKAR: We've had a good discussion here today. I did want to take this time to remark about how well I feel the work that has been done in this area over the past few years including, of course, the spent fuel pool study being developed and then used as the basis for a lot of the work in this evaluation, how well

that work has been done, how much effort has gone into it, how much engineering evaluation has been done.

I was - and with regard to that, I was a little disappointed in the evaluation that was done here, in fact, going back and pulling in work that was done, 15, 20, 25 years ago to try to assure that conclusions that were reached way back when, when we didn't have the analysis capability that we do today, that even those evaluations and analyses and results were kind of pulled in to make sure everything was covered and that we used conservative assumptions to bound things that were concluded many, many years ago.

I'm not sure that was - from one perspective, it was a good thing to do. But in terms of doing a strict and detailed engineering evaluation, one could have used what was developed for the spent fuel pool study.

And, again, if you had done that, you would have demonstrated even more clearly that there's no differentiation between Alternative 1 and 2.

With regard to the discussions we've had today as we said in several different ways, making assumptions about what will be applicable to Alternative 1 versus Alternative 2 based upon different engineering assumptions, if you will call it that, which are not,

in fact, practically real, creates a situation wherein the results may be taken improperly out of context.

And as we've said, one may attribute to Alternative 1 that you can't mitigate the accident. Where, in fact, as we discussed, either event can be mitigated.

So, I think going forward we need to continue to realize what assumptions we've made here for the purposes of performing the analysis we've done, but we need to - we need to retain the knowledge that we made those assumptions for a particular purpose that is documented here. And those assumptions should not be taken forward to presume that the risk associated with spent fuel pool accidents is documented in this study.

It's documented in the spent fuel pool study very nicely. And when you rack up all of the conservatisms that are in that study and take them out, you see an extremely low likelihood of event and extremely low consequence.

It's almost difficult to describe associated with the risk of spent fuel pool accident and consequence.

CHAIRMAN ARMIJO: Dick.

MEMBER SKILLMAN: Thank you for your

presentation. I would like to suggest that what you've included in your regulatory analysis in Section C3, implementation assumptions, deserves some accreditation in your conclusions specifically what it takes for plant staff to move all of that fuel, the risks associated with dropping a fuel assembly versus dropping a cask or both.

Those are real risks. And if one were to simply glance at this study and say, well, the cost benefit shows that it really is not beneficial to do all these moves, it seems the Commissioners ought to realize if they were to move to want early transfer, there is a whole other side of risk associated with it. It seems to me that that's mighty important, because of people like us that do this work.

The second thing I'll let Harold touch on as he wishes to with the issue of the number of casks and what it would take to obtain all of those casks, but those two items are the objective of your C3 writeup. And it seems like a piece of each of those should be flagged in your conclusions. Thank you.

CHAIRMAN ARMIJO: Okay, Harold.

MEMBER RAY: Okay. I just want to say that I think that we are over - we may be overrelying on the idea that the quantitative health objectives are the

principal criterion for something of this kind.

And that because others may not see that as clearly, the fact - and for reasons Steve himself mentioned, the claims benefit is relevant and not just something that can be easily dismissed, because the QHOs are what they are.

We've all said it, I've said it, there isn't alternative but to, I guess, reiterate the importance now of making clear that the cost benefit was what it is, but I'll just say I'm concerned - the principal concern is that we are too prepared to say, oh, well, look at the QHOs. That's what you've got to do first. And if that isn't met, then the rest doesn't matter. And in any case, it's, in our judgment, doesn't justify doing anything further.

I don't think it's fair to ask others to look at things that way. And for that reason, how the cost benefit is presented becomes, I think, more important than we've treated it here probably inadvertently, but for other reasons, I mean, than that it was definitive or dispositive of what should be done.

But it is what it is, like I say. And, therefore, I'm concerned going forward that it will assume a life that it wasn't intended to have, but that there won't be anything then we can do about it and it

will just have to be worked through on the basis that people are inclined to believe that there is sufficient benefit to warrant the cost. And that would be a very unfortunate conclusion. That's it.

CHAIRMAN ARMIJO: Ron.

MEMBER BALLINGER: I'd like to just go over what Steve has said. There's an awful lot of work that's been done.

With respect to this document, though, I'm concerned that the original goal was to sort of find out or demonstrate that there was no benefit.

I think what we may have done is to just show the opposite in the sense that he said no further work needs to be done.

I think maybe what the sort of cherry-picking of assumptions, if you want to use it that way, different things, what we've really done is to make an argument for further work.

And in this environment that we live in, I guess the horse is out of the barn, the report is issued and all that stuff, but I just don't think it's going to end there. And so, I don't think we've heard the last of it. I think we're going to - we have not.

CHAIRMAN ARMIJO: Okay, Dr. Banerjee.

MEMBER BANERJEE: Sam, are we going to write

a letter on this?

CHAIRMAN ARMIJO: I can't help it. Absolutely, we are going to have to write a letter. I think we have an SRM that -

MEMBER BANERJEE: Well, we are going to -

CHAIRMAN ARMIJO: Typically, we would write a letter.

MEMBER BANERJEE: We need to, I guess, separate the spent fuel pool study from this study, right?

CHAIRMAN ARMIJO: Sure.

MEMBER BANERJEE: So, in some ways I have the same concerns as Ron has. I don't think that this has come to an end here.

And part of it is because there's enough grounds to demand more work be done to quantify the benefits. And I can understand as John said, why you went about doing things the way you did, which was to give the best shot to, you know, moving fuel into casks and quantifying those benefits as large as they seemed feasible in some way.

But in doing that, I think you've opened up some issues which we probably need to get addressed. So, we really need to think about how to address this.

CHAIRMAN ARMIJO: Yeah, in our letter.

MEMBER BANERJEE: Yes, this is really -

CHAIRMAN ARMIJO: It will be up to the Committee what direction they want to take.

MEMBER BANERJEE: Yeah. So, I think it's going to probably remain an open issue for a while. Anyway, that's my -

CHAIRMAN ARMIJO: Pete.

MEMBER RICCARDELLA: I don't have anything to add that wouldn't be redundant to what my colleagues have already said.

CHAIRMAN ARMIJO: Okay. Well, on behalf of the Subcommittee, first of all, we may have given you a very hard time, but that's just -

PARTICIPANT: I'm sure it was intentional.

(Laughter.)

CHAIRMAN ARMIJO: And I'm sure you enjoyed it, but I'd like to say you've done a ton of work. We know it was on a tough time schedule and this new version is a definite improvement over the draft that we saw earlier. There are still some matters.

And before I open it up for public comment, I want to just summarize just two points that I had that kind of a little bit repeats what everybody else has said.

The philosophy of trying to maximize the

benefits of Alternative 2, you know, I guess I see some art, some value in that, but I think what really the analysis philosophy should be, to show what is the inherent advantage of the alternative. What's built into it?

And when you start dicing it with assumptions that favor one advantage, one alternative versus the other, you lose the inherent advantage of the option.

And there could be some situations where Alternative 2 has built in without any crutches or favors of built in advantage, but I don't think it's there, but it could be.

So, I kept looking for analyses that show inherent advantage, and that's when - the way you get that is by just same assumptions for either case where it's appropriate.

Then the other point I made is in the sensitivity studies and even in the basic studies, I just don't see the - how one can count the person-rem costs of - that are accumulated by people who are living in an area that meets habitability criteria. I don't think you can have it both ways.

If there's a person-rem cost that you really believe should be maybe people shouldn't be living

there, something is wrong here. You can't have it both ways. So, anyway, I leave it at that.

I should have probably earlier asked for a comment from members of the public. I know Dr. Lyman was going to be on the bridge line. I hope he's still on.

Maybe we could open up the bridge line and see if he cares to make some comments. And is there anyone in the room that would like to make a comment?

There's no one in the room. So -

MR. LYMAN: Yes, I am on the bridge line.

Can you hear me?

CHAIRMAN ARMIJO: Okay. Yes, please go ahead.

MR. LYMAN: Thanks. Well, we've already gone on the record with regard to our views on this issue. I'm not going to go over that.

But I would like to address the issue that Dr. Armijo has raised repeatedly, and that's the issue of accounting for the dose for people who return to their home.

And the confusion here is that the habitability criteria are based on limiting individual risks. So, what you're talking about here is an assessment of the societal risk, the cumulative risk.

So, even though individual decisions to return are based on an incremental individual risk which society judges to be acceptable on an individual basis, that the cumulative impact of all those people who are allowed to return because their individual risks are limited, that that cumulative impact could be considerable and that's what you're evaluating here.

So, there is a mismatch between the criteria that are used to determine habitability and those that would be used to evaluate whether the cumulative societal impact is unacceptable. So, that's the issue.

And putting aside whether there's a threshold or not, because I - well, I think, I mean, there is no - well, I don't want to get into that argument.

(Laughter.)

CHAIRMAN ARMIJO: Yeah, look, I appreciate that point. And I got to think about it some more, but I appreciate you bringing it up.

MR. LYMAN: Yeah. And you have to look at the societal impact of what has happened with a much smaller cesium release from Fukushima and the decisions that those people have to make, because the Government judges if they can return to their homes, it's safe for them to return.

Plus the dose, you know, if you're talking about a 20 millisievert per year or something, that is a considerable additional risk of dose background.

So, they have to make these decisions based on whether to return to their home if they're not going to be compensated, because the Government says it's safe. And so, that's an additional layer that's just not present.

The other thing I wanted to say was if you're going to start looking at things like the - a factor of two difference in the cross-side of the equation, then you need to look at uncertainties that lead to significant differences and benefits as well.

And there are assumptions built in, I'd be happy to show them to the Committee, that would lead to a factor of ten or more increase in benefits depending on additional sensitivities other than what the staff has looked at.

So, there are additional uncertainties of both sides, and I'll stop there. Thank you.

CHAIRMAN ARMIJO: Okay. Thank you, Dr. Lyman. I think unless we have any other comments - staff, nothing more? We've certainly made our comments.

Again, thank you very much. We're about

ten minutes behind schedule, but good job. We're adjourned.

(Whereupon, at 12:10 o'clock p.m. the meeting was adjourned.)

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Japan Lessons Learned Tier 3 Issue: Expedited Transfer of Spent Fuel to Dry Cask Storage

Kevin Witt, NRR/JLD/PSB
Steven Jones, NRR/DSS/SBPB
Fred Schofer, NRR/DPR/PRMB

ACRS Subcommittee Briefing
November 19, 2013

Agenda

- Background
- Tier 3 Evaluation Process
- Regulatory Analysis Modeling, Assumptions, and Results
- Upcoming Meetings

Background

- Tier 3 Project Plan:
  - Determine whether the NRC should consider expedited transfer of spent fuel to dry casks
    - » Follows normal regulatory process utilizing Regulatory Analysis Guidelines (NUREG/BR-0058)
    - » Utilizes information from past SFP evaluations and the SFPS
  - Current phase evaluates whether additional studies are necessary to determine if regulatory action might be warranted
    - » Conservative analysis that maximizes calculated benefits of expedited transfer (i.e., not best-estimate risk study)
    - » Provides information for decision regarding further research
  - May 2013 Memo provided updated plan to Commission

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Stakeholder Interactions

- Two public meetings held (August 22 and September 18)
  - Questions involving both SFPS and Expedited Fuel Transfer
  - Responding to letters received from stakeholders
- Spent Fuel Pool Study
  - Draft issued for public comment - June 2013
  - Written comments addressed in final report - October 2013
- Expedited Transfer Memorandum and Regulatory Analysis
  - Draft issued for public review - September 2013
  - ACRS Presentation – October 2013
  - Non-concurrence from NRC staff
  - In response to stakeholder feedback, the staff provided additional detail addressing specific issues and reformatted analysis for clarity

Overview

Generic Regulatory Analysis

- Regulatory Assessment
- Expanded Plants (Generic by Groups)
- Expanded Scenarios

Regulatory Analysis for Reference Plant (Appendix D)

- Regulatory Assessment
- Specific Plant
- Expanded Scenarios

Spent Fuel Pool Study

- Consequence Study
- Specific Plant
- Specific Scenario

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Generic Regulatory Analysis

- **Spent Fuel Pool Study (Appendix D) and Tier 3 Regulatory Analysis consider initiating events beyond the event in SFPS:**
  - more severe earthquake
  - cask drop
  - loss of power/loss of coolant inventory events
- **Tier 3 Regulatory Analysis covers all SFP designs used with operating reactors in the Eastern and Central U.S.**
  - PWRs and BWRs with Mark III containments (spent fuel stored in at-grade pool separate from reactor building)
  - Western plants to be revisited following seismic re-evaluations
  - new reactors (AP-1000)
- **Assessment of security events handled separately**
  - regulatory changes implemented (e.g., 10 CFR 50.54(hh))
  - effect of security changes reflected in regulatory baseline

Tier 3 Evaluation Process

- Safety Goal Screening Evaluation
  - Based on the Commission Safety Goal Policy Statement
  - Used the Quantitative Health Objectives to evaluate achievement of the safety goals
- Cost/Benefit Analysis
  - Intended to identify maximum potential benefit
  - Analyzes costs and benefits for representative pool design groups
- Sensitivity Studies
  - Evaluates key factors to illustrate their effect on the final result

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Safety Goal Screening

Bounding Release Frequency

- Bounding frequency of SFP release about 1 in 29,000 years (3.46×10^{-5} per year)
- Regulatory Analysis Table 43, High Estimate for Group 4 (highest total release frequency)

Conditional Probability of Fatal Cancer

- Conditional probability of an individual developing a fatal latent cancer within a ten-mile radius calculated to be 4.4×10^{-4} given a large SFP release from high-density pool (SFPS Table 34)
- Linear – no-threshold model with protective actions implemented

Individual Latent Cancer Fatality Risk

- Conservative latent cancer fatality risk estimate to an average individual within ten miles of 1 in 66 million (1.52×10^{-8} per year)
- Less than one percent of the individual risk goal of less than one-tenth of one percent of the average chance of developing a fatal cancer in the U.S. (2×10^{-6} per year)

Safety Goal Screening Results

- Marginal safety benefit based on comparison with QHOs
  - No risk of fatalities due to nature of release
  - Potential benefit is a very small fraction (0.76%) of latent cancer goal
  - Cancer risk relatively insensitive to magnitude of release due to slow accident progression and effective protective actions (SFPS)
- Proceeded to cost/benefit analysis even though process allows stopping when evaluation shows safety benefit below threshold of safety goal screening

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Cost-Benefit Analysis Overview

- Evaluated one alternative - Expedited Transfer
  - Transfer fuel with more than 5 years decay to dry casks
  - Store remaining fuel in low-density configuration in existing racks
- Established Seven SFP Groups
  - Three groups not evaluated due to low risk
  - Four groups evaluated representing operating and new plants
- Major Assumptions (Regulatory Analysis Table 2)
  - Initiating SFP Event Frequencies and Accident Progression
  - Economic modeling (e.g., definition of representative plants, future spent fuel discharge projections, etc.)
  - Timing (e.g., dry cask storage loading, occupational dose, etc.)
- Established a base case
- Performed sensitivity studies

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Assumptions to Maximize Calculated Benefit

- Release fraction and mitigation effectiveness assumptions provide conservative estimate of potential benefit
- Regulatory Baseline – Maintain the Existing Spent Fuel Storage Requirements
  - High cesium release fractions (SFPS value of ~40% for Elevated Pools and NUREG-1738 value of 75% for other groups in base case)
  - Ineffective mitigation (all fuel heat-up events lead to large release)
- Expedited Transfer Alternative - Low-density Spent Fuel Pool Storage
  - Low cesium release fractions (SFPS value of 3% for all groups in base case)
  - Effective mitigation (19 of 20 fuel heat-up events result in no release due to effective mitigation)

Base Case Analysis

- Staff considers base case appropriate for decision whether to pursue additional studies to refine assumptions
- Base case includes appropriately conservative assumptions, but not bounding values, for the following:
  - Initiating Events (USGS 2008 information for Peach Bottom seismic hazard, and NUREG-1738 and NUREG-1353 for other initiators)
  - Seismic liner fragilities (based on results of SFPS and NUREG-1738)
  - Cesium inventories for each group (based on SFP capacity, reactor power, and fuel burnup for reactors in group)
  - Plume dispersion (uses MAACS2 and Peach Bottom Meteorology)
  - Population density and economic activity (used data for Surry)
  - Industry implementation costs (EPRI information modified for representative site)

Base Case Analysis (Continued)

- Uncertainty regarding spent fuel pool conditions (i.e., pool water level, fuel distribution, and location of liner tears)
  - Generally make bounding assumption of inadequate heat removal if fuel is uncovered for base case
  - Results in dominant initiating events progressing to fuel heat-up
  - Conservative because SFPS and other studies indicate substantial potential for air cooling when pool is drained or decay heat is low
  - Exception for Mark I and II BWRs
    - SFPS reduces uncertainty for specific scenario evaluated
    - Used SFPS information of 8% inadequate cooling for 0.7g PGA quake

Effect of Assumptions

- Seismic event frequencies
  - Peach Bottom frequencies used, which falls close to the upper end of all sites evaluated
  - Lower than bounding site (Sequoyah) by factor of ~3.4
- Population Demographics
  - Surry population demographics used
  - About mean population density (above median) of all plant sites evaluated
  - Use of 90th percentile demographics would increase benefits within 50 miles by about 28 percent
- Other assumptions have smaller impacts

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Base Case Frequencies

Event	Base Case Frequency	Pool Liner Fragility	Inadequate Cooling	Fuel Heat-up Frequency	Comments
Seismic Bin 3 (0.7g PGA) Elevated Pool At-Grade Pool	Peach Bottom $1.65x10^{-5}$ $1.65x10^{-5}$	 10% 5%	 8% 100%	 $1.35x10^{-7}$ $8.25x10^{-7}$	 SFPS result
Seismic Bin 4 (1.2g PGA) Elevated Pool At-Grade Pool	Peach Bottom $4.90x10^{-6}$ $4.90x10^{-6}$	 100% 50%	 100% 100%	 $4.90x10^{-6}$ $2.45x10^{-6}$	
Cask Drop All Pools	 $2.0x10^{-7}$	 100%	 100%	 $2.0x10^{-7}$	Not always credible
Other Initiators Elevated Pool At-Grade Pool	 $2.37x10^{-7}$ $2.67x10^{-7}$	 Not Applicable	 100% 100%	 $2.37x10^{-7}$ $2.67x10^{-7}$	
Total Elevated Pool At-Grade Pool				 $5.47x10^{-6}$ $3.74x10^{-6}$	About 90% seismic contribution

Cost-Benefit Analysis Results

- Base case costs outweigh benefits
  - Benefits based on $2000/person-rem within 50 miles
  - Changes in discount rate do not change result
- Sensitivity Analyses ($4000/person-rem and analysis beyond 50 miles) produce marginal benefits
  - Base case costs outweigh benefits for Groups 1 & 2
  - Base case benefits marginally outweigh costs for Groups 3 & 4
- The staff considers the base case an appropriately conservative analysis for use as the primary basis for the staff's recommendation that additional studies not be pursued and Tier 3 issue be closed.

Safety Perspectives

- Pools provide adequate protection and defense-in-depth
- Overall estimated frequency of damage to stored fuel is low
  - Base case release frequencies for existing pools are on the order of a few times in a million years
  - These frequencies exclude effective deployment of mitigation capability and generally exclude consideration of air cooling (SFPS)
- Spent Fuel Pool Maintains Defense-in-Depth
  - Defense-in-depth consists of layers of protection with reliability of each layer commensurate with the frequency of challenges
  - SFP designed to prevent coolant inventory loss under accident conditions, which results in a low frequency of coolant inventory loss
  - Fuel dispersal, coolant makeup, and spray capability have reliability commensurate with the low frequency of coolant inventory loss

Use of QHOs for Screening

- Acknowledge that current safety goal screening, including QHOs, developed for reactor accidents
- Recognize that SFP accidents could result in larger affected areas and populations
- Could develop alternate societal measures but with continued focus on public health and safety (SRM for SECY-12-0110)

Other Alternatives

- Examples include:
  - Alternative loading patterns
  - Direct offload of fuel into more coolable patterns
  - Enhancement of mitigation strategies
- Staff has considered these possible changes but determined that they do not provide a substantial safety enhancement such that generic regulatory action would be warranted

Conclusion

- The safety goal screening evaluation concludes that SFP accidents contributes less than 1% to the overall risks for public health and safety. Enhancements to SFP designs or operations or would therefore provide only minor or limited safety benefit.

- The staff conducted a cost-benefit analysis, which finds that the added costs involved with expedited transfer of spent fuel to dry cask storage to achieve the low-density SFP storage alternative are not warranted in light of the marginal safety benefits from such an action.

Conclusion (Continued)

- Based on the generic assessment and the other considerations detailed in this paper, the staff finds that additional studies are not needed to reasonably conclude that the expedited transfer of spent fuel to dry cask storage would provide only a marginal increase in the overall protection of public health and safety, and would not be warranted due to the expected implementation costs

- **No further regulatory action is recommended for the resolution of this issue**

Upcoming Meetings

- Final COMSECY-13-0030
  - Signed November 12, 2013
- ACRS Meeting
  - Full Committee in December
- Commission Meeting on Spent Fuel Safety
  - January 2014

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Backup Slides

Groups

1. BWR Mark I / II with non-shared spent fuel pool (SFP) located well above grade (Excluding Western U.S. Reactor - Columbia)
2. PWR & BWR Mark III with non-shared SFP located at grade with at least one exposed side (Excluding Western U.S. Reactors – Diablo Canyon and Palo Verde)
3. Combined Operating License Holder SFPs (AP-1000)
4. PWRs with Shared SFPs
5. SFPs located below grade with backfill on all sides (low probability of inventory loss, but evaluated with Group 2)
6. SFPs at decommissioned plants (fuel in pool) (not evaluated based on low decay heat rate)
7. Sites where fuel is in dry casks

Release Sequence of Events

Initiating Events
- Bin 3 Seismic Event (0.7g PGA)
- Bin 4 Seismic Event (1.2g PGA)
- Cask Drop

AC Power Fragility
- Reflects unreliability of both AC power and installed makeup systems for event mitigation
- Installed makeup systems often inadequate to mitigate significant damage even if AC power available

Liner Fragility
- Reflects likelihood of coolant inventory loss that uncovers the fuel given the initiating event
- Uncertainty in seismic response of pool liner to events well beyond design basis; near certainty for cask drop

No Air Cooling
- Reflects likelihood that air circulation inadequately cools fuel given fuel has been uncovered
- Inadequate cooling could result from: recently discharged fuel, non-dispersed configuration, and partial drainage states

Fuel Heat-up
- Product of values of above items plus other initiators that involve evaporative loss of coolant inventory
- Values for frequencies of other initiators drawn from prior studies (i.e., NUREG-1738 and NUREG-1353)

Mitigation Failure
- Applied to maximize benefit of expedited transfer
- Reduces frequency of release for low-density storage states by a factor of 19; high density release frequency unaffected

Release
- Combined with estimated consequences of release to obtain risk metrics
- Consistent with past studies, seismic events continue to dominate release frequency

Consequence Analysis

Cesium Inventory

- Calculated for representative plants based on licensed power, licensed inventory and burnup
- Selected from plants within each group
  - Base case used inventory representative of average inventory of plants in group

Release Fraction

- High-density base case uses 40% for Group 1 (SFPS) and 75% for other groups
- Low-density base case uses 3% (SFPS result for low-density unmitigated scenario)

Release Plume

- MAACS2 Model using Peach Bottom Meteorology
- Plume characteristics based on MELCOR release information from SFPS

Health and Economic Effects

- Relocation Based on Protective Action Limits
  - Base Case used 2 rem first year / 500 mrem thereafter
- Linear – No-Threshold Dose-Response Model
- Population Density and Economic Activity based on mean site (Surry) for base case

Accident Progression – Group 1

Parameter	Base Case	High Est.	Notes
Site seismic hazard • Bin 3 (0.7g PGA) • Bin 4 (1.2g PGA)	Peach Bottom 1.65×10^{-5} 4.90×10^{-6}	Limerick 2.24×10^{-5} 7.09×10^{-6}	Limerick is Group 1 site with highest seismic hazard
Liner fragility • Bin 3 (SFPS) • Bin 4 • Cask Drop	 10% 100% (bounding) 100%	 100% (bounding) 100% (bounding) 100%	For high estimate, specified initiators always result in coolant inventory leak
Insufficient nat. circ • Bin 3 • Bin 4 • Cask Drop • Other Initiators	 8% 100% (bounding) 100% (bounding) 100% (bounding)	 100% (bounding) 100% (bounding) 100% (bounding) 100% (bounding)	High est. never air coolable – bounds: • uniform dist. • partial drain • closed cell racks
Release Fraction • Alternative 1 • Alternative 2	 40% 3%	 90% 5%	Alternative 2 models successful mitigation - additional factor of 19 reduction

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Accident Progression – Groups2- 4

Parameter	Base Case	High Est.	Notes
Site seismic hazard • Bin 3 (0.7g PGA) • Bin 4 (1.2g PGA)	Peach Bottom 1.65×10^{-5} 4.90×10^{-6}	[Highest in Group] 2.9×10^{-5} to 5.6×10^{-5} 9.1×10^{-6} to 2.0×10^{-5}	Highest Hazard Sites: Gr. 2: Watts Bar Gr. 3: Summer Gr. 4: Sequoyah
Liner fragility • Bin 3 • Bin 4 • Cask Drop	 5% 50% 100%	 25% 100% (bounding) 100%	Bin 4 Earthquake and cask drop always result in loss of coolant inventory
Insufficient nat. circ • Bin 3 • Bin 4 • Cask Drop • Other Initiators	 100% (bounding) 100% (bounding) 100% (bounding) 100% (bounding)	 100% (bounding) 100% (bounding) 100% (bounding) 100% (bounding)	Base & High case not air coolable – bounds: • uniform dist. • partial drain • closed cell racks
Release Fraction • Alternative 1 • Alternative 2	 75% 3%	 90% 5%	Alternative 2 models successful mitigation - additional factor of 19 reduction

Source Term (MCi Cesium)

Group	Low Est.	Base Case	High Est.
Source term	**Alt 1/Alt 2**	**Alt 1/Alt 2**	**Alt 1/ Alt 2**
Group 1 (BWR)	40.6 / 19.8	52.7 / 22.0	63.3 / 26.4
Group 2 (PWR)	57.4 / 15.7	67.9 / 17.4	78.2 / 20.9
Group 3 (New)	33.7 / 15.7	44.4 / 17.4	54.2 / 20.9
Group 4 (Shared)	63.6 / 31.4	101.1 / 34.8	142.2 / 41.8

Regulatory Analysis Inputs

Parameter	Low Est.	Base Case	High Est.
Dose Consequence Analysis			
Population density & demographics	169 people/sq.mi. (Palisades)	**317 people/sq.mi. (Surry)**	722 people/sq.mi. (Peach Bottom)
Weather conditions & modeling	Same as SFPS (Peach Bottom)	**Same as SFPS (Peach Bottom)**	Same as SFPS (Peach Bottom)
Habitability Limit & health effects	500 mrem annual - LNT	**2 rem first year, 500 mrem thereafter - LNT**	2 rem annual - LNT
Evacuation assumptions & modeling	Same as SFPS (Peach Bottom)	**Same as SFPS (Peach Bottom)**	Same as SFPS (Peach Bottom)
Offsite Property Analysis			
Economic data	Site specific using SECPOP2000) (Palisades)	**Site specific using SECPOP2000) (Surry)**	Site specific using SECPOP2000) (Peach Bottom)

COMSECY-13-0030 Encl 1 Regulatory Analysis

- **Revised Format**

EXECUTIVE SUMMARY
1. INTRODUCTION
2. ANALYSIS OF IDENTIFIED ALTERNATIVE
3. SAFETY GOAL SCREENING EVALUATION
4. COST-BENEFIT ANALYSIS
5. CONCLUSION
6. REFERENCES

APPENDIX A: SPENT FUEL POOL CHARACTERISTICS
APPENDIX B: SPENT FUEL STORAGE STRATEGIES
APPENDIX C: ANALYSIS MODEL INFORMATION
APPENDIX D: SENSITIVITY ANALYSIS INFORMATION
APPENDIX E: INDUSTRY IMPLEMENTATION MODEL OF MOVING SPENT FUEL TO DRY CASK STORAGE
APPENDIX F: SPENT FUEL DATA AND TABLES
APPENDIX G: QUESTIONS RAISED BY THE PUBLIC

COMSECY-13-0030 Encl 1 - Table 2

Topical Area	Major Assumption	Comment
Overall Approach	The fleet of U.S. reactor SFPs were classified in the following groups: 1. BWRs with elevated pools 2. PWRs and BWRs with dedicated pools near grade 3. New AP1000 reactors 4. PWRs that share a single pool 5. PWRs with pools that cannot rapidly drain 6. Decommissioning reactors For the first four groups, representative characteristics of the spent fuel and SFP loading conditions that were conservative with respect to the majority of SFPs within each group were selected. The remaining two groups were not evaluated due to the much lower potential for runaway zirconium oxidation.	The configuration of the plant is considered in determining potential bounding conditions regarding the potential drainage paths from the pools and the potential for natural circulation air cooling. The inventory of fuel, reactor thermal power, and fuel burn-up at reactors within each group are considered in determining the representative inventory of radioactive material present in the pool. Plant characteristics and accident progression for BWRs with elevated pools were drawn from the SFPS. Remaining plant characteristics and accident progression assumptions are drawn from NUREG-1353 and NUREG-1738.
Regulatory Baseline Condition	High-density loading configuration with one full core reserve capacity during which mitigation capability is assumed to be ineffective.	This loading configuration approximates the maximum fuel inventory normally maintained in the SFP. The assumption of ineffective mitigation maximizes the potential release frequency.

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Topical Area	Major Assumption	Comment
Alternative Condition	Low-density loading configuration with fuel decayed more than five years removed from the SFP and mitigation 95% effective.	This loading configuration approximates the minimum fuel inventory for an operating reactor SFP. The assumption of 95% effective mitigation minimizes the frequency of potential releases.
Seismic Hazard Characterization	Seismic hazard models – this analysis used the USGS 2008 model instead of the model currently under development in an ongoing regulatory program. While the USGS (2008) hazard model is not sufficiently detailed for regulatory decisions, it is appropriate to use for this analysis because it was the most recent and readily available hazard model for the central and eastern U.S. plant sites. Hazards for the western sites will be evaluated when the updated model is complete.	A new probabilistic seismic hazard model is currently being developed and will consist of two parts: (1) a seismic source zone characterization and (2) a ground motion prediction equation (GMPE) model. Although part (1) is now complete (Ref. 16), it was not available at the start of this analysis. In addition, the GMPE update is still in progress. Furthermore, the NRC is currently developing an independent probabilistic seismic hazard assessment (PSHA) computer code to incorporate part (1) and part (2) when complete.
Earthquake Frequency	Earthquake frequencies are based on hazard curves developed from 2008 USGS data for two bins having peak ground accelerations of 0.7g and 1.2g, respectively. Large earthquakes with frequencies on the order of a few occurrences every 100,000 years to once every 1,000,000 years have the potential to damage the SFP structure.	The USGS data provides a consistent method of quantifying earthquake frequency east of the Rockies. The low and base cases use the seismic hazard estimate for the SFPS reference plant, which results in higher earthquake frequency estimates than the USGS model for most plants. The high case uses the USGS model results for the site within each group with the highest earthquake frequency.

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Topical Area	Major Assumption	Comment
Cask Drop Frequency	A cask drop frequency of $2x10^{-7}$ per year is used for each SFP.	This value is drawn from an evaluation in NUREG-1738 and represents the potential for cask drops during routine transfer activities to maintain assumed SFP storage inventory. Additional cask movements associated with achieving low-density SFP storage are conservatively not evaluated.
AC Power Fragility	AC power is conservatively assumed to fail during earthquake and cask drop initiators to reflect loss of installed forced cooling and coolant makeup systems.	This assumption results in loss of forced cooling and other minor coolant leaks progressing to uncover the stored fuel unless mitigation is effectively deployed.
Liner Fragility	The values conservatively selected for the base case are: • 10% (SFPS) – 0.7g earthquake for BWRs with elevated pools • 25% (NUREG-1353) – 0.7g earthquake for all other groups • 100% for the 1.2g earthquake • 100% for the cask drop event	Liner Fragility represents the conditional probability of leakage from the SFP at locations that uncover the stored fuel, given an earthquake or cask drop occurs. The high case uses 100% for all initiators.
Other Initiating Event Frequencies	Loss of forced cooling and loss of coolant inventory events are conservatively represented by a total initiating event frequency of $2.37x10^{-7}$ per year.	Individual initiating events affecting loss of forced cooling, loss of AC power, loss of coolant inventory, and seal failures were drawn from NUREG-1738 and NUREG-1353.

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Topical Area	Major Assumption	Comment
Unavailability of Natural Circulation Air Cooling – Partial Drain Conditions	The conservative values selected for the base case are: • 8% – 0.7g earthquake for BWRs with elevated pools (SFPS) • 100% – 0.7g earthquake for all other groups • 100% for the 1.2g earthquake • 100% for the cask drop event • 100% for all other initiators	Unavailability of natural circulation air cooling reflects various conditions that could lead to inadequate heat removal and progression to runaway zirconium cladding oxidation. Conditions bounded by this result include: • fuel with high decay heat • recently discharged fuel in a contiguous pattern rather than distributed pattern • partial drain conditions with racks that block air cooling The high case uses 100% for all initiators.
Mitigation	Effective deployment of mitigation is conservatively assumed to reduce the frequency of release for low-density storage cases by a factor of 19.	Conservative assumption to maximize difference in release frequency between low-density and high-density storage configurations.
Release Frequency Determination	The release frequencies are calculated as the product of the frequency fuel becomes uncovered and the unavailability of air cooling. The frequency fuel becomes uncovered is the product of the initiating event frequency, ac power fragility, and liner fragility for the seismic and cask drop initiators. For all other initiators, the initiating event frequency is the frequency fuel becomes uncovered. For low-density storage configurations, the release frequency is reduced by a factor of 19 to reflect mitigation.	The earthquake and cask drop initiators dominate the events potentially leading to inadequate cooling of the fuel because these events are most likely to cause a leak from the pool at or below the elevation of the stored fuel. Other initiators are conservatively assumed to progress such that the coolant inventory does not adequately cool the stored fuel because of uncertainties in the accident progression.

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Topical Area	Major Assumption	Comment
Cs-137 Release fraction	The SFP Group 1 high-density loading release fractions are: • 3% for the low estimate • 40% for the base case • 90% for the high estimate	The SFPS (Table 27) shows that for the high-density scenarios involving a leak without mitigation measures, the maximum release is approximately 40%, which was used for the base case. A 90% release fraction is used for the high estimate to account for SFP variations within the group and uncertainties in the accident progression.
	The SFP Groups 2, 3 and 4 high-density loading release fractions used are: • 10% for the low estimate • 75% for the base case • 90% for the high estimate	These release fractions are consistent with the range of release fractions used in previous SFP studies.
	The SFP Group 1, 2, 3, and 4 low-density loading release fractions are: • 0.5% for the low estimate • 3% for the base case • 5% for the high estimate	The SFPS (Table 28) shows that for the low-density scenarios involving a leak without mitigation measures, the maximum release is approximately 3%, which was used for the base case. A 5% release fraction is used for the high estimate to account for SFP variations within the group and uncertainties in the accident progression. The release fractions are the same for all groups because only the most recently discharged fuel is expected to be involved.

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Topical Area	Major Assumption	Comment
Radionuclide Source Term	A source term calculated by the MELCOR code based on the cesium release fraction.	The MELCOR code models the fuel damage state, radionuclide release, and holdup of aerosols.
Atmospheric Modeling and Meteorology	The atmospheric transport and dispersion model used in this analysis is based on the MACCS2 model developed using weather data for the Peach Bottom site, which is described in Section 7.1.2 of the SFPS.	A straight-line Gaussian plume segment dispersion model is used for the atmospheric transport.
Population and Economic Data	Representative site demographics are selected to represent the 90th percentile, the mean, the median, and the 20th percentiles. For each representative site, the site population and economic data is established for use in the consequence analysis.	Representative sites for the 90th percentile, the mean, the median, and the 20th percentile are Peach Bottom, Surry, Palisades, and Point Beach, respectively. To identify the specific effect of these values, the staff performed sensitivity studies where only one parameter was varied from a low to high value. Section 4 discusses this sensitivity study in more detail.
Emergency Response Model	The site-specific emergency response model from the SFPS is used to model evacuation timing and speed within the emergency planning zone.	The conditional individual risk measures near the site are expected to be relatively insensitive to site-specific characteristics (i.e., emergency response measures). This is because the predicted releases allow time for effective protective actions to limit exposures to the public.

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Topical Area	Major Assumption	Comment
Long-Term Habitability Criteria	The long-term phase is modeled for 50 years to calculate the consequences of exposure to the average person assuming habitation is limited to areas where annual dose is within the criteria. The base case uses habitability criteria of 2 rem in the first year and 500 mrem each year thereafter. The high case uses a criterion of 2 rem annually.	The selected habitability criteria affect the values of offsite property damage used in this analysis. Certain metrics such as offsite property damage, the number of displaced individuals (either temporarily or permanently) and the extents to which such actions may be needed are inversely proportional to changes in collective dose resulting from changes in habitability criteria.
Accident Occupational Exposure	Occupational exposures related to accident mitigation and recovery are estimated based on actual worker doses collected for the Fukushima Dai-ichi site.	The assumed accident period extends for one year and involves a work force of 3,700 people.
Health Consequences	The Linear No Threshold (LNT) dose-response model is used as the base for reporting results. The dose truncation methodology, introduced in the SOARCA analyses documented in NUREG-1935, is provided as a sensitivity analysis.	For large populations exposed to low annual doses, which is the case for some of the SFP accident scenarios, the health effects to populations in habitable zones dominate the health effects when the LNT model is used.

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Topical Area	Major Assumption	Comment
Implementation Cost Approach and Timing of Cask Loading	For the regulatory baseline, the plant is expected to load the required number of dry storage casks each refueling cycle to retain sufficient space in the SFP to discharge one full core of fuel. For the low-density storage alternative in Groups 1, 2, and 4, the plant is assumed to transfer all fuel that has greater than 5 years decay within a 5 year period and then continue loading dry storage casks each refueling cycle as necessary to maintain a full core reserve. For the low-density storage alternative in Group 3, the plant is expected to begin loading dry storage casks once the pool reaches the allowed capacity in a low-density (1x4) configuration.	Group dry storage cask loading is based on a representative plant selected within each group. The total number of dry storage casks necessary for the low-density storage alternative is higher than for the regulatory baseline because fuel assemblies that have decayed for shorter periods have higher decay heat levels, and the higher decay heat per assembly reduces the allowed capacity below its nominal capacity.
Occupational Dose	For the low-density storage alternative, each cask loaded in addition to the number required by the regulatory baseline is estimated to result in an incremental 400 person-mrem dose.	This radiation dose is consistent with the exposure value used in EPRI TR-1021049 (Ref. 17) and in EPRI TR-1018058 (Ref. 18), which analyzed worker impacts associated with loading spent fuel for transport to the proposed Yucca Mountain repository.
Incremental Upfront Cost of ISFSI Capacity	Each additional dry storage cask is expected to require engineering, design and construction costs of $657,700 in 2012 dollars.	Each of these cost components are further described in EPRI TR-1021048, “Industry Spent Fuel Storage Handbook.”

COMSECY-13-0030 Encl 1 - Table 2

Topical Area	Major Assumption	Comment
Incremental Cost of Additional Cask purchase and Loading	The base cost for purchase and loading of a dry storage cask is assumed to be $1,300,000. When only 5-year decayed, high-burnup fuel is available for loading, additional shielding; engineering, licensing, and operational expenses are assumed to increase the cost to $1,466,400 per cask.	These cost estimates are based on the DSC unit costs that EPRI used for a generic interim storage facility and documented in EPRI TR-1025206.
Incremental Annual ISFSI Operating Costs	The majority of reactor sites in Groups 1, 2, and 4, have operational ISFSIs, and the incremental operating cost for increased capacity is considered negligible for these groups. For Group 3, maintenance of low-density storage is expected to require early operation at an incremental cost of $1.1 million per year.	EPRI reports a wide variability in published estimates of annual ISFSI operating costs that range from $212,000 to $2 million per year in 2012 dollars and reported their estimate of $1.1 million per year for an ISFSI at an operating nuclear power plant site.

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Net Monetary Savings (or Costs) – Total Present Value	Sensitivity Studies	Qualitative Benefits and (Costs)
Expedited Transfer Alternative – Low-density Spent Fuel Pool Storage		
Group 1 – BWR Mark I and Mark II with non-shared SFPs		
Group 1 Industry (Costs): **Base case** **($52 million) using a 7% discount rate** **NRC (Costs):** **Not calculated** **Benefits:** **Base case** **$8.6 million using a 7% discount rate** **Group 1 Net Benefit = Benefits + (Costs)** **Base case: $8.6M + ($52M) = ($43.4M)** **Conclusion: Not cost beneficial**	Group 1 Sensitivity Studies Industry (Costs) Sensitivity Studies ($53 million) using a 2% discount rate ($55 million) using a 3% discount rate Benefit Sensitivity Studies Low estimate $0.2 million using a 2% discount rate $0.2 million using a 3% discount rate $0.1 million using a 7% discount rate High estimate $123 million using a 2% discount rate $109 million using a 3% discount rate $73 million using a 7% discount rate Net Benefit Sensitivity Studies Low estimate ($52.8M) using a 2% discount rate ($54.8M) using a 3% discount rate ($51.9M) using a 7% discount rate High estimate $70 million using a 2% discount rate $54 million using a 3% discount rate $21 million using a 7% discount rate	Qualitative Benefits and (Costs) Qualitative (Costs): Cost Uncertainties (Repackaging Costs) Qualitative Benefits: Modeling Uncertainties. (Cask Handling Risk) Mitigating Strategies

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Group 2 – PWR and BWR Mark III with non-shared SFPs		
Group 2 Industry (Costs): **Base case** **($51 million) using a 7% discount rate** **NRC (Costs):** **Not calculated** **Benefits:** **Base case** **$7.9 million using a 7% discount rate** **Group 2 Net Benefit = Benefits + (Costs)** **Base case: $7.9M + ($51M) = ($43.1M)** **Conclusion: Not cost beneficial**	Group 2 Sensitivity Studies Industry (Costs) Sensitivity Studies ($51 million) using a 2% discount rate ($54 million) using a 3% discount rate Benefit Sensitivity Studies Low estimate $0.3 million using a 2% discount rate $0.3 million using a 3% discount rate $0.2 million using a 7% discount rate High estimate $137 million using a 2% discount rate $121 million using a 3% discount rate $77 million using a 7% discount rate Net Benefit Sensitivity Studies Low estimate ($50.7M) using a 2% discount rate ($53.7M) using a 3% discount rate ($50.8M) using a 7% discount rate High estimate $86 million using a 2% discount rate $67 million using a 3% discount rate $26 million using a 7% discount rate	Qualitative Benefits and (Costs) Qualitative (Costs): Cost Uncertainties (Repackaging Costs) Qualitative Benefits: Modeling Uncertainties. (Cask Handling Risk) Mitigating Strategies

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Group 3 – New reactor SFPs		
Group 3 Industry (Costs): **Base case** **($17 million) using a 7% discount rate** **NRC (Costs):** **Not calculated** **Benefits:** **Base case** **$5.6 million using a 7% discount rate** **Group 3 Net Benefit = Benefits + (Costs)** **Base case: $5.6M + ($17M) = ($11.4M)** **Conclusion: Not cost beneficial**	Group 3 Sensitivity Studies Industry (Costs) Sensitivity Studies ($42 million) using a 2% discount rate ($36 million) using a 3% discount rate Benefit Sensitivity Studies Low estimate $0.3 million using a 2% discount rate $0.3 million using a 3% discount rate $0.1 million using a 7% discount rate High estimate $108 million using a 2% discount rate $81 million using a 3% discount rate $34 million using a 7% discount rate Net Benefit Sensitivity Studies Low estimate ($41.7M) using a 2% discount rate ($35.7M) using a 3% discount rate ($16.9M) using a 7% discount rate High estimate $66 million using a 2% discount rate $45 million using a 3% discount rate $17 million using a 7% discount rate	Qualitative Benefits and (Costs) Qualitative (Costs): Cost Uncertainties (Repackaging Costs) Qualitative Benefits: Modeling Uncertainties. (Cask Handling Risk) Mitigating Strategies

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Group 4 – Reactor units with shard SFPs		
Group 4 Industry (Costs): **Base case** **($46 million) using a 7% discount rate** **NRC (Costs):** **Not calculated** **Benefits:** **Base case** **$8.9 million using a 7% discount rate** **Group 4 Net Benefit = Benefits + (Costs)** **Base case: $8.9M + ($46M) = ($37.1M)** **Conclusion: Not cost beneficial**	Group 4 Sensitivity Studies Industry (Costs) Sensitivity Studies ($49 million) using a 2% discount rate ($50 million) using a 3% discount rate Benefit Sensitivity Studies Low estimate $0.3 million using a 2% discount rate $0.3 million using a 3% discount rate $0.2 million using a 7% discount rate High estimate $205 million using a 2% discount rate $182 million using a 3% discount rate $120 million using a 7% discount rate Net Benefit Sensitivity Studies Low estimate ($48.7M) using a 2% discount rate ($49.7M) using a 3% discount rate ($48.8M) using a 7% discount rate High estimate $156 million using a 2% discount rate $132 million using a 3% discount rate $74 million using a 7% discount rate	Qualitative Benefits and (Costs) Qualitative (Costs): Cost Uncertainties (Repackaging Costs) Qualitative Benefits: Modeling Uncertainties. (Cask Handling Risk) Mitigating Strategies

Figure 9 Comparison of annual PGA exceedance frequencies for U.S. BWR Mark I and Mark II reactors (USGS 2008 model)

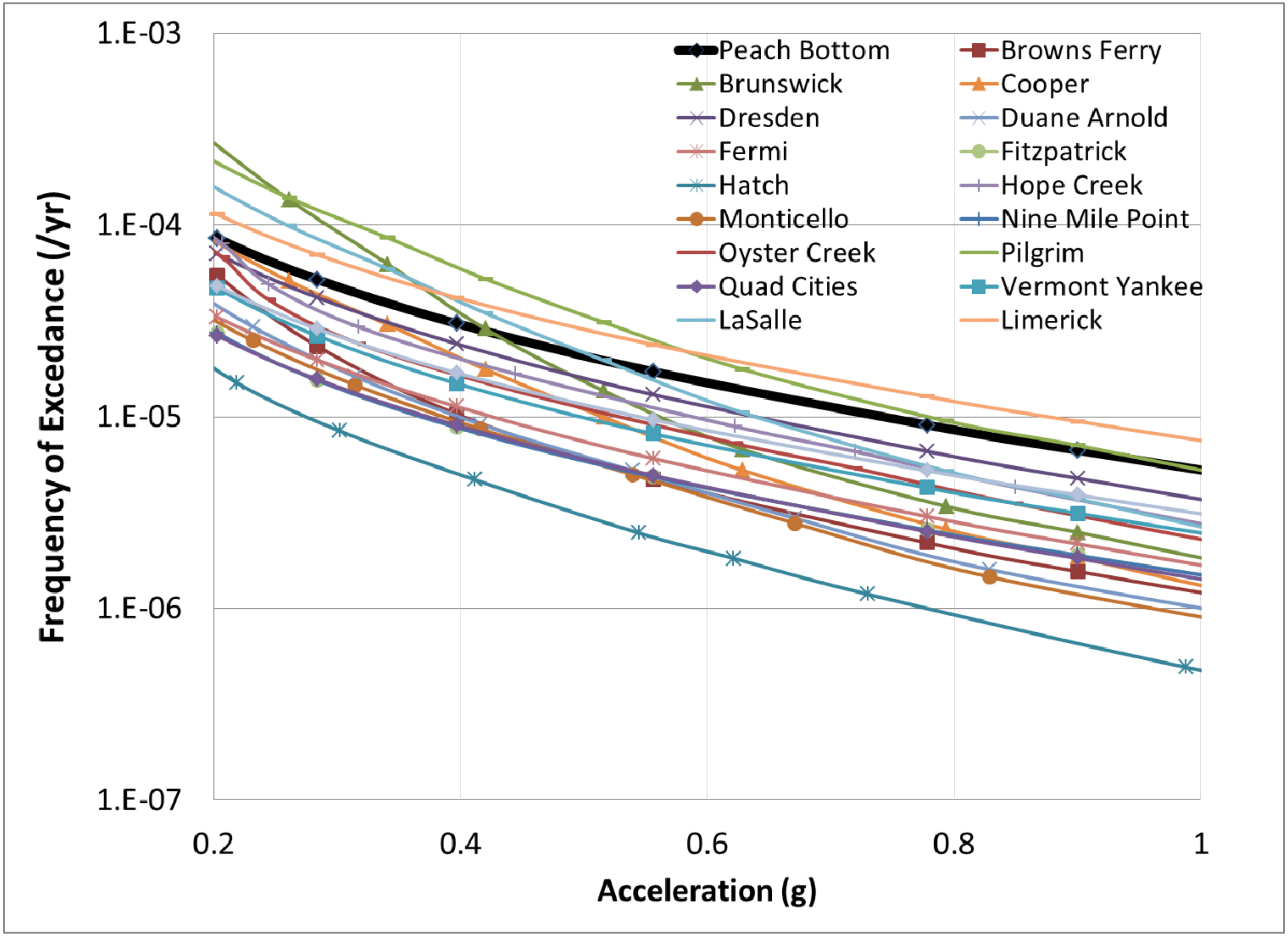

Figure 10 Comparison of annual PGA exceedance frequencies for U.S. PWR and BWR Mark III reactors (USGS 2008 model)

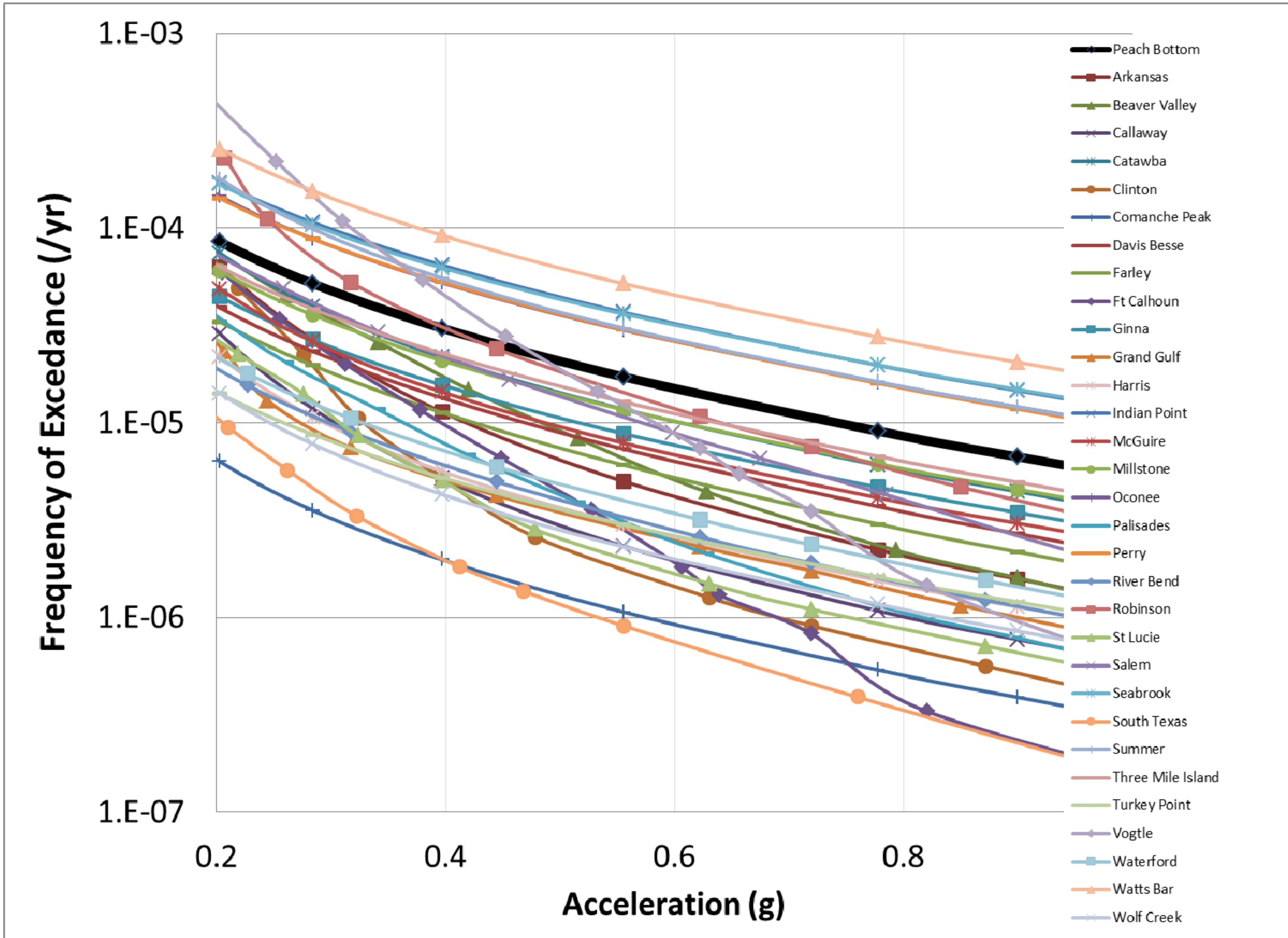

Figure 11 Comparison of annual PGA exceedance frequencies for new U.S. reactors (USGS 2008 model)

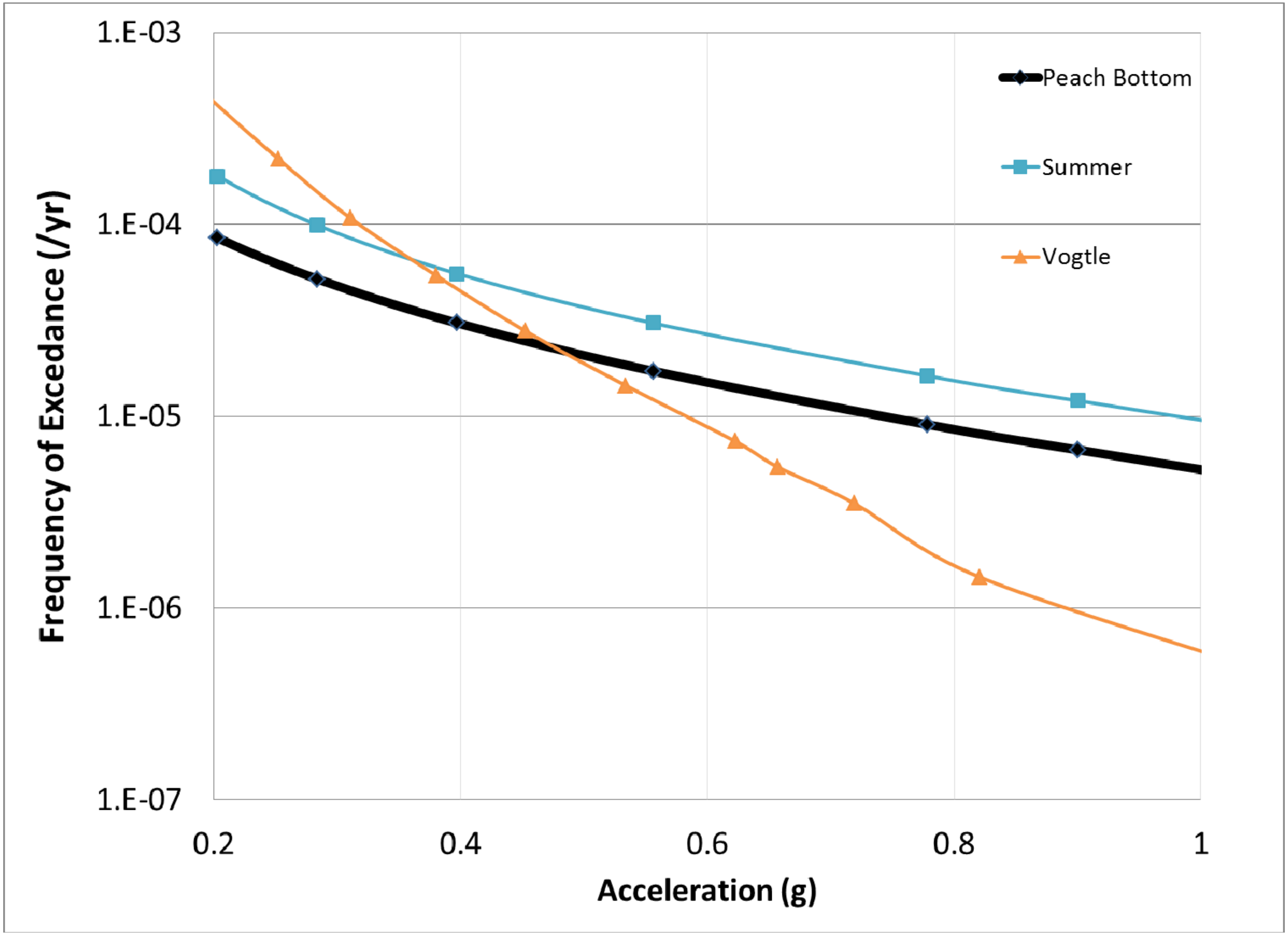

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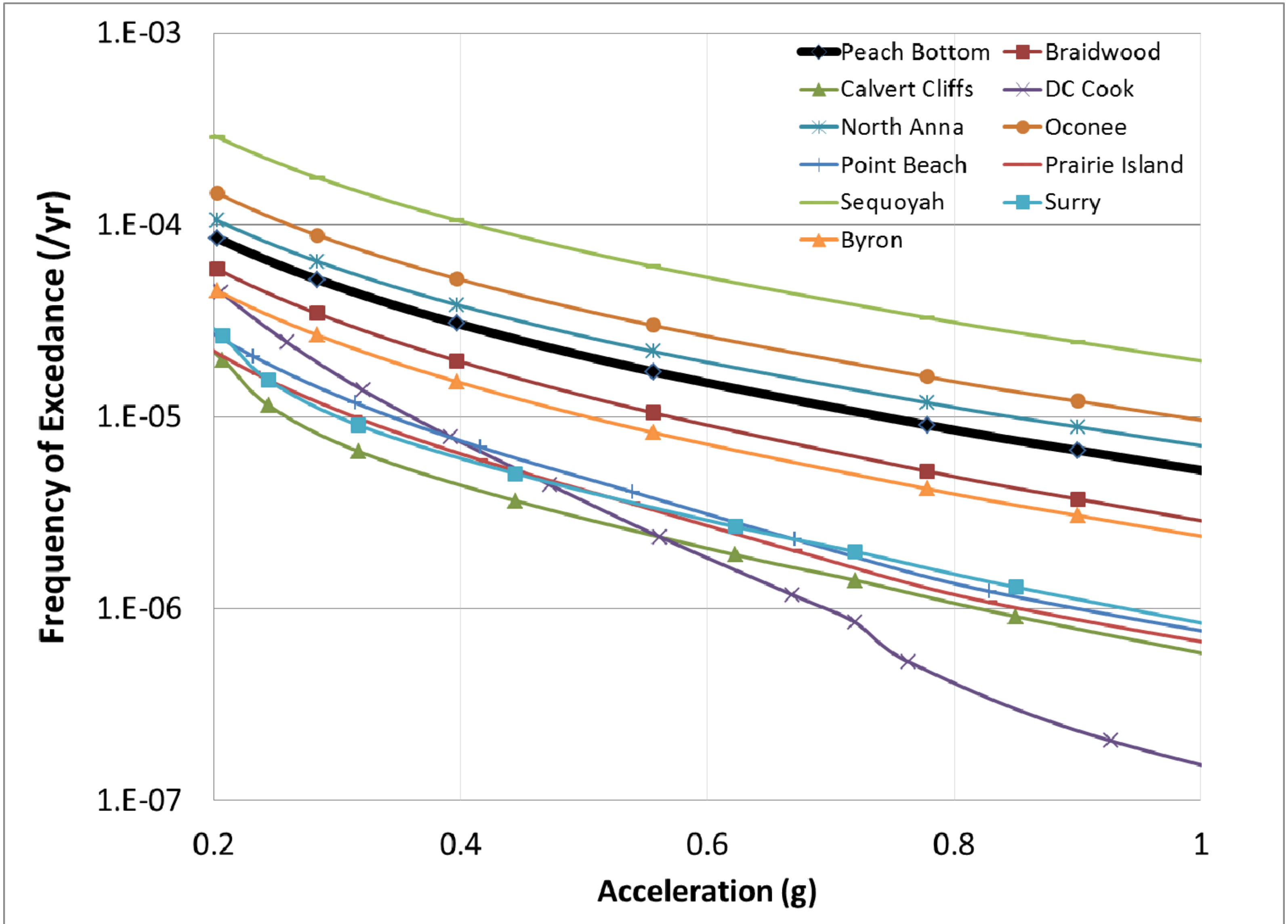
Figure 12 Comparison of annual PGA exceedance frequencies for U.S. reactors with a shared spent fuel pool (USGS 2008 model)



COMSECY-13-0030

November 12, 2013

MEMORANDUM TO: Chairman Macfarlane
Commissioner Svinicki
Commissioner Apostolakis
Commissioner Magwood
Commissioner Ostendorff

FROM: Mark A. Satorius ***/RA/***
Executive Director for Operations

SUBJECT: STAFF EVALUATION AND RECOMMENDATION FOR JAPAN LESSONS-LEARNED TIER 3 ISSUE ON EXPEDITED TRANSFER OF SPENT FUEL

The purpose of this memorandum is to provide the Commission with information and a recommendation on whether additional study is warranted to assess possible regulatory action to require expeditious transfer of spent fuel from nuclear power plants' spent fuel pools to dry cask storage.

SUMMARY:

The accident at the Fukushima Dai-ichi nuclear facility in Japan led to questions about the safe storage of spent fuel and whether the U.S. Nuclear Regulatory Commission (NRC) should require expedited transfer of spent fuel to dry cask storage at nuclear power plants in the United States (U.S.). The staff completed a regulatory analysis (provided in Enclosure 1 of this memorandum) to determine if additional study of this issue is warranted (i.e., on whether reactor licensees should be required to reduce the amount of spent fuel stored in their spent fuel pools (SFPs)). The staff has considered a broad history of NRC oversight of spent fuel storage, SFP operating experience (domestic and international), past studies of SFP safety, and the October 2013 SFP study. In addition, the staff considered international practices related to the transfer of spent fuel from wet to dry storage, and stakeholder comments received during two public meetings.

To determine whether regulatory action might be warranted, the staff has conducted an analysis of expediting the transfer of spent fuel assemblies. As part of its regulatory analysis, the staff first conducted a safety goal screening evaluation using the Commission's safety goal policy statement. Although the agency's guidance would normally allow the staff to stop the evaluation upon determining that the proposed action does not provide a sufficient safety enhancement to meet the threshold of the safety goal screening, the staff proceeded to perform a cost benefit analysis to provide the Commission additional information. The staff concludes that the

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expedited transfer of spent fuel to dry cask storage would provide only a minor or limited safety benefit (i.e., less than safety goal screening criteria), and that its expected implementation costs would not be warranted. The staff therefore recommends that additional studies and further regulatory analyses of this issue not be pursued, and that this Tier 3 Japan lessons-learned activity be closed.

Some staff expressed comments that resulted in a non-concurrence on this memorandum, which is provided as Enclosure 2. The non-concurrence advocates performing additional studies of possible cost-effective approaches to improving the safety of SFPs. The non-concurrence also suggests that the supporting analyses should have been performed differently, that several policy issues should be identified to the Commission, and that the resulting information should be presented in a more neutral manner. The staff made improvements to this memorandum in response to the questions and comments identified in the non-concurrence. However, after considering the analysis results, operating history, and limited safety benefits of possible plant changes, the staff finds that further study would be unlikely to support a requirement that reactor licensees expedite the transfer of spent fuel from their SFPs into dry cask storage.

BACKGROUND:

There are a variety of postulated events or conditions that can challenge the ability of a SFP to provide adequate cooling to spent fuel assemblies. A loss of heat removal from the SFP, which could be caused by a loss of electrical power, produces a slowly evolving event that could be mitigated with a high probability of success by plant staff and available equipment. Potentially more significant events involve coolant inventory loss resulting from a loss of pool integrity. These events could result from low likelihood initiators such as a large earthquake producing ground accelerations well above those considered in the design of the facility. Past and recent studies have shown that these types of events could potentially lead to large radiological releases. Common to all event scenarios, significant radiological releases can only result if spent fuel heat loads exceed heat removal capacity such that fuel cladding temperature increases are sufficient to cause zirconium cladding ignition and resultant fire. However, regardless of the initiator, this outcome evolves relatively slowly, with time for mitigative and/or protective actions to prevent a release or otherwise ensure public health and safety.

On March 11, 2011, a 9.0-magnitude earthquake struck Japan and was followed by a 45-foot tsunami, which resulted in extensive damage to the nuclear power reactors at the Fukushima Dai-ichi facility. After the onset of core damage in some units, there were significant concerns about the integrity of SFPs and the possible release of radioactive materials from the spent fuel assemblies. However, subsequent inspections determined that pool integrity had been maintained, the integrity of the spent fuel cladding had not been challenged, and equipment to restore coolant inventory had been successfully deployed, despite radiological hazards and extensive damage to the surrounding structures from the tsunami and hydrogen explosions. While the SFPs and the spent fuel assemblies at the site remained intact, the event led to questions about the safe storage of spent fuel and whether the NRC should require expedited transfer of spent fuel to dry cask storage at nuclear power plants.

In the summer of 2011, the staff initiated a research project entitled, “Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water

Reactor." The resultant report, dated October 2013 (commonly referred to as the SFP study), can be accessed in the Agencywide Documents Access and Management System (ADAMS) under Accession No. ML13256A342. The purpose of the SFP study was to provide additional information to help determine if accelerated transfer of spent fuel from the SFP to dry cask storage significantly reduces risks to public health and safety. The SFP study provides consequence estimates for a hypothetical SFP accident initiated by a low likelihood seismic event at a reference plant for both a fully loaded (high-density) and minimally loaded (low-density) SFP. The SFP study contributed to the resolution of this Tier 3 issue by providing a measure of the change in potential consequences resulting from a change in spent fuel storage density for a reference plant.

In SECY-11-0137, "Prioritization of Recommended Actions To Be Taken in Response to Fukushima Lessons Learned," dated October 3, 2011 (ADAMS Accession No. ML11272A111), the staff identified six additional issues that may warrant regulatory action but were not included with the Near-Term Task Force (NTTF) recommendations. One additional issue was the expedited transfer of spent fuel to dry cask storage. The staff judged this issue to warrant further consideration and prioritization based on potential safety significance, nexus to NTTF recommendations, and other ongoing staff activities. As directed by a Staff Requirements Memorandum (SRM), SRM-SECY-11-0137, dated December 15, 2011 (ADAMS Accession No. ML113490055), the staff conducted an assessment of whether this issue should be included with the Japan lessons-learned activities and whether any regulatory action is recommended or necessary. The staff applied the same prioritization process described in SECY-11-0137. In SECY-12-0025, "Proposed Orders and Requests for Information in Response to Lessons Learned from Japan's March 11, 2011, Great Tohoku Earthquake and Tsunami," dated February 17, 2012 (ADAMS Accession No. ML12039A103), the staff prioritized this issue in the Tier 3 category since it required further staff study to determine if regulatory action is warranted.

In SECY-12-0095, "Tier 3 Program Plans and 6-Month Status Update in Response to Lessons Learned from Japan's March 11, 2011, Great Tohoku Earthquake and Subsequent Tsunami," dated July 13, 2012 (ADAMS Accession No. ML12165A092), the staff provided a five-step plan to evaluate whether regulatory action is warranted for the expedited transfer of spent fuel from SFPs into dry cask storage. After submitting the Tier 3 program plan, the staff received direction in several SRMs:

- In SRM-M120607C, "Staff Requirements—Meeting with the Advisory Committee on Reactor Safeguards, 9:30 A.M., Thursday, June 7, 2012, Commissioners' Conference Room, One White Flint North, Rockville, Maryland (Open to Public Attendance)," dated July 16, 2012 (ADAMS Accession No. ML121980043), the Commission provided the staff with direction on several topics on additional research activities (e.g., human reliability analysis and comparative assessment to previous SFP studies) that the SFP study should address.

- In SRM-M120807B, "Staff Requirements—Briefing on the Status of Lessons Learned from the Fukushima Dai-ichi Accident, 9:00 A.M., Tuesday, August 7, 2012, Commissioners' Conference Room, One White Flint North, Rockville, Maryland (Open to Public Attendance)," dated August 24, 2012 (ADAMS Accession No. ML122400033), the

Commission directed the staff to address international practices related to spent fuel management as part of the Tier 3 program plan for expedited transfer of spent fuel.

In a memorandum to the Commission entitled, "Updated Schedule and Plans for Japan Lessons-Learned Tier 3 Issue on Expedited Transfer of Spent Fuel," dated May 7, 2013 (ADAMS Accession No. ML13105A122), the staff outlined a three phase plan for evaluating whether regulatory action is warranted to require licensees to expedite transfer of spent fuel from SFPs to dry cask storage. The program plan calls for preparing this memorandum under Phase 1 to help determine if additional study is warranted. If the results of Phase 1 would indicate that additional study is warranted, Phases 2 and 3 of the program plan would be conducted to refine assumptions used in the analyses to determine whether any regulatory action is warranted. The Phase 1 analysis is the subject of this memorandum and considers the results of the SFP study along with previous studies and operating experience. The results are discussed below.

DISCUSSION:

In evaluating if additional studies are needed on whether to require expedited transfer of spent fuel to dry cask storage, the staff has considered a broad history of NRC oversight of spent fuel storage, SFP operating experience (domestic and international), past studies of SFP safety, and the October 2013 SFP study. The NRC's regulatory activities and past studies have shown that SFPs are effectively designed to prevent accidents that could affect the safe storage of spent fuel. The past studies of SFP safety and the October 2013 SFP study provide detailed assessments of SFP safety. Operating experience has shown that SFPs have safely withstood challenging events, maintaining structural integrity and a large inventory of coolant to protect the stored fuel.

Design and Licensing

The SFPs at operating U.S. reactors were designed and licensed to maintain a large inventory of coolant to protect and cool the fuel under accident conditions, including earthquakes. SFPs were constructed to be robust structures with very thick steel-reinforced concrete walls and floors. The pools' thick walls, floors, and stainless steel liner help maintain the coolant inventory and protect the fuel from the effects of natural phenomena. SFPs are generally configured to protect against a substantial loss of coolant inventory by locating penetrations in the SFP wall above the top of the stored fuel, and by providing anti-siphon features for piping that extend below the top of the fuel within the pool. These features limit the likelihood of losing substantial coolant inventory due to mechanical failures or operational errors. Through the NRC's regulatory oversight for all SFPs, the staff has determined that they provide a safe means of storing spent fuel.

Operating Experience

Operating experience with spent fuel storage in pools confirms that SFPs have provided adequate protection of public health and safety. The staff previously completed a detailed review of SFP operating experience in NUREG 1275, Volume 12, "Operating Experience

Feedback Report, Assessment of Spent Fuel Cooling," dated February 1997 (ADAMS Accession No. ML010670175), and the staff performs annual reviews of U.S. and international operating experience with spent fuel storage and handling. The robustness of SFP designs in preventing significant loss of inventory or cooling has been demonstrated by the minor impact of events identified in these reviews. For example, early problems with seal leakage around large penetrations above the elevation of the stored fuel have been resolved by seal design changes. Operational issues affecting configuration control of SFP cooling and purification systems also have decreased in frequency. Operating experience reviews have indicated that events involving loss of coolant inventory or loss of forced cooling have had no more than a minor effect (e.g., increases in water temperature) on spent fuel storage conditions.

The staff has reviewed information on the effect of earthquakes up to several times greater than design-basis values on the integrity of SFPs and has determined that the SFPs are robust and in all cases have maintained safe storage of spent fuel. The staff has reviewed information on SFP performance during the March 11, 2011, Great Tohoku Earthquake and the July 16, 2007, Niigataken Chuetsu-Oki earthquake, which affected 20 operating reactors in Japan, including Fukushima Dai-ichi and Kashiwazaki-Kariwa. Of the SFPs at these 20 reactors, there was no observed significant damage of the SFP structure or any penetrations (i.e. no loss of integrity), and any water loss caused by sloshing resulted in only a minor loss of coolant inventory. A complete discussion of this evaluation is provided in Section 4.3 of the SFP study. Additionally, the Mineral, Virginia, earthquake of August 23, 2011, which occurred near the North Anna nuclear power plant, produced ground motions near the design basis for that plant, and did not result in damage or loss of water from that plant's SFP.

Recent Regulatory Actions To Enhance Safety

In response to the Fukushima Dai-ichi accident, the staff is currently implementing regulatory actions, which originated from the NTTF recommendations, to further enhance reactor and SFP safety. On March 12, 2012, the staff issued Order EA-12-051, "Issuance of Order To Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation," (ADAMS Accession No. ML12054A679), which requires that licensees install reliable means of remotely monitoring wide-range SFP levels to support effective prioritization of event mitigation and recovery actions in the event of a beyond-design-basis external event. Although the primary purpose of the order was to ensure that operators were not distracted by uncertainties related to SFP conditions during the accident response, the improved monitoring capabilities will help in the diagnosis and response to potential losses of SFP integrity. In addition, on March 12, 2012, the staff issued Order EA-12-049, "Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events" (ADAMS Accession No. ML12054A735), which requires licensees to develop, implement, and maintain guidance and strategies to maintain or restore SFP cooling capabilities, independent of alternating current power, following a beyond-design-basis external event. These requirements ensure a more reliable and robust mitigation capability is in place to address degrading conditions in SFPs than was assumed in the SFP study. For the purpose of evaluating the potential benefits of expedited transfer of spent fuel to dry cask storage, the enclosed analysis used a conservative approach to mitigation by crediting successful mitigation to the low-density SFP storage alternative (i.e., conditions following expedited transfer) and assumed no successful mitigation for the high-density SFP storage regulatory baseline.

Evaluation of Expedited Transfer of Spent Fuel to Dry Cask Storage

To evaluate whether additional studies are needed to assess possible regulatory actions, the staff has prepared the enclosed regulatory analysis of expedited transfer of spent fuel to dry cask storage. A regulatory analysis is an established analytical tool to help determine if a proposed regulatory action should be implemented.

In the first step of the analysis, the staff used the quantitative health objectives (QHO)[1] in conducting its safety goal screening evaluation. The QHOs are used as a surrogate for the safety goal as outlined in the Commission's safety goal policy statement. Although the QHOs were developed based on the risk from severe reactor accidents, they provide the only readily-available risk criteria for regulatory decisionmaking regarding non-reactor accidents. A further discussion of the basis and background for using the QHOs in assessing SFP accidents is included in the October 2013 SFP study and in Section 3 of Enclosure 1. The staff relied on information from past studies, the October 2013 SFP study, and operating experience to conduct the safety goal screening evaluation. The safety goal screening evaluation concludes that SFP accidents are a small contributor to the overall risks for public health and safety (less than one percent of the QHOs), and therefore any reductions in risk associated with expedited transfer of spent fuel would only have a marginal safety benefit. Due to the safety goal screening criterion not being satisfied, the staff recommends that no further generic assessments be pursued. Although the regulatory analysis guidelines would normally allow the staff to stop the evaluation at this step, the staff proceeded to perform a cost-benefit analysis to provide additional information for the Commission's consideration.

In its cost-benefit analysis, the staff develops estimates of costs and quantified benefits, together with a conclusion as to whether the proposed regulatory action is cost-beneficial. "Cost-beneficial" means that the benefits of the proposed action are equal to, or exceed, the costs of the proposed action. The NRC's practice of assessing whether potential benefits of new regulations warrant the associated costs is similar to that used by other federal agencies. Within the enclosed analysis, the staff provides a "base case" which generally used conservative assumptions for key parameters such as conditional probabilities of SFP liner failures and loss of adequate cooling to increase the calculated benefits of expedited transfer of spent fuel (i.e., to skew the calculations towards pursuing additional studies). The benefits calculated for the base case evaluations are less than the estimated costs for requiring expedited transfer of spent fuel to dry cask storage. Although the base case is used as the

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1 The two QHOs are a prompt fatality QHO and a latent cancer fatality QHO. The prompt fatality QHO is that 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 1/10 of 1 percent (0.1 percent) of the sum of prompt fatality risks resulting from other accidents to which members of the U.S. population are generally exposed. This represents a frequency of prompt fatalities of less than $5x10^{-7}$ per year for an average individual within 1 mile of a plant. The cancer fatality QHO is that 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 1/10 of 1 percent (0.1 percent) of the sum of cancer fatality risks resulting from all other causes. This represents a frequency of cancer fatalities of less than $2x10^{-6}$ per year for an average individual within 10 miles of a plant. ("Safety Goals for Nuclear Power Plant Operation," NUREG-0880, Rev. 1, issued May 1983.)

primary basis for the staff's recommendation, the staff also analyzed additional cases where key parameters are varied to provide low and high estimates of the calculated benefits. The staff used bounding or conservative values in the analysis for several parameters, particularly in the high estimate cases, to ensure that design, operational, and other site variations among the new and operating reactor fleet were addressed and to generally increase the calculated benefits from the proposed action.

Sensitivity studies were also conducted on key factors such as the dollars per person-rem conversion factor, population density, habitability criteria and consideration of consequences beyond 50 miles to measure each attribute's effect upon the overall result. The sensitivity of the dollars per person-rem conversion factor is important to consider because related guidance is currently being updated. The sensitivity of consequences beyond 50 miles is important to consider for accidents involving SFP fires, as the spread of radioactive materials could extend over long distances. The supporting analysis used key insights from operating experience, the October 2013 SFP study, and previous studies on SFP safety, such as the plant damage state for seismic events, probability of a release for specific pool damage states, and the expected amount and type of radioactive material released. The various cases and sensitivity studies show that while the impacts on public health and safety for an average individual are, for the most part, very low, collective dose and economic consequences for these low probability events can be very large.[2] The combination of high estimates for important parameters assumed in some of the sensitivity cases presented in Enclosure 1 result in large economic consequences, such that, the calculated benefits from expedited transfer of spent fuel to dry cask storage for those cases outweigh the associated costs (see Section 4.4.1.4 in enclosed regulatory analysis). However, even in these cases, there is only a limited safety benefit when using the QHOs and the expected implementation costs would not be warranted. In addition, in the staff's judgment, the various assumptions made in the analysis of the "base case" result in an overall cost-benefit assessment that is appropriately conservative for a generic regulatory decision and justify using the "base case" as the primary basis for the staff's recommendation. Based on the generic assessment and the other considerations detailed in this memorandum, the staff finds that additional studies are not needed to reasonably conclude that the expedited transfer of spent fuel to dry cask storage would provide only a minor or limited safety benefit, and that its expected implementation costs would not be warranted.

The staff evaluated seismic risks and other types of severe events when considering the safety significance and the relationship of costs and benefits of possible regulatory actions to expedite removing fuel assemblies from SFPs. In past SFP studies and the October 2013 SFP study, the staff has evaluated seismic events because they have been identified as the largest risk contributor to SFP safety. Based on the latest seismic hazard curves developed for nuclear power plant sites in the central and eastern United States, the overall estimated frequency of significant spent fuel damage continues to be very low for these facilities (approximately five times per million years). Updated structural and seismic hazard information for operating reactors in the western United States is being developed as part of NTTF Recommendation 2.1 activities. Considering the robust designs of SFPs, especially in more seismically active areas

[2] The staff notes that in its SRM on SECY-12-0110, the Commission stated "economic consequences should not be treated as equivalent in regulatory character to matters of adequate protection of public health and safety."

in the western United States, the staff concludes that public health and safety are adequately protected. At the completion of the NTTF Recommendation 2.1 seismic reevaluation, the staff will confirm that the seismic risk for SFPs is consistent with that considered in the enclosed analysis. Because various studies and regulatory changes implemented following the terrorist attacks of September 11, 2001, have considered security issues associated with SFPs, malevolent acts are not included in this analysis. The details of the staff's review of security issues involve sensitive and classified information and are therefore not available to the public.

In addition to assessing whether further studies of expedited transfer of spent fuel to dry cask storage are warranted, the SFP study and staff's interactions with stakeholders identified other possible improvements to the storage of spent fuel. Examples include the possible investigation of alternate loading patterns (e.g., the 1 x 8 high-density loading pattern assessed in the SFP study, in addition to the standard 1 x 4 high-density loading pattern), capability of licensees to directly offload fuel into more coolable patterns, and the possible enhancement of mitigation strategies during identified periods when the heat load from recently discharged fuel assemblies is especially high. The staff has considered these possible improvements, and notes that these alternatives would likely involve lower costs than would the expedited transfer of spent fuel to dry cask storage. However, these alternatives would provide only a limited safety benefit when using the QHOs, and their implementation costs would not be warranted. This finding reflects the low probability of the initiating events that would challenge the integrity of the SFPs and the fact that these alternative actions would have similar or lesser safety benefit in comparison to those estimated for the expedited transfer of spent fuel. However, licensees will be informed of and encouraged to assess and implement, as appropriate, such improvements on their own initiative to help manage the risks associated with plant specific SFP designs, operating practices, and mitigation capabilities.

International Practices

As directed in SRM-M120807B, the staff assessed international practices related to spent fuel storage and determined that current U.S. fuel storage practices are consistent with international practices. The staff determined that commercial U.S. operating reactor sites typically have greater inventories of spent fuel stored on site than otherwise comparable foreign reactors. This principally reflects the longer period of operation and the high capacity factors that U.S. operators have achieved. Countries with options for centralized storage, either in preparation for disposal (e.g., Sweden) or reprocessing (e.g., England, France, and Japan), have nevertheless adopted high-density storage at reactor sites. The staff's review did not identify any country with an explicit policy for early transfer of fuel to dry or centralized storage to maintain low density storage in the onsite SFPs.

Stakeholder Interactions

To provide additional insights on the need for regulatory action, the staff interacted with various stakeholders. The nuclear industry provided insights to the staff through various interactions and also through reports prepared by the Electric Power Research Institute. Several nongovernmental organizations and individuals provided correspondence and attended public meetings to give information to the staff. Public meetings were held on August 22, 2013 (meeting summary in ADAMS under Accession No. ML13253A162), and September 18, 2013 (meeting summary in ADAMS under Accession No. ML13281A201), to provide stakeholders a forum for discussing and asking questions about the June 2013 draft SFP study, provide an overview of the analysis conducted in this memorandum, and solicit feedback. Most of the individuals and organizations participating in the meetings said they favored expedited transfer of spent fuel to dry cask storage. Several points were raised by stakeholders, including the staff's focus on the seismic initiator in the SFP study, no consideration of partial SFP drainage interfering with air cooling, and limited alternatives being considered (e.g., not assessing low density, open frame rack designs). Each of these has been addressed by the conservative assumptions used in the enclosed analysis. The industry provided its views that spent fuel is continuing to be stored safely in SFPs. A transcript of the September 18, 2013 meeting is available in ADAMS under Accession No. ML13277A215. The staff considered this stakeholder feedback in the development of this memorandum. The staff also benefited from internal discussions, including a non-concurrence filed by a member of the staff. Addressing the issues raised by the non-concurrence process improved this memorandum, but the staff was not able to resolve all of the differing opinions offered (see Enclosure 2). Additionally, on October 2, 2013, the staff briefed the Advisory Committee on Reactor Safeguards (ACRS) on the results of its assessments and evaluations, as well as the resulting conclusions and recommendations. The staff is planning another briefing of the ACRS in December 2013. The ACRS is expected to provide a letter to the Commission in December 2013, regarding its review of the staff's assessment and its recommendations about whether regulatory action might be warranted and whether additional studies should be pursued.

Within this Tier 3 analysis, the staff has considered the agency's activities on the waste confidence generic environmental impact statement (GEIS) and rulemaking, and it has ensured that the availability of these documents and interactions with stakeholders are coordinated to facilitate the public's involvement in these activities. Although this Tier 3 analysis was not specifically referenced in the draft GEIS, those who prepared the draft GEIS were aware of the conclusions in this Tier 3 analysis, and the staff has coordinated this activity with the relevant sections of the draft GEIS. To facilitate the public's ability to provide input, a draft of the October 2013 SFP study was released for public review and comment on July 1, 2013. Additionally, the draft evaluation of this Tier 3 issue was released to the public on September 26, 2013, well before the draft GEIS public comment period ends on December 20, 2013.

Staff Recommendation

The staff's assessment concludes that the expedited transfer of spent fuel to dry cask storage would provide only a minor or limited safety benefit, and that its expected implementation costs would not be warranted. Therefore, the staff recommends that no further generic assessments[3] be pursued related to possible regulatory actions to require the expedited transfer of spent fuel to dry cask storage and that this Tier 3 Japan lessons-learned activity be closed.

SECY, please track.

Enclosures:
1. Regulatory Analysis for Japan Lessons Learned Tier 3 Issue on Expedited Transfer of Spent Fuel
2. Non-Concurrence Package 2013-013

cc: SECY
OCA
OGC
OPA
CFO

[3] The staff will confirm that the seismic risk for western nuclear power plant SFPs is consistent with the analysis in the enclosure at the completion of the NTTF Recommendation 2.1 seismic reevaluation activity.

REGULATORY ANALYSIS FOR
JAPAN LESSONS-LEARNED TIER 3 ISSUE ON
EXPEDITED TRANSFER OF SPENT FUEL

U.S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation

FOREWORD

On March 11, 2011, the Great Tōhoku earthquake and subsequent tsunami in Japan resulted in significant damage to the site of the Fukushima Dai-ichi nuclear power station. The spent fuel pools and the used fuel assemblies stored in the pools remained intact at the plant. Even so, the event led to questions about the safe storage of spent fuel. In a memorandum to the Commission entitled, "Updated Schedule and Plans for Japan Lessons Learned Tier 3 Issue on Expedited Transfer of Spent Fuel," dated May 7, 2013 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML13105A122), the staff outlined a plan for evaluating whether the U.S. Nuclear Regulatory Commission (NRC) should undertake a regulatory action to require the expedited transfer of spent fuel from pools to dry cask storage containers at U.S. nuclear power plants.

To determine if additional studies are needed to further assess potential regulatory action on expedited transfer, the staff has performed this regulatory analysis. The staff assessed the potential safety benefits by using the Commission's 1986 Safety Goal Policy Statement (Ref. 4). Then, to provide additional information to support the Commission's deliberations, the staff performed a cost-benefit analysis. The staff concluded that requiring the expedited transfer of spent fuel would provide only a minor or limited safety benefit (i.e., below the safety goal screening criteria), and that its expected implementation costs would not be warranted. The results of this analysis support the staff recommendation that the NRC conduct no further generic assessments on expedited transfer, and that this Tier 3 Japan lessons learned activity be closed. The NRC staff continues to believe, based on this analysis and previous studies that spent fuel pools provide adequate protection of public health and safety.

EXECUTIVE SUMMARY

The NRC evaluates within this analysis whether additional study of expedited transfer of spent fuel from spent fuel pools (SFPs) (i.e., expedited transfer) to dry cask storage might be warranted. This analysis was undertaken to support development of a technical basis for the program plan described in a memorandum to the Commission, "Updated Schedule and Plans for Japan Lessons-Learned Tier 3 Issue on Expedited Transfer of Spent Fuel," dated May 7, 2013 (Ref. 1). In the memorandum, the staff outlined a three-phase plan for evaluating if regulatory action should be pursued to require licensees to expedite transfer of spent fuel from SFPs to dry cask storage. The program plan calls for preparing this analysis under Phase 1 to help determine if additional study is warranted. If the results of Phase 1 indicate that additional study is warranted, Phases 2 and 3 of the program plan would be conducted to refine assumptions used in the analyses to determine whether any regulatory action is warranted. The Phase 1 screening analysis is documented in this regulatory analysis, and considers the results of the SFP study (SFPS) (Ref. 2), along with previous studies. For this analysis, the NRC evaluated the merits of additional research by comparing the status quo to a scenario in which expedited transfer would be required.

The SFPS provides consequence estimates of a hypothetical SFP accident initiated by a low likelihood seismic event at a reference plant for both a fully loaded (high-density) and minimally loaded (low-density) SFP. The SFPS contributed to the resolution of this Tier 3 issue by providing a measure of the change in potential consequences resulting from a change in spent fuel storage density for a reference plant. The staff completed a regulatory analysis in Appendix D of the SFPS, which indicates that expediting movement of spent fuel for the reference plant would provide only a minor or limited safety benefit, and that this benefit would be outweighed by the expected implementation costs. The staff's analysis herein expands the regulatory analysis in the SFPS by covering SFP designs used in the operating and decommissioned reactors in the United States.

To determine if additional studies are needed to further assess potential regulatory action on expedited transfer, the staff conducted a two-part analysis of expedited transfer. The staff first assessed the potential safety benefits by using the Commission's 1986 Safety Goal Policy Statement (Ref. 4). Although the regulatory analysis guidelines would normally allow the staff to stop the evaluation upon finding that the proposed action does not provide a sufficient safety enhancement to meet the threshold of the safety goal screening, the staff proceeded to perform a cost-benefit analysis to provide additional information for the Commission's consideration.

Whereas the SFPS addressed the consequences of a selected event at a reference plant, this analysis is expanded to consider a variety of possible initiating events and to determine whether expedited spent fuel transfer may be warranted at SFPs across the U.S. fleet of nuclear power plants and independent wet spent fuel storage facilities. The staff accounted for the differences in the SFPs by categorizing them into several groups with similar properties. The categorization process is further described in Section 4.1.1 of this regulatory analysis. The staff used conservative values for parameters in the base case analysis to ensure that effects of design, operational and other site variations among the licensed reactor fleet were encompassed. The base case was supplemented with low and high sensitivity calculations to address uncertainties in the analysis.

To the extent practicable, the staff used conservative estimates and assumptions to bound the variations in SFP parameters across the fleet for this analysis. This analysis determines

whether regulatory action may be appropriate, or whether additional generic studies are needed. In accordance with Phases 2 and 3 of the program plan, if the Commission directs additional studies, then the staff would refine the conservative assumptions used in this regulatory analysis to increase realism, and consider additional factors such as the risks associated with the transfer of spent fuel assemblies to casks, and storage of the casks in the associated storage facilities. These risks were not included in this study so as to bias the results in favor of taking regulatory action. The staff's judgment is that these refinements would likely reduce the benefit associated with expedited transfer, resulting in a more negative cost-benefit assessment.

The staff used the U.S. Geological Survey (USGS) 2008 model to evaluate seismic hazards at central and eastern U.S. (CEUS) nuclear power plant sites in this analysis. Although the USGS model considers sites in the western United States (including Columbia, Diablo Canyon, Palo Verde, and San Onofre), the staff has not performed the necessary analyses for these sites to include them in this analysis. Considering the robust designs of SFPs, especially in more seismically active areas in the western United States, the staff concludes that public health and safety are adequately protected. Upon completion of the Near-Term Task Force Recommendation 2.1 seismic reevaluation, the staff will confirm that the seismic risk for SFPs is consistent with the risk assumed in this analysis.

This analysis and the supporting references, in general, do not include events caused by sabotage. For nuclear power plants, security requirements are established to provide high assurance of adequate protection from radiological sabotage of the nuclear power plant reactor and SFP. The NRC continually monitors threat conditions and, as was done after the September 11, 2001 attacks, makes adjustments, as appropriate in the governing security requirements and in actions to oversee their effective implementation. Based on the staff's view that security issues are effectively addressed in the existing regulatory program, they are not part of this analysis.

In this analysis, the risks associated with a severe SFP accident at the plants studied are compared to the Safety Goal Policy Statement (Ref. 4) to determine if requiring the expedited transfer of spent fuel to dry cask storage would provide more than a minor safety benefit. Despite the large releases for some low probability accident progressions analyzed, the projected consequences indicate that there are no offsite early fatalities from acute radiation effects. The analysis also shows that the risk of an individual dying from cancer from the radioactive release is less than 0.76% of the Commission's Quantitative Health Objective of two in one million ($2x10^{-6}$) per year. The risks are similar between different spent fuel loading and mitigation scenarios because of modeled offsite protective actions that include evacuation, sheltering, relocation, and decontamination. Additionally, these individual risks are dominated by long-term exposures to very lightly contaminated areas for which doses are small enough for the areas to be considered habitable.

In addition, the staff conducted a cost-benefit analysis, which finds that the added costs involved with expedited transfer of spent fuel to dry cask storage to achieve the low-density SFP storage alternative are not warranted in light of the benefits from such expedited transfer. The combination of high estimates for important parameters assumed in some of the sensitivity cases presented in this analysis result in large economic consequences, such that, the calculated benefits from expedited transfer of spent fuel to dry cask storage for those cases outweigh the associated costs. However, even in these cases, there is only a limited safety benefit when using the QHOs and the expected implementation costs would not be warranted. In addition, in the staff's judgment, the various assumptions made in the analysis of the "base

case" result in an overall cost-benefit assessment that is appropriately conservative for a generic regulatory decision and justify using the "base case" as the primary basis for the staff's recommendation. Based on the generic assessment and the other considerations detailed in this analysis, the staff finds that additional studies are not needed to reasonably conclude that the expedited transfer of spent fuel to dry cask storage would provide only a minor or limited safety benefit (i.e., below the safety goal screening criteria), and that its expected implementation costs would not be warranted.

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ABBREVIATIONS AND ACRONYMS

ac	alternating current
ADAMS	Agencywide Documents Access and Management System
BLS	Bureau of Labor Statistics
BWR	boiling-water reactor
CEUS	central and eastern United States
CFR	*Code of Federal Regulations*
CoC	certificate of compliance
CPI-U	consumer price index—all urban consumer inflator
Cs	cesium
DOE	U.S. Department of Energy
DSC	dry storage cask systems
EPA	U.S. Environmental Protection Agency
EPRI	Electric Power Research Institute
FR	*Federal Register*
FTE	full-time equivalent
GMPE	ground motion prediction equation
GWd	gigawatt-day
ISFSI	independent spent fuel storage installation
LCF	latent cancer fatality
LNT	linear no-threshold
LOOP	loss of offsite power
MACCS2	MELCOR Accident Consequence Code System, Version 2
MELCOR	(not an acronym)
MTU	metric ton heavy metal or metric ton uranium
MW_t	megawatt thermal
NGO	non-government organizations
NPV	net present value
NRC	Nuclear Regulatory Commission
NTTF	Near-Term Task Force
OCP	operating cycle phase
OMB	Office of Management and Budget
ORIGEN	(not an acronym)
PAG	protective action guides
PGA	peak ground acceleration
PRM	petition for rulemaking
PSHA	probabilistic seismic hazard assessment
PWR	pressurized water reactor
RA	regulatory analysis
SCALE	(not an acronym)

SFP	spent fuel pool
SOARCA	State-of-the-Art Reactor Consequence Analyses
SRM	staff requirements memorandum
USGS	U.S. Geological Survey
VSL	value of a statistical life

1. INTRODUCTION

The NRC evaluates within this regulatory analysis whether additional study of expedited transfer of spent fuel from spent fuel pools (SFPs) (i.e., expedited transfer) to dry cask storage might be warranted. The NRC evaluated the merits of additional research by comparing the status quo to one in which expedited transfer would be required. The staff assessed the potential safety benefits of requiring expedited transfer by using the Commission's 1986 Safety Goal Policy Statement (Ref. 4). Then, to provide additional information to support the Commission's deliberations, the staff performed a cost-benefit analysis of requiring expedited transfer. This work was conducted in accordance with the program plan described in a memorandum to the Commission, "Updated Schedule and Plans for Japan Lessons-Learned Tier 3 Issue on Expedited Transfer of Spent Fuel," dated May 7, 2013 (Ref. 1).

In conducting the analyses described herein, the staff considered the results of the Spent Fuel Pool Study (SFPS) (Ref. 2) along with previous studies and operating experience. The SFPS analyzed the risks and consequences of postulated spent fuel pool accidents for a reference plant (a General Electric (GE) Type 4 boiling-water reactor (BWR) with a Mark I containment). Since seismic events dominate SFP damage risk, seismic events were modeled. The other risk contributors, such as equipment failures and human errors, were derived from previous studies and were factored into the analysis. Mechanistic modeling was applied to develop the source term for the SFP accident since it differs from that associated with severe core damage accidents. The consequences of a SFP accident, which results in the loss of cooling or the loss of pool water inventory and a radiological release, are dominated by the long-lived isotopes, such as cesium. The results of the SFPS showed that the overall level of safety with respect to spent fuel storage in a SFP currently achieved at the reference plant is high and that the level of risk at the reference plant is very low. The staff therefore found that adequate protection is assured. Additionally, the SFPS included a regulatory assessment that considered various initiating events and concluded that the incremental safety benefit associated with expedited transfer of spent fuel at the reference plant was minor, far from the threshold that the NRC uses to inform its decisionmaking, and was also not warranted in light of the added costs involved with expediting the movement of spent fuel from the pool to achieve low-density fuel pool storage. The regulatory analysis is included in Appendix D of the SFPS. The results of the SFPS are consistent with earlier research conducted over the last several decades, as summarized in NUREG 1353, "Regulatory Analysis for the Resolution of Generic Issue 82, Beyond Design Basis Accidents in Spent Fuel Pools," dated April 1989; in NUREG/CR 6451, "A Safety and Regulatory Assessment of Generic BWR and PWR [pressurized-water reactor] Permanently Shutdown Nuclear Power Plants," dated April 1997, and in NUREG 1738, "Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants," dated February 2001.

The SFPS was an important input to this analysis but is not the sole technical study or basis for the following analysis and related findings. The SFPS addressed the consequences of a selected seismic event that could result in the loss of SFP integrity at a reference plant. The staff's analysis herein expands the regulatory analysis in the SFPS by covering SFP designs used in the operating and decommissioned reactors in the United States (as used throughout this document, the operating reactor fleet includes the recently licensed but not yet operating AP1000 plants).

This Tier 3 analysis assesses whether the proposed expedited spent fuel transfer alternative would have more than a minor safety benefit, and in doing so the staff uses the quantitative

health objectives (QHOs). The QHOs are used as a surrogate for the safety goal as outlined in the Commission's Safety Goal Policy Statement (Ref. 4). A further discussion of the basis and background for using the QHOs in assessing SFP accidents is included in Section 3 of this regulatory analysis. The staff relied on information from past studies, the recently completed SFPS, and operating experience in conducting this analysis.

To determine if additional studies are needed to further assess whether expedited transfer should be required, the staff conducted a two-part analysis. The staff first assessed the potential safety benefits of requiring expedited transfer using the Commission's 1986 Safety Goal Policy Statement to conduct a safety goal screening evaluation. Although the agency's guidance would normally allow the staff to stop the evaluation upon determining that the proposed action does not provide a sufficient safety enhancement to meet the threshold of the safety goal screening, the staff proceeded to perform a cost benefit analysis (summarized below) to provide the Commission additional information.

In addition to safety benefits, the staff's cost-benefit analysis considers wider societal measures, such as averted offsite property damage. The staff developed estimates of benefits and costs, which are quantified, when possible, together to conclude whether requiring expedited transfer would be cost-beneficial[1].

Within this cost-benefit analysis, the staff developed a base case that generally used conservative assumptions for key parameters such as conditional probabilities of pool failures and zirconium fires to increase the calculated net benefits of the expedited transfer of spent fuel alternative for each SFP grouping and to generally bound the parameters that vary among spent fuel pools. The benefits calculated for these base case evaluations provide only a minor or limited safety benefit that is far from the threshold that the NRC uses to inform its regulatory decisionmaking. In addition, the benefits calculated for the base case evaluations are less than the estimated costs for expedited transfer of spent fuel. There are some plants that for a particular parameter are not bounded by the base case. However, the amount of conservatism used in the other parameters overwhelm the slight non-conservatism in the particular outlying parameter. Therefore, the overall results of the base case is conservative for all plants. This analysis approach greatly simplifies the analysis and precludes the need to model each plant in detail. To provide additional information for the Commission's consideration, the staff also analyzed additional cases where the key input parameters are varied to provide a low to high estimate of the calculated benefits. In addition, to identify the specific effect of certain parameters, the staff performed sensitivity studies where only one parameter was varied from a low to high value. Sensitivity studies were conducted on key factors such as the dollars per person-rem conversion factor and consideration of consequences beyond 50 miles (80 kilometers) to measure each attribute's effect upon the overall result. The sensitivity of the dollars per person-rem conversion factor is important because it provides the Commission with additional information to inform regulatory decisionmaking. The cost-benefit analysis used key insights from operating experience and the recent SFPS, such as the plant damage state for seismic events, probability of a release for specific pool damage states, and the expected amount and type of radioactive material released.

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[1] Cost-beneficial means that the benefits of the proposed action are equal to, or exceed, the costs of the proposed action.

1.1 Statement of the Problem

The federal government's decision to stop work on a deep geologic repository at Yucca Mountain, and the events in Japan following the March 2011 earthquake, have rekindled public and industry interest in understanding the consequences from postulated accidents associated with high-density SFP storage, and the relative benefits of low-density SFP storage. In response to these events, as discussed in a memorandum to the Commission, "Updated Schedule and Plans for Japan Lessons-Learned Tier 3 Issue on Expedited Transfer of Spent Fuel" (Ref. 1), the staff determined that it should confirm whether high-density SFP configurations continue to provide adequate protection and assess whether any safety benefits (or detriments) would occur in requiring the expedited transfer of spent fuel to dry cask storage.

U.S. nuclear power plants store spent fuel in pools for varying periods of time using a high-density configuration. Various risk studies (such as NUREG-1738, "Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants," February 2001 (Ref. 5)) have shown that storage of spent fuel in a high-density configuration in SFPs is safe, and that the risk of accidental release of a significant amount of radioactive material to the environment is low. These studies used simplified and sometimes bounding assumptions and models to characterize the likelihood and consequences of beyond-design-basis SFP accidents.[2] As part of the 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-September 11 security assessments, the NRC in 2009 issued 10 CFR 50.54(hh)(2) (Ref. 6) as a final rule, which requires reactor licensees to develop and implement strategies intended, in part, to maintain or restore SFP cooling capabilities in the event of explosions or fires caused by beyond-design-basis events.

The NRC had previously restated its views on the safety of spent fuel stored in high-density configurations in a response to Petition for Rulemaking (PRM)-51-10 (Ref. 7) and PRM-51-12 (Ref. 8) (73 FR 46204, August 8, 2008), and in revising NUREG-1437, "Generic Environmental Impact Statement for License Renewal of Nuclear Plants, Draft Report for Comment" (Ref. 9). However, the NRC's position relies, in part, on the findings of the aforementioned security assessments, which are not publicly available.

1.2 Overview of the Safety Goal Screening Evaluation

As part of the NRC staff's regulatory analysis, the risks associated with a severe SFP accident at the plants studied are compared to the Commission's 1986 Safety Goal Policy Statement (Ref. 4) to determine if requiring the expedited transfer of spent fuel to dry cask storage would provide more than a minor safety benefit. Despite the large releases for some low probability accident progressions analyzed, the projected consequences indicate there are no offsite early fatalities from acute radiation effects. The analysis also shows that the risk of an individual dying from cancer from the radioactive release is less than 0.76% of the Commission's QHO of two in one million ($2x10^{-6}$) per year. The risks are similar between different spent fuel loading and mitigation scenarios because of modeled offsite protective actions that include evacuation, sheltering, relocation, and decontamination. Additionally, these individual risks are dominated

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2 An overview of previous studies is provided in section 10.2 to the SFPS (Ref. 2).

by long-term exposures to very lightly contaminated areas for which doses are small enough for the areas to be considered habitable. The QHOs are used as a surrogate for the safety goal as outlined in the Commission's 1986 policy statement. Section 3 below discusses the safety goal screening evaluation in more detail.

1.3 Overview of the Cost-Benefit Analysis

This analysis uses information contained in the SFPS for its structural analysis and related damage characterization, its accident progression analysis, and its offsite consequences analysis. These results are supplemented with results from previous studies and conservative assumptions in the cost-benefit analysis to broaden the assessment to generically address the SFP risk at multiple facilities.

This analysis calculates the potential benefit per reactor year resulting from expedited fuel transfer by comparing the safety of high-density fuel pool storage relative to low-density fuel pool storage and related alternatives. The comparison uses the initiating frequency and consequences from the SFPS as an indicator of any changes in the NRC's understanding of safe storage of spent fuel following a beyond-design-basis seismic event. The staff also used calculated results from previous SFP studies (i.e., NUREG-1353 and NUREG-1738) to extend the applicability of this evaluation to include other initiators, which could challenge SFP cooling or integrity and incorporated inputs representing the range of U.S. SFP characteristics to extend the analysis applicability to SFPs within other U.S. reactor designs.

Within this cost-benefit analysis, the staff developed a base case that generally used conservative assumptions for key parameters such as conditional probabilities of pool failures and zirconium fires to increase the calculated net benefits of the expedited transfer of spent fuel alternative for each SFP grouping and to generally bound the parameters that vary among spent fuel pools. The benefits calculated for these base case evaluations provide only a minor or limited safety benefit that is far from the threshold that the NRC uses to inform its regulatory decisionmaking. In addition, the benefits calculated for the base case evaluations are less than the estimated costs for expedited transfer of spent fuel. There are some plants that for a particular parameter are not bounded by the base case. However, the amount of conservatism used in the other parameters overwhelms the slight non-conservatism in the particular outlying parameter. Therefore, the overall results of the base case are conservative for all plants. This analysis approach greatly simplifies the analysis and precludes the need to model each plant in detail. To provide additional information for the Commission's consideration, the staff also analyzed additional cases where the key input parameters are varied to provide a low to high estimate of the calculated benefits. In addition, to identify the specific effect of certain parameters, the staff performed sensitivity studies where only one parameter was varied from a low to high value. Section 4 below discusses the staff's cost-benefit analysis in more detail.

2. ANALYSIS OF IDENTIFIED ALTERNATIVE

The U.S. Nuclear Regulatory Commission (NRC) considered the regulatory baseline and one alternative to change this baseline as discussed below. The baseline is used to estimate the incremental costs of the alternative.

2.1 Regulatory Baseline—Maintain the Existing Spent Fuel Storage Requirements

The baseline would be maintained if the Commission decides not to require the expedited transfer of spent fuel from pools to dry cask storage, but to continue with the NRC's existing licensing requirements for spent fuel storage. Spent fuel must now be moved into dry cask storage only as necessary to accommodate fuel assemblies being removed from the core during refueling operations. Fuel storage in the spent fuel pool (SFP) is managed to maintain sufficient empty space in the pool for removal of one full core of reactor fuel in case of emergencies (referred to as full core discharge) or other operational contingencies. The NRC also assumes in this analysis that all applicable requirements and guidance to date have been implemented, there are no unevaluated degraded or nonconforming conditions, and no implementation is assumed for related generic issues or other staff requirements or guidance that is unresolved or still under review.

The baseline condition is the storage of spent fuel in high-density racks[3] in the SFP, a relatively full SFP, and compliance with all current regulatory requirements. The regulatory requirements include design features intended to prevent a substantial loss in water inventory under accident conditions and those requirements for emergency abnormal conditions associated with the following[4]:

- Title 10 of the *Code of Federal Regulations* (10 CFR) 50.54(hh)(2) (Ref. 6) with respect to spent fuel configuration and SFP preventive and mitigative capabilities

For the purpose of evaluating the potential benefits of expedited transfer of spent fuel to dry cask storage, this analysis used a conservative approach by crediting successful mitigation for the low-density SFP alternative and assumed no successful mitigation for the high-density SFP storage regulatory baseline. Furthermore, because SFPs have limited available storage, even after licensees expanded their storage capacity using high-density storage racks, the current practice of transferring spent fuel to dry storage in accordance with 10 CFR Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater Than Class C Waste," (Ref. 12) is assumed to continue.[5]

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[3] Most nuclear power plant SFPs were originally designed for temporary storage of spent fuel. Starting in the 1980s, most pools were "re-racked" to use hardware that stores the assemblies in a more closely spaced arrangement, thus allowing the storage of more assemblies in a high-density configuration.

[4] The following regulatory requirements apply to operating power reactors considered in this analysis.

[5] Maintenance of the existing SFP storage requirements would not limit the Commission's authority to add new requirements or update regulatory guidelines, as necessary. These actions and activities are a part of the regulatory baseline. However, these activities would be pursued as separate regulatory actions to resolve particular technical issues. In the baseline case, the NRC would take no

The NRC has required through orders that licensees enhance their ability to respond to beyond-design-basis events. The additional capabilities to do so were not quantitatively considered in this analysis. The orders include:

- Order EA-12-049 (Ref. 10) that requires licensees to develop, implement, and maintain guidance and strategies to maintain or restore core cooling, containment, and SFP cooling capabilities following a beyond-design-basis external event

- Order EA-12-051 (Ref. 11) that requires licensees to install reliable means of remotely monitoring wide-range SFP levels to support effective prioritization of event mitigation and recovery actions in the occurrence of a beyond-design-basis external event

2.2 Expedited Transfer Alternative—Low-Density Spent Fuel Pool Storage

This proposed alternative would require older spent fuel assemblies[6] to be expeditiously moved from SFP storage to dry cask storage beginning in year 2014, to achieve and maintain a low-density loading of spent fuel in the existing high-density racks as a preventive measure. Because of the low-density SFP loading, this alternative has less long-lived radionuclide inventory in the SFP, a lower overall heat load in the pool, and a slight increase in the initial water inventory that displaces the removed spent fuel assemblies.

Because of the uncertainty over the availability of a spent fuel repository, many plants have plans to establish onsite storage capacity (in-pool capacity and dry storage) sufficient to store all of the spent fuel discharged over the operating life of the plant until repository capacity becomes available. As of early 2013, all but 5 of the 65 U.S. sites with operating nuclear power reactors had either built or were seeking licenses to build dry storage facilities (Ref. 19).

Recently, some non-government organizations (NGOs) concerned about the hazards of nuclear power indicated preference for onsite dry storage instead of reprocessing or central storage. Those NGOs have also called for spent fuel to be placed in onsite dry casks after, at most, five years of cooling in spent-fuel pools.

There are cost and risk impacts associated with the transfer of spent fuel from the SFP to cask storage and during long-term cask storage.[7] These cost and risk impacts reduce the overall net benefit of this alternative in relation to the regulatory baseline. However, the added risks of handling and moving casks were conservatively not included in this analysis to maximize the delta benefit of the expedited transfer alternative.

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action to require facilities to expedite the movement of spent fuel to achieve low-density loading in the SFP.

6 Older spent fuel assemblies are those that have been placed in the SFP to cool for at least five years after discharge from the reactor core.

7 EPRI report TR-1021049 (Ref. 17) assesses the cost and risk impacts from a worker dose perspective associated with transfer of spent nuclear fuel from SFPs to dry storage after five years of cooling. The report concludes that expedited fuel movement would result in an increase cost to the U.S. nuclear industry of $3.6 billion, with the increase primarily related to the additional capital costs for new casks and construction costs for the dry storage facilities.

3. SAFETY GOAL SCREENING EVALUATION

The Commission has directed that NRC's regulatory actions affecting nuclear power plants be evaluated for conformity with NRC's Policy Statement on Safety Goals for the Operations of Nuclear Power Plants (Ref. 4). The Safety Goal Policy Statement sets out two qualitative safety goals and two quantitative objectives. Both the goals and objectives apply only to the risks to the public from the accidental or routine release of radioactive materials from nuclear power plants.

The two qualitative safety goals are as follows:

(1) Individual members of the public should be provided a level of protection from the consequences of nuclear power plant operation such that individuals bear no significant additional risk to life and health.

(2) Societal risks to life and health from nuclear power plant operation should be comparable to or less than the risks of generating electricity by viable competing technologies and should not be a significant addition to other societal risks.

The following quantitative health objectives are to be used in determining achievement of the above safety goals:

(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 1/10 of 1 percent (0.1 percent) of the sum of prompt fatality risks resulting from other accidents to which members of the U.S. population are generally exposed.

(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 1/10 of 1 percent (0.1 percent) of the sum of cancer fatality risks resulting from all other causes.

An important part of the implementation of the policy statement is its incorporation into the NRC's processes for evaluating possible changes in regulations or other requirements imposed on licensees. Within the NRC's Regulatory Analysis Guidelines, the safety goal screening evaluation is designed to answer when a regulatory requirement should not be imposed generically on nuclear power plants because the residual risk is already acceptably low. This evaluation is intended to eliminate some proposed requirements from further consideration independently of whether they could be cost-beneficial. Note that performing a safety goal screening evaluation requires judgment by the NRC staff and Commission as to whether the evaluation provides an unreasonable finding on whether a proposed action provides more than a marginal safety improvement.

The Safety Goals for the Operation of Nuclear Power Plants: Policy Statement defines the early fatality area calculation as that within 1.6 kilometers (1 mile) from the site boundary. The prompt fatality QHO represents a 5×10^{-7} per year objective for an average individual within 1 mile ("Safety Goals for Nuclear Power Plant Operation," NUREG-0880, Rev. 1, issued May 1983.) (Ref. 14)

The second quantitative objective of the policy relates to ensuring that the cancer fatality risks from nuclear power plant operations remain a small fraction of the overall cancer risks from all

causes. The cancer fatality QHO represents a $2x10^{-6}$ per year objective for an average individual within 16 kilometers (10 miles) (NUREG-0880). The staff assessed the criteria based on recent data (http://www.cancer.org/research/cancerfactsfigures/index), and found that the total fatality rate from cancer in the United States is 580,350 per 315,747,500 persons (http://www.census.gov/popclock/) or a risk of $1.84x10^{-3}$ per year. 1/10 of 1 percent of this value results in a safety goal of $1.84x10^{-6}$ per year (i.e., little changed from the value in NUREG-0880).

Using the bounding frequency of damage to the spent fuel of $3.46x10^{-5}$ per year[8], which considers all initiators that could challenge SFP cooling or integrity, and the estimates from the SFPS for conditional individual latent cancer fatality risk within a ten-mile radius of $4.4x10^{-4}$ yields a conservative high estimate of individual latent cancer fatality risk of $1.52x10^{-8}$ cancer fatalities per year. This calculated value of $1.52x10^{-8}$ individual latent cancer fatality risk per reactor-year associated with a SFP accident is less than one percent of the $1.84x10^{-6}$ per year societal risk goal value based on the calculation area specified in the Safety Goal Policy Statement.[9] The factors leading to this low likelihood, as discussed above, are summarized in Figure 1.

Comparing the results of this analysis to the NRC Safety Goal Policy Statement involves important limitations.

(1) First, the safety goal is intended to encompass all accident scenarios on a nuclear power plant site, including those involving reactors and spent fuel. This analysis does not examine reactor scenarios that would need to be considered, although the analysis does consider the most important contributors to SFP risk. As a result, comparison of the calculated individual latent cancer fatality (LCF) risk to the NRC Safety Goal Policy Statement is incomplete. However, it is intended to show that SFP risk is less than one percent of the individual LCF risk that corresponds to the overall or total safety goal for latent cancer fatalities for a nuclear power plant site. It is unlikely that the additional reactor accident scenarios would contribute significantly to overall risks and introduce significant challenges to the Commission's Safety Goal Policy Statement.

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8 See Table 43 in Appendix C for frequencies of all groups. The value of the highest frequency of group 4 is 3.46×10^{-5} per year is and is greater than the frequency of any of the other groups.

9 The safety goals and related QHOs were developed to assess aggregate risks and to be used for making decisions on rulemakings or other major agency actions. It is necessary to keep this in mind when using the QHOs to evaluate specific issues or plant specific concerns. In this case, the risks associated with high-density loadings in spent fuel pools contribute only a small fraction of the overall societal risk goal and so the staff concludes that the issue would not result in additional risks that would cause the cumulative risk of nuclear power to exceed the established safety goals.

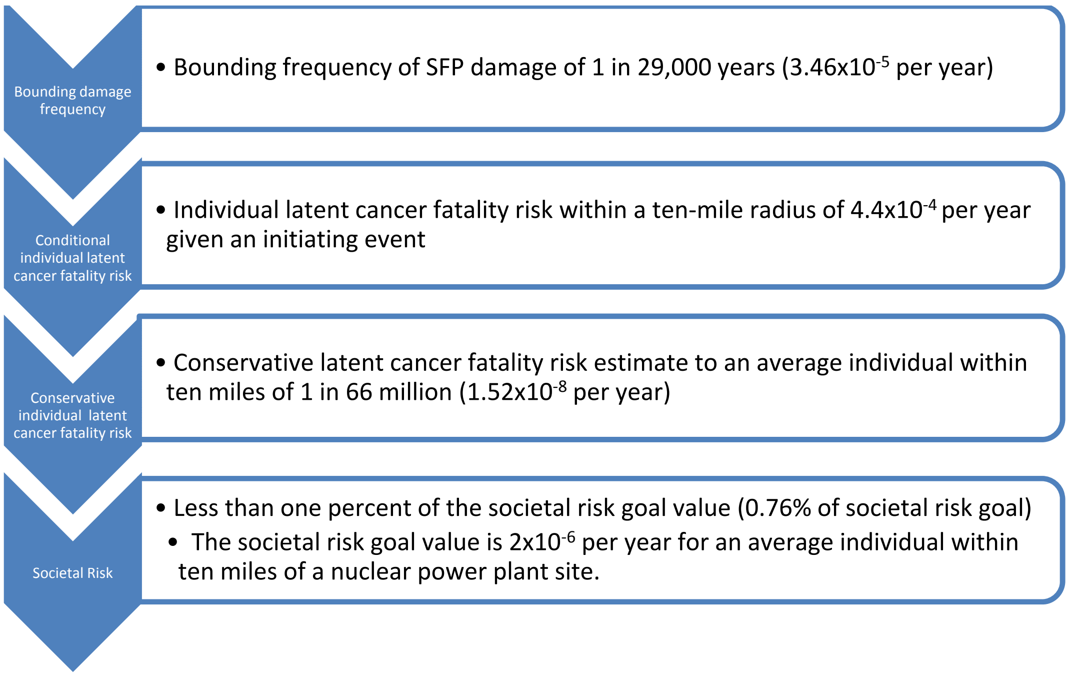

Figure 1 Factors Used in Evaluating Societal Risk

(2) The QHOs effectively establish expectations related to the frequency of severe accidents associated with nuclear reactors and the potential for release of radioactive materials from an operating reactor core. Previous NRC evaluations of SFPs, including NUREG-1353 and NUREG-1738, compared the estimated risks from SFP accidents to the QHOs as part of the rationale for determining appropriate regulatory actions. Some considerations in comparing SFP risks to the QHOs are that the potential consequences of a SFP accident can exceed those of reactor accidents in terms of the amount of long-lived radioactive material released, the land area affected, and the economic consequences. The safety goal relates the risks to an individual from nuclear power in comparison to other risks that an individual faces. The staff uses the safety goal in regulatory decisionmaking processes as a measure of health consequences to determine if a potential action provides a substantial safety improvement. Although a SFP accident might affect larger areas and more people than a reactor accident, protective actions such as relocation of the public and decontamination of affected areas would result in the risks to individuals beyond ten miles to be similar to individuals located closer to the plant. For this reason, the staff uses the existing QHOs for determining whether the substantial safety enhancement threshold is met.

(3) A possible issue with use of the existing guidance and QHOs for SFP accidents relates to the inclusion of emergency planning (i.e., evacuation, sheltering, and relocation of populations) within the analyses. Given that the same measures would be taken for releases following accidents involving high-density or low-density spent fuel pools, the difference in risks to individuals does not increase as much as might be expected from the large differences in the amount of radioactive material released and populations affected. So while the risk of individuals, either close to or far from the plant, remains below the QHOs, the total or cumulative radiation dose to the population might be higher for a SFP accident than for a reactor accident. This would be in large part due to low doses to larger populations associated with the potentially expanded land areas affected

by a SFP accident. The discussions of larger affected populations and areas regarding SFP accidents, as compared to reactor accidents, leads to questions about the use of QHOs as a screening metric as well as questions about underlying Commission policies on estimating the health effects of ionizing radiation (i.e., linear no-threshold model).

The significant difference between the calculated consequences of a SFP accident and a reactor accident has led some stakeholders to propose alternate performance measures to help in the decisionmaking process. Such measures could include a revised consideration of economic consequences, collective dose to populations, or other estimates that reflect the large consequences and reduce the influence of the low event frequencies and implementation of protective actions in assessing the overall societal risks associated with SFP accidents. However, the Commission has previously directed that these performance measures should be consistent with the overall safety goals the Commission policy established and should not be so conservative that it creates a de facto new policy.[10] In addition, the Commission stated in the staff requirements memorandum for SECY-12-0110, "Consideration of Economic Consequences within the U.S. Nuclear Regulatory Commission's Regulatory Framework," that developing guidance for other regulatory applications should be limited and should be resourced as a lower priority than applying State-of-the-Art Reactor Consequence Analyses (SOARCA) insights and improving guidance and analysis tools.

The development of surrogate measures for SFPs could be useful if the conditional probability of a significant SFP accident is very high for particular event scenarios (a so-called cliff-edge effect). Although the staff has used various conservative assumptions in this assessment in order to estimate the potential benefits of reducing the density of spent fuel stored in pools, the expected ability of pools to retain their integrity and the availability of mitigation capabilities leads the staff to conclude that exceeding design basis values associated with SFPs are unlikely to result in such a cliff-edge effect and that the frequency of damage to stored fuel is appropriately low to satisfy overall societal risk goals. Therefore, the staff has not identified this as an area for which it needs to develop new methodologies, guidance, or criteria. In the SRM for SECY-12-0110, the Commission directed the staff to proceed with improvements to the guidance for estimating offsite economic costs. The staff is continuing its efforts and planning related to the SRM and is scheduled to provide the Commission with a paper in December 2013. Factors considered likely to change as a result of the staff's activities (e.g., dollars per person-rem conversion factor) have been addressed in this evaluation through the presentation of additional cases and sensitivity studies.

The staff has concluded that the continued operation of nuclear power plants with high-density loadings in their SFPs does not challenge the NRC's safety goals or related QHOs. Therefore, in the staff's judgment, a regulatory action to require reducing the inventory of spent fuel in the pools would provide no more than a minor safety improvement.

[10] Commission Guidance on Implementation of the NRC's Safety Goal Policy," memorandum from the Secretary of the Commission to the EDO, dated November 6, 1987.

4. COST-BENEFIT ANALYSIS

To support Commission's deliberations, the staff conducted a cost-benefit analysis using current policies and guidance. Recently the staff completed the SFPS, producing updated consequence estimates which were used in this analysis. The SFPS provides consequence estimates of a hypothetical SFP accident initiated by a low likelihood seismic event at a reference plant for both a fully loaded (high-density) and minimally loaded (low-density) SFP. Appendix D of the SFPS evaluates whether the benefits would be cost-justified and substantial enough at the reference plant to require a change from high- to low-density storage configurations in the SFP.

To determine whether further study of expedited spent fuel transfer may be appropriate, the staff herein conducts a more expansive analysis using insights from the SFPS and previous studies. This generic analysis addresses the different types of SFPs at U.S. nuclear power plants. The process the staff used to conduct the generic analysis is described in the following sections and referenced appendices.

4.1 Spent Fuel Pool Characteristics and Operation Strategies

4.1.1 Spent Fuel Pool Groupings

Based on the variation in SFP configurations, rack designs, and SFP capacities provided in detail in Appendix A, the following groupings were created for use in this analysis.

Table 1 Average Reactor Operation Expectancy by Grouping

SFP Group No.	Description	No. of reactor units	No. of spent fuel pools	Average Year when the Reactor Operating License Expires
1	BWR Mark I and Mark II with nonshared SFPs	31	31	2037
2	PWR and Mark III with nonshared SFPs	49	49	2040
3	AP1000 SFPs	4	4	2078
4	Reactor units with shared SFPs	20	10	2038
5	SFPs located below grade[1]	(included in Group 2 numbers)		
6	Decommissioned plants with spent fuel stored in pool[2,3]	7	6	N/A
7	Decommissioned plants with fuel stored in an ISFSI using dry casks	21	N/A	N/A

1. Group 5 is a special set of currently operating PWRs where damage to the pool structure would not result in a rapid loss of water inventory.
2. The Zion 1 and 2 decommissioned reactor units share a single SFP.
3. The GE-Hitachi Morris wet ISFSI site is included in Group 6.

This cost-benefit analysis focuses on the first four groups identified in Table 1. Group 5 SFPs are excluded from the analysis because they are a special set of SFPs that are less susceptible to the formation of small or medium leaks due to the absence of open space around the pool liner and concrete structure. The spent fuel in Group 6 SFPs are no longer receiving discharged fuel following reactor decommissioning and several plants had extended plant

outages before announcing cessation of plant operation. The spent fuel in Group 7 is already in dry cask storage.

4.1.2 **Operation Strategies**

The operation strategies include the interim storage operations to expand onsite storage and cask loading strategies; these strategies are provided in detail in Appendix B.

4.2 Estimation and Evaluation of Costs and Benefits

This section discusses how the costs and benefits of the proposed alternative are evaluated and presented relative to the baseline. Ideally, all costs and benefits are converted into monetary values. The total of costs and benefits are then algebraically summed to determine whether the difference between the costs and benefits is a positive benefit. However, in some cases the assignment of monetary values to benefits is not provided because meaningful quantification is not possible.

4.2.1 **Identification of Affected Attributes**

This section identifies the factors within the public and private sectors that the expedited transfer are expected to affect. These factors are classified as attributes using the list of potential attributes provided by the NRC in Chapter 5 of its Regulatory Analysis Technical Evaluation Handbook (NUREG/BR-0184) (Ref. 15). The basis for selecting each attribute is presented below.

Affected attributes are the following:

- Public Health (Accident). This attribute measures expected changes in radiation exposure to the public caused by changes in accident frequencies or accident consequences associated with the proposed action (i.e., delta risk). The expected changes in radiation exposure are measured over a 50-mile (80-kilometer) radius from the plant site. The dose to the public is from reoccupation of the land and other activities following a severe accident. In addition, the dose to the public includes the occupational dose to workers for cleanup and decontamination of the contaminated land offsite.

- Occupational Health (Accident). This attribute measures occupational health effects, both immediate and long-term, associated with site workers because of changes in accident frequency or accident consequence. The short-term occupational exposure related to the accident occurs at the time of the accident and during the immediate management of the emergency and during decontamination and decommissioning of the onsite property. The radiological occupational exposure resulting from cleanup and refurbishment or decommissioning activities of the damaged facility to occupational workers are found within the long-term occupational exposure.

- Occupational Health (Routine). This attribute accounts for radiological exposures to workers during normal facility operations (i.e., nonaccident situations). These occupational exposures occur during dry storage cask (DSC) loading and handling activities; ISFSI operations, maintenance, and surveillance activities; and preparing to ship the spent fuel offsite.

This attribute represents an estimate of health effects incurred during normal facility operations so accident probabilities are not relevant. As is true of other types of exposures, a net decrease in worker exposures is taken as a positive benefit; a net increase in worker exposures is taken as a negative benefit.

- Offsite Property. This attribute measures the expected total monetary effects on offsite property resulting from the proposed action. Changes to offsite property can take various forms, both direct, (e.g., land, food, and water) and indirect (e.g., tourism). This attribute is typically the product of the change in accident frequency and the property consequences from the occurrence of an accident.

  The offsite property costs are any property consequences resulting from any radiological release from the occurrence of an accident. Normal operational releases and those releases before severe accident are outside the scope of this cost-benefit analysis.

- Onsite Property. This attribute measures the expected monetary effects on onsite property, including replacement power costs, decontamination, and refurbishment costs, from the proposed action. There are two forms of onsite property costs that are evaluated. The first type is the cleanup and decontamination costs for the damaged unit. The second type is the cost to replace the energy from the damaged or shutdown units.

- Industry Implementation. This attribute accounts for the projected net economic effect on the affected licensees to implement the mandated changes. Costs include procedural and administrative activities. Additional costs above the regulatory baseline are considered negative and cost savings are considered positive.

- Industry Operation. This attribute accounts for the projected net economic effect caused by routine and recurring activities required by the proposed alternative on all affected licensees.

- NRC Implementation. This attribute accounts for the projected net economic effect on the NRC to place the proposed alternative into operation. NRC implementation costs and benefits incurred in addition to those expected under the regulatory baseline are included. Additional rulemaking, policy statements, new or expedited revision of guidance documents, and inspection procedures are examples of such costs.

- NRC Operation. This attribute accounts for the projected net economic effect on the NRC after the proposed action is implemented. Additional inspections, evaluations, or enforcement activities are examples of such costs.

Attributes that are not expected to be affected under any of the alternatives include the following: public health (routine), other government, general public or antitrust considerations, safeguards and security considerations, regulatory efficiency, improvements in knowledge, and environmental considerations addressing section 102(2) of the National Environmental Policy Act of 1979.

4.2.2 **Methodology for Evaluation of Benefits and Costs**

This section describes the process used to evaluate benefits and costs associated with the proposed alternatives. The benefits (values) include desirable changes in affected attributes (e.g., monetary savings and improved security and safety). The costs (impacts or burdens) include undesirable changes in affected attributes (e.g., increased monetary costs and decreased security and safety).

The cost-benefit analysis methodology is specified by various guidance documents. The two documents that govern the NRC's voluntary regulatory analysis process are NUREG/BR-0058, Revision 4, "Regulatory Analysis (RA) Guidelines of the U.S. Nuclear Regulatory Commission," dated September 2004 (RA Guidelines) (Ref. 3), and NUREG/BR-0184, "Regulatory Analysis Technical Evaluation Handbook," dated January 1997 (RA Handbook) (Ref. 15). The analysis identifies all attributes impacted by the proposed alternative and analyzes them either quantitatively or qualitatively.

For the quantified cost-benefit analysis, the NRC staff develops expected values for each cost and benefit. The expected value is the product of the probability of the cost or benefit occurring and the consequences that would occur assuming the event happens. For each alternative, the staff first determines the probabilities and consequences for each cost and benefit, including the year the consequence is incurred. The NRC staff then discounts the consequences in future years to the current year of the regulatory action for purposes of evaluating benefits and costs (i.e., providing a net present value). Finally, the NRC staff sums the costs and the benefits for each alternative and compares them.

After performing a quantitative regulatory analysis, the NRC staff adds attributes that could only be qualified.[11] Based on the qualification of each attribute, uncertainties, sensitivities, and the quantified costs and benefits, the staff provides a recommendation for each alternative. If the benefits, both quantified and qualified, are greater than the quantified and qualified costs, then the staff recommends the alternative be implemented. If the benefits, both quantified and qualified, are less than the quantified and qualified costs, then the staff recommends the alternative not be implemented.[12]

There are a number of tables presented throughout this analysis. Generally, the tables include the SFP group[13], the case, the dose averted, the dose conversion factor, and the benefits/costs/cost offsets provided based on the net present value (NPV)[14]. There are two formats that the case information is presented in the tables. In one format, the information is

[11] See the NRC's Regulatory Analysis Technical Evaluation Handbook*,* Section 4.3, "Estimation and Evaluation of Values and Impacts" (Ref. 15).

[12] See the NRC's Regulatory Analysis Technical Evaluation Handbook*,* Section 4.5, "Decision Rationale" (Ref. 15). Nonquantifiable attributes can only be factored into the decision in a judgmental way; the experience of the decisionmaker will strongly influence the weight that they are given. Qualitative attributes may be significant factors in regulatory decisions and should be considered, if appropriate.

[13] Information on the SFP groups is found in Section 4.1.1 and Appendix E.

[14] Information on net present value is found in Appendix C, Section C.1.3 and Appendix D, Section D.1.

presented as low estimate, base case, and high estimate. In the other format, the base case evaluations are presented as Expedited Transfer Alternative–Low-density storage for each SFP group.

The dose averted and the dose conversion factors are only provided in tables that relate to health benefits. The dose averted is the amount of probability-weighted dose (i.e., risk) that is prevented due to the alternative based on a linear no threshold dose response model per year (i.e., the delta risk per year between the regulatory baseline and the alternative). The dose conversion factor (dollar per person-rem) is used to monetize the averted dose to allow comparison to other attributes.[15] The product of the dose averted and the dose conversion factor provides the monetized benefit per year.

The last row of the tables in this analysis provides the total benefit or cost offset for the attribute in 2012 dollars and is provided based on the NPV. The benefits and cost offsets are calculated by using the benefit/cost offset per year and applying it to the average remaining life of the affected entities. The way to apply the information to the average life is by discounting each year in the future by the discount rate. The formula for calculating NPV is

$$NPV = {}^{FV}\!\big/_{(1+r)^t}$$

where FV is future value, r is the discount rate, and t is the number of years from the base year to the year the benefit/cost offset is incurred. For example, \$100 in year 2013 (FV) would be worth \$97 in 2012 dollars (NPV) at a 3 percent discount rate. To determine the total benefit/cost/cost offset for an attribute, each year of the attribute is summed into a total that is provided within the table.

4.2.3 **Assumptions**

This section provides an overview of the assumptions used by the staff in this analysis to estimate the costs and benefits associated with expedited transfer. This section describes:

- Assumptions associated with economic modeling, the definition of representative plants, projection of future spent fuel discharges, and requirements for dry storage. This includes assumptions regarding fuel burnup, decay heat, and cesium-137 source term, as well as wet and dry storage technology capacity and heat load capability.

- Assumptions associated with SFP accident modeling and evaluation. This includes assumptions regarding the probability of initiating events challenging SFP integrity and spent fuel cooling, radiological release source term, atmospheric modeling and meteorology, post-accident radiological doses, population demographics and surrounding area economic data, long-term habitability criteria, and emergency response modeling.

- Assumptions associated with time periods required to load dry storage cask systems (DSCs) and occupational dose received during cask loading operations.

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15 Additional information on dollar per person-rem is found in Appendix C, Section C.2.5 and Appendix D, Section D.2.

- Assumptions regarding the costs of construction and operation of an at-reactor ISFSI, cost increases associated with expedited transfer, cost increases associated with the need for a short-term increase in DSC fabrication capacity, costs to load additional DSCs, and the need to increase shielding capability of DSCs to store spent fuel with shorter cooling times.

Assumptions used are documented throughout this report. For reader convenience, major assumptions are listed in Table 2.

Table 2 Major Assumptions

Topical Area	Major Assumption	Comment
Overall Approach	The fleet of U.S. reactor SFPs were classified in the following groups: 1. BWRs with elevated pools 2. PWRs and BWRs with dedicated pools near grade 3. New AP1000 reactors 4. PWRs that share a single pool 5. PWRs with pools that cannot rapidly drain 6. Decommissioning reactors For the first four groups, representative characteristics of the spent fuel and SFP loading conditions that were conservative with respect to the majority of SFPs within each group were selected. The remaining two groups were not evaluated due to the much lower potential for runaway zirconium oxidation.	The configuration of the plant is considered in determining potential bounding conditions regarding the potential drainage paths from the pools and the potential for natural circulation air cooling. The inventory of fuel, reactor thermal power, and fuel burn-up at reactors within each group are considered in determining the representative inventory of radioactive material present in the pool. Plant characteristics and accident progression for BWRs with elevated pools were drawn from the SFPS. Remaining plant characteristics and accident progression assumptions are drawn from NUREG-1353 and NUREG-1738.
Regulatory Baseline Condition	High-density loading configuration with one full core reserve capacity during which mitigation capability is assumed to be ineffective.	This loading configuration approximates the maximum fuel inventory normally maintained in the SFP. The assumption of ineffective mitigation maximizes the potential release frequency.
Alternative Condition	Low-density loading configuration with fuel decayed more than five years removed from the SFP and mitigation 95% effective.	This loading configuration approximates the minimum fuel inventory for an operating reactor SFP. The assumption of 95% effective mitigation minimizes the frequency of potential releases.
Seismic Hazard Characterization	Seismic hazard models – this analysis used the USGS 2008 model instead of the model currently under development in an ongoing regulatory program. While the USGS (2008) hazard model is not sufficiently detailed for regulatory decisions, it is appropriate to use for this analysis	A new probabilistic seismic hazard model is currently being developed and will consist of two parts: (1) a seismic source zone characterization and (2) a ground motion prediction equation (GMPE) model. Although part (1) is now complete (Ref. 16), it was not

Topical Area	Major Assumption	Comment
	because it was the most recent and readily available hazard model for the central and eastern U.S. plant sites. Hazards for the western sites will be evaluated when the updated model is complete.	available at the start of this analysis. In addition, the GMPE update is still in progress. Furthermore, the NRC is currently developing an independent probabilistic seismic hazard assessment (PSHA) computer code to incorporate part (1) and part (2) when complete.
Earthquake Frequency	Earthquake frequencies are based on hazard curves developed from 2008 USGS data for two bins having peak ground accelerations of 0.7g and 1.2g, respectively. Large earthquakes with frequencies on the order of a few occurrences every 100,000 years to once every 1,000,000 years have the potential to damage the SFP structure.	The USGS data provides a consistent method of quantifying earthquake frequency east of the Rockies. The low and base cases use the seismic hazard estimate for the SFPS reference plant, which results in higher earthquake frequency estimates than the USGS model for most plants. The high case uses the USGS model results for the site within each group with the highest earthquake frequency.
Cask Drop Frequency	A cask drop frequency of $2x10^{-7}$ per year is used for each SFP.	This value is drawn from an evaluation in NUREG-1738 and represents the potential for cask drops during routine transfer activities to maintain assumed SFP storage inventory. Additional cask movements associated with achieving low-density SFP storage are conservatively not evaluated.
AC Power Fragility	AC power is conservatively assumed to fail during earthquake and cask drop initiators to reflect loss of installed forced cooling and coolant makeup systems.	This assumption results in loss of forced cooling and other minor coolant leaks progressing to uncover the stored fuel unless mitigation is effectively deployed.
Liner Fragility	The values conservatively selected for the base case are: • 0.7g PGA earthquake - 10% for BWRs with elevated pools (SFPS) and 5% for all other groups • 1.2g PGA earthquake - 100% for BWRs with elevated pools and 50% for all other groups • Cask drop event - 100%	Liner Fragility represents the conditional probability of leakage from the SFP at locations that uncover the stored fuel, given an earthquake or cask drop occurs. The high case uses 100% for all initiators.
Other Initiating Event Frequencies	Loss of forced cooling and loss of coolant inventory events are conservatively represented by a total initiating event frequency of $2.37x10^{-7}$ per year.	Individual initiating events affecting loss of forced cooling, loss of AC power, loss of coolant inventory, and seal failures were drawn from NUREG-1738 and NUREG-1353.
Unavailability of	The conservative values selected for	Unavailability of natural circulation

Topical Area	Major Assumption	Comment
Natural Circulation Air Cooling – Partial Drain Conditions	the base case are: • 8% – 0.7g earthquake for BWRs with elevated pools (SFPS) • 100% – 0.7g earthquake for all other groups • 100% for the 1.2g earthquake • 100% for the cask drop event • 100% for all other initiators	air cooling reflects various conditions that could lead to inadequate heat removal and progression to runaway zirconium cladding oxidation. Conditions bounded by this result include: • fuel with high decay heat • recently discharged fuel in a contiguous pattern rather than distributed pattern • partial drain conditions with racks that block air cooling The high case uses 100% for all initiators.
Mitigation	Effective deployment of mitigation is conservatively assumed to reduce the frequency of release for low-density storage cases by a factor of 19.	Conservative assumption to maximize difference in release frequency between low-density and high-density storage configurations.
Release Frequency Determination	The release frequencies are calculated as the product of the frequency fuel becomes uncovered and the unavailability of air cooling. The frequency fuel becomes uncovered is the product of the initiating event frequency, ac power fragility, and liner fragility for the seismic and cask drop initiators. For all other initiators, the initiating event frequency is the frequency fuel becomes uncovered. For low-density storage configurations, the release frequency is reduced by a factor of 19 to reflect mitigation.	The earthquake and cask drop initiators dominate the events potentially leading to inadequate cooling of the fuel because these events are most likely to cause a leak from the pool at or below the elevation of the stored fuel. Other initiators are conservatively assumed to progress such that the coolant inventory does not adequately cool the stored fuel because of uncertainties in the accident progression.
Cs-137 Release fraction	The SFP Group 1 high-density loading release fractions are: • 3% for the low estimate • 40% for the base case • 90% for the high estimate	The SFPS (Table 27) shows that for the high-density scenarios involving a leak without mitigation measures, the maximum release is approximately 40%, which was used for the base case. A 90% release fraction is used for the high estimate to account for SFP variations within the group and uncertainties in the accident progression.
	The SFP Groups 2, 3 and 4 high-density loading release fractions used are: • 10% for the low estimate • 75% for the base case • 90% for the high estimate	These release fractions are consistent with the range of release fractions used in previous SFP studies.

Topical Area	Major Assumption	Comment
	The SFP Group 1, 2, 3, and 4 low-density loading release fractions are: • 0.5% for the low estimate • 3% for the base case • 5% for the high estimate	The SFPS (Table 28) shows that for the low-density scenarios involving a leak without mitigation measures, the maximum release is approximately 3%, which was used for the base case. A 5% release fraction is used for the high estimate to account for SFP variations within the group and uncertainties in the accident progression. The release fractions are the same for all groups because only the most recently discharged fuel is expected to be involved.
Radionuclide Source Term	A source term calculated by the MELCOR code based on the cesium release fraction.	The MELCOR code models the fuel damage state, radionuclide release, and holdup of aerosols.
Atmospheric Modeling and Meteorology	The atmospheric transport and dispersion model used in this analysis is based on the MACCS2 model developed using weather data for the Peach Bottom site, which is described in Section 7.1.2 of the SFPS.	A straight-line Gaussian plume segment dispersion model is used for the atmospheric transport.
Population and Economic Data	Representative site demographics are selected to represent the 90th percentile, the mean, the median, and the 20th percentiles. For each representative site, the site population and economic data is established for use in the consequence analysis.	Representative sites for the 90th percentile, the mean, the median, and the 20th percentile are Peach Bottom, Surry, Palisades, and Point Beach, respectively. To identify the specific effect of these values, the staff performed sensitivity studies where only one parameter was varied from a low to high value. Section 4 discusses this sensitivity study in more detail.
Emergency Response Model	The site-specific emergency response model from the SFPS is used to model evacuation timing and speed within the emergency planning zone.	The conditional individual risk measures near the site are expected to be relatively insensitive to site-specific characteristics (i.e., emergency response measures). This is because the predicted releases allow time for effective protective actions to limit exposures to the public.
Long-Term Habitability Criteria	The long-term phase is modeled for 50 years to calculate the consequences of exposure to the average person assuming habitation is limited to areas where annual dose	The selected habitability criteria affect the values of offsite property damage used in this analysis. Certain metrics such as offsite property damage, the number of

Topical Area	Major Assumption	Comment
	is within the criteria. The base case uses habitability criteria of 2 rem in the first year and 500 mrem each year thereafter. The high case uses a criterion of 2 rem annually.	displaced individuals (either temporarily or permanently) and the extents to which such actions may be needed are inversely proportional to changes in collective dose resulting from changes in habitability criteria.
Accident Occupational Exposure	Occupational exposures related to accident mitigation and recovery are estimated based on actual worker doses collected for the Fukushima Dai-ichi site.	The assumed accident period extends for one year and involves a work force of 3,700 people.
Health Consequences	The Linear No Threshold (LNT) dose-response model is used as the base for reporting results. The dose truncation methodology, introduced in the SOARCA analyses documented in NUREG-1935, is provided as a sensitivity analysis.	For large populations exposed to low annual doses, which is the case for some of the SFP accident scenarios, the health effects to populations in habitable zones dominate the health effects when the LNT model is used.
Implementation Cost Approach and Timing of Cask Loading	For the regulatory baseline, the plant is expected to load the required number of dry storage casks each refueling cycle to retain sufficient space in the SFP to discharge one full core of fuel. For the low-density storage alternative in Groups 1, 2, and 4, the plant is assumed to transfer all fuel that has greater than 5 years decay within a 5 year period and then continue loading dry storage casks each refueling cycle as necessary to maintain a full core reserve. For the low-density storage alternative in Group 3, the plant is expected to begin loading dry storage casks once the pool reaches the allowed capacity in a low-density (1x4) configuration.	Group dry storage cask loading is based on a representative plant selected within each group. The total number of dry storage casks necessary for the low-density storage alternative is higher than for the regulatory baseline because fuel assemblies that have decayed for shorter periods have higher decay heat levels, and the higher decay heat per assembly reduces the allowed capacity below its nominal capacity.
Occupational Dose	For the low-density storage alternative, each cask loaded in addition to the number required by the regulatory baseline is estimated to result in an incremental 400 person-mrem dose.	This radiation dose is consistent with the exposure value used in EPRI TR-1021049 (Ref. 17) and in EPRI TR-1018058 (Ref. 18), which analyzed worker impacts associated with loading spent fuel for transport to the proposed Yucca Mountain repository.
Incremental Upfront Cost of	Each additional dry storage cask is expected to require engineering,	Each of these cost components are further described in

Topical Area	Major Assumption	Comment
ISFSI Capacity	design and construction costs of $657,700 in 2012 dollars.	EPRI TR-1021048, “Industry Spent Fuel Storage Handbook.”
Incremental Cost of Additional Cask purchase and Loading	The base cost for purchase and loading of a dry storage cask is assumed to be $1,300,000. When only 5-year decayed, high-burnup fuel is available for loading, additional shielding; engineering, licensing, and operational expenses are assumed to increase the cost to $1,466,400 per cask.	These cost estimates are based on the DSC unit costs that EPRI used for a generic interim storage facility and documented in EPRI TR-1025206.
Incremental Annual ISFSI Operating Costs	The majority of reactor sites in Groups 1, 2, and 4, have operational ISFSIs, and the incremental operating cost for increased capacity is considered negligible for these groups. For Group 3, maintenance of low-density storage is expected to require early operation at an incremental cost of $1.1 million per year.	EPRI reports a wide variability in published estimates of annual ISFSI operating costs that range from $212,000 to $2 million per year in 2012 dollars and reported their estimate of $1.1 million per year for an ISFSI at an operating nuclear power plant site.

4.2.4 **Sensitivity Analysis**

Table 3 provides a list of sensitivity studies performed to estimate the effect upon the results of variations in input parameters. The output from the sensitivity studies is used to determine the importance of the evaluated parameters. The table below provides the parameter evaluated in the left column, what the parameter value is for the base case for the staff’s recommendation and sensitivities that the staff performed as additional information for the Commission, and whether it was determined to be a key parameter[16]. Additional detail describing these sensitivity studies is contained in Section 4.3 of this analysis and in Appendix D.

Table 3 Sensitivity Study Parameters

Parameters	Methodology		Key Parameter
	Base Case	Sensitivity	
Present value calculations	7% net present value	2% and 3% net present value	Yes
Dollar per person-rem conversion factor	$2,000	$4,000	Yes
Replacement energy costs (annual) (Constant 2012 dollars)	$2.3 million	Range: $729,000 to $57.3 million Average: $10.1 million Median: $6.7 million	No
Calculated consequences from site	50 miles	Beyond 50 miles	Yes

[16] A key parameter is a variable that can significantly affect calculation results.

Parameters	Methodology		Key Parameter
	Base Case	Sensitivity	
Uniform fuel pattern during outage	1x4 arrangement	Uniformly arranged for a short period	No
Population density	Surry	Range: Point Beach to Peach Bottom Median: Palisades	No
Habitability criteria	2 rem in the first year and 500 mrem each year thereafter	500 mrem per year and 2 rem per year	Yes
Seismic initiator frequency[1]	Bin 3: 1.65×10^{-5} Bin 4: 4.90×10^{-6}	Bin 3: 2.24×10^{-5} – 5.64×10^{-5} Bin 4: 7.09×10^{-6} – 2.00×10^{-5}	Yes

1 As discussed in section 3.2 of the SFPS, damage to the SFP and other relevant structures, systems, and components is not credible for events in Bins 1 and 2. These bins are further discussed in Appendix C, Section C 2.2.

4.3 Evaluation of Alternative—Low-Density Spent Fuel Pool Storage

This section discusses the costs and benefits of the evaluated alternative (i.e., expedited transfer) relative to the baseline or current practices. As described in the previous section, costs and benefits are provided for the various attributes addressed within a regulatory analysis and for a range of assumptions for various parameters (i.e., low estimate, base case, and high estimate). Information is also provided regarding the sensitivity of the cost/benefit assessments to several key factors. A qualitative discussion is provided for those issues not easily represented in monetary values.

4.3.1 **Public Health (Accident)**

This attribute measures expected changes in radiation exposure to the public caused by change in accident frequencies or accident consequences associated with the proposed action. The expected changes in radiation exposure are predicted over a 50-mile radius from the plant site. The calculated radiation dose to the public is primarily from reoccupation of the land and other activities following the SFP accident. In addition, the calculated radiation dose to the public includes the occupational dose to workers for cleanup and decontamination of contaminated land not onsite. The incremental radiation doses are calculated by subtracting the values for the alternative from those of the regulatory baseline. The difference (delta) is the averted dose benefit of this alternative in units of person-rem. The quantitative results for public health that could affect SFP risk are provided for each SFP grouping. These values are based on MACCS2 analyses and probabilistic considerations described in further detail in Appendix C of this analysis. The assumptions with regard to the base case seismic event frequencies are discussed in Appendix section C.2.2 and with regard to release frequencies are found in Appendix section C.2.3 of this cost-benefit analysis.

As Table 4 shows, the base case of the delta benefit for averted public health (accident) radiation exposure from a SFP accident resulting in spent fuel damage is approximately 1,740 person-rem for the Group 1 SFP and varies for each grouping. This dose represents the reduction of public health risk that results from a policy decision to transfer spent fuel from the SFP to dry storage in order to achieve low-density spent fuel loading in the pool. For a single

BWR Mark I or Mark II reactor with a non-shared SFP (Group 1), the averted delta dose exposure is approximately 69.6 person-rem per year over a remaining licensed commercial operation of the reactor of 24-years (until year 2037). The value assumes a U.S. reactor site average population density of approximately 300 people per square mile within a 50-mile radius from the site. The calculated dose is the difference between an uncontrolled release of radionuclides from a full high-density SFP with no credit for successful mitigation to a full low-density SFP with credit for successful mitigation. The averted doses reflects the calculated health benefits that result if adherence to the EPA intermediate phase protective action guides that allow a dose of 2 rem in the first year and 500 mrem each year thereafter are used.

To provide the Commission with additional information to inform its regulatory decisionmaking, an evaluation of the sensitivity of the results to a change in the dollar per person-rem conversion value from $2,000 to $4,000 per person-rem averted was performed and the results are also provided in Table 4.

Table 4 Summary of Public Health (Accident) for Expedited Transfer Alternative– Low-density Spent Fuel Pool Storage (Base case with $2,000 and $4,000 per person-rem)

SFP Group	Case	Dose conversion factor ($/person-rem)	Dose (averted person-rem per pool	Benefits (2012 million dollars)		
				2% NPV	3% NPV	7% NPV
1	Alternative 2 - Low-density storage	$2,000	1,740	$2.72	$2.42	$1.62
		$4,000		$5.43	$4.85	$3.24
2	Alternative 2 - Low-density storage	$2,000	1,630	$2.45	$2.15	$1.38
		$4,000		$4.90	$4.30	$2.75
3	Alternative 2 - Low-density storage	$2,000	3,020	$3.14	$2.37	$0.99
		$4,000		$6.28	$4.75	$1.98
4	Alternative 2 - Low-density storage	$2,000	1,690	$2.62	$2.33	$1.54
		$4,000		$5.25	$4.66	$3.08

4.3.1.1 Population Demographic Sensitivity

Population densities and distributions characteristics for SFP sites are examined to provide perspective on how important changes to these site demographic characteristics are for this cost-benefit analysis. The base case and the three additional site population densities and distributions near SFP locations and the results are discussed in Appendix C Section C.2.12.

4.3.1.2 Habitability Criteria Sensitivity

A long-term cleanup policy for recovery after a severe nuclear power plant accident does not currently exist. The actual decisions regarding how land would be recovered and populations relocated after an accident would be made by a number of local, State, and Federal jurisdictions and would most likely be based on a long-term cleanup strategy, which is currently being developed by the NRC, the U.S. Environmental Protection Agency (EPA), and other Federal agencies. Furthermore, a cleanup standard may not have an explicit dose level for cleanup. Instead, the cleanup strategy may give local jurisdictions the ability to develop localized cleanup goals after an accident, to allow for a number of factors that include sociopolitical, technical, and economic considerations.

For habitability, most States adhere to EPA intermediate phase protective action guides that allow a dose of 2 rem in the first year and 500 mrem each year thereafter. This habitability criterion was used in previous SFP studies, which used 4 rem in 5 years to represent these protective action guideline levels (e.g., 2 rem in year one, followed by 0.5 rem each successive

year). Further discussion of this approach is provided in Appendix section C.2.13 of this analysis.

4.3.1.3 Seismic Initiator Frequency Assumptions Sensitivity

Although the SFPS reference plant hazard exceedance frequencies curves discussed in Appendix section C.2.2 of this cost-benefit analysis falls close to the upper end of each group in terms of hazard estimates, there are some central and eastern United States (CEUS) sites that exceed those estimates. To analyze the seismic risk hazard for these CEUS sites in each SFP group, a high estimate using the largest site hazard exceedance frequency curve in the group is used to in this sensitivity study. The seismic frequencies are provided in Table 37 in Appendix section C.2.2. Other bounding seismic assumptions include the loss of all ac power for all SFP initiators, a conservative liner fragility value is discussed in Appendix section C.2.3 even though a detailed analysis may be able to justify a value of factor of 2 or more lower, and assuming a bounding value of 1.0 for the conditional probability of failure to successfully mitigate the high-density storage spent fuel accident. These conservative (bounding) assumptions were used in order to calculate a high value estimate for the seismic initiating frequency sensitivity analysis in order to analyze the effect on the public health (accident) attribute. Further discussion of this approach is provided in Appendix section C.2.4 of this analysis.

4.3.1.4 Sensitivity to a Uniform Fuel Pattern during an Outage

The base case of this cost-benefit analysis assumes that each licensee has prearranged the SFP such that discharged assemblies can be placed directly into a 1x4 arrangement for the discharges of the last two outages. However, those requirements do allow for the fuel to be stored in a less favorable configuration for some time following discharge if other considerations prevent prearrangement. To capture the effects of nonbeneficial arrangement of discharged fuel, this cost-benefit analysis evaluates the situation in which the discharged spent fuel is uniformly arranged during the outage to evaluate the effect of this aspect on public health (accident) attribute. For the offsite consequence analysis, the sequences with recently discharged fuel in a uniform configuration were binned in a similar manner to the low-density and high-density (1x4) loading scenarios. Because licensees are required to move their recently discharged fuel to a more favorable configuration after a certain amount of time, this sensitivity assumes that the high-density uniform case becomes identical to the high-density (1x4) case by the end of operating cycle phase 2 (OCP 2) or within 25 days.[17] Further discussion of this approach is provided in Appendix section C.2.15 of this analysis.

4.3.2 **Occupational Health (Accident)**

Occupational health measures both short-term and long-term health effects associated with site workers as a result of changes in accident frequency or accident mitigation. Within the regulatory baseline, the short-term occupational exposure related to the accident occurs at the time of the accident and during the immediate management of the emergency and during decontamination and decommissioning of the onsite property. The radiological occupational exposure resulting from cleanup and refurbishment or decommissioning activities of the damaged facility to occupational workers are estimated within the long-term occupational

[17] To analyze this scenario the plant operating cycle is divided into numerous small periods of time or operating cycle phases (OCPs). The definitions for the modeled operating cycle phases is provided in Table 16 of the SFPS.

exposure. The quantitative results for occupational health (accident) considering the contribution of all initiators that could affect SFP risk is provided in Table 5 and is based on the release frequencies discussed in Appendix section C.2.1and the occupational health (accident) assumptions found in Appendix section C.2.9. The high estimate also incorporates the seismic initiator frequency assumptions described in Section 4.3.1.3.

Table 5 Summary of Occupational Health (Accident) Benefits for Low-density Spent Fuel Pool Storage (Base case with \$2,000 and \$4,000 per person-rem and with Low and High Estimates)

SFP Group	Occupational Health (Accident) Case	Dose conversion factor (\$/person-rem)	Dose averted per pool (person-rem)	Benefits (2012 dollars)		
				2% NPV	3% NPV	7% NPV
1	Low Estimate	\$2,000	0.60	\$942	\$840	\$562
		\$4,000		\$1,884	\$1,730	\$1,203
	Base Case	\$2,000	5.49	\$8,579	\$7,652	\$5,121
		\$4,000		\$17,159	\$15,763	\$10,959
	High Estimate	\$2,000	67	\$105,037	\$93,684	\$62,697
		\$4,000		\$210,075	\$192,988	\$134,171
2	Low Estimate	\$2,000	0.34	\$500	\$400	\$300
		\$4,000		\$1,000	\$900	\$600
	Base Case	\$2,000	4.36	\$6,600	\$5,800	\$3,700
		\$4,000		\$13,100	\$11,500	\$7,400
	High Estimate	\$2,000	25	\$37,300	\$32,700	\$21,000
		\$4,000		\$74,600	\$65,500	\$41,900
3	Low Estimate	\$2,000	0.71	\$700	\$600	\$200
		\$4,000		\$1,500	\$1,100	\$500
	Base Case	\$2,000	9.16	\$9,500	\$7,200	\$3,000
		\$4,000		\$19,100	\$14,400	\$6,000
	High Estimate	\$2,000	52	\$54,200	\$41,000	\$17,100
		\$4,000		\$108,400	\$82,000	\$34,200
4	Low Estimate	\$2,000	0.30	\$500	\$400	\$300
		\$4,000		\$900	\$800	\$600
	Base Case	\$2,000	3.91	\$6,000	\$5,400	\$3,600
		\$4,000		\$12,100	\$10,700	\$7,100
	High Estimate	\$2,000	22	\$34,300	\$30,500	\$20,200
		\$4,000		\$68,700	\$61,000	\$40,400

As Table 5 shows, the total delta benefit for short- and long-term occupational health (accident) range between 3.91 and 9.16 person-rem averted per SFP for the base case. The estimated total benefit of the occupational health (accident) attribute for low-density SFP storage relative to the regulatory baseline, using the \$2,000 per person-rem averted conversion factor, net present value ranges are insignificant for the base case and do not warrant further sensitivity analysis. The high estimate includes the conservative inputs and assumptions for the seismic initiator frequency sensitivity analysis discussed in Section 4.3.1.3 of this cost-benefit analysis.

4.3.3 **Occupational Health (Routine)**

Occupational health (routine) accounts for radiological exposures to workers during normal facility operations (i.e., non-accident situations). These occupational exposures occur during DSC loading and handling activities, ISFSI operations, and maintenance and surveillance activities. The assumptions in relation to the exposures for occupational health (routine) are found in Section 4.3.3 of this cost-benefit analysis.

Table 6 Summary of Occupational Health (Routine) Costs for Low-Density Spent Fuel Pool Storage (Base Case with $2,000 and $4,000 per Person-rem)

SFP Group	No. of DSCs required through end of operation: High-density storage (Alternative 1)	Low-density storage (Alternative 2)	Delta Dose (p-rem)	Dose conversion factor ($/p-	Costs (2012 dollars): 2% NPV	3% NPV	7% NPV
1	107	119	6.84	$2,000	$25,400	$27,800	$28,200
				$4,000	$50,800	$55,600	$56,300
2	75	90	8.55	$2,000	$27,200	$29,100	$28,900
				$4,000	$54,500	$58,300	$57,700
3	77	87	5.70	$2,000	$14,500	$12,900	$6,400
				$4,000	$29,000	$25,800	$12,800
4	130	141	6.27	$2,000	$22,700	$24,700	$24,800
				$4,000	$45,400	$49,400	$49,700

As Table 6 shows, the delta benefit for occupational health (routine) is an increase of between 5.70 and 8.55 person-rem in worker exposure resulting from DSC loading and handling activities; ISFSI operations; and maintenance and surveillance activities depending on the SFP grouping. The estimated cost to the occupational health (routine) for low-density spent fuel storage relative to the regulatory baseline for all SFP groups and calculated in accordance with the current regulatory framework, ranges from $14,500 to $27,200 (2 percent net present value), $12,900 to $29,100 (3 percent net present value), and $6,400 to $28,900 (7 percent net present value) using the $2,000 per person-rem averted conversion factor. These ranges are insignificant for this analysis and do not warrant further sensitivity analysis.

4.3.4 **Offsite Property**

The offsite property attribute measures the expected total monetary effects on offsite property resulting from the proposed action. Changes to offsite property can take various forms, both direct, (e.g., land, food, and water) and indirect (e.g., tourism). This attribute is the product of the change in accident frequency and the property consequences from the occurrence of a SFP accident.

For the regulatory baseline, the offsite property costs are any property consequences resulting from any radiological release from the occurrence of an accident. Plant releases not related to the severe accident analyzed are outside the scope of this cost-benefit analysis.

The cost offsets for the analyzed SFP accident are quantified relative to the regulatory baseline based on the MACCS2 calculation results and probabilistic considerations. The results for the consequences from a low-density spent pool accident are compared to those from the regulatory baseline SFP accident. The calculation is the difference between the calculated consequences resulting from a low-density and a high-density SFP accident. The results are provided in Table 7. The assumptions with regard to the base case seismic event frequencies are discussed in Appendix section C.2.2 and with regard to release frequencies are found in Appendix section C.2.3 of this cost-benefit analysis.

Table 7 Summary of Offsite Property Cost Offsets for Expedited Transfer Alternative– Low-Density Spent Fuel Pool Storage within 50 Miles (Base Case)

SFP Group	Case	Offsite Property Cost Offsets (2012 million dollars)		
		2% NPV	3% NPV	7% NPV
1	Alternative 2 - Low-density storage	$8.96	$7.99	$5.35
2	Alternative 2 - Low-density storage	$9.03	$7.93	$5.08
3	Alternative 2 - Low-density storage	$11.45	$8.66	$3.61
4	Alternative 2 - Low-density storage	$9.81	$8.71	$5.76

As Table 7 shows, the estimate of offsite property damage from a SFP accident resulting in spent fuel damage, ranges from $8.96 million (2 percent net present value) to $5.35 million (7 percent net present value) for Group 1 SFPs and varies for each grouping. This value assumes a U.S. reactor site average population density of approximately 300 people per square mile within a 50-mile radius from the site and is representative of the associated property values found near the Surry power plant site. This base case uses the EPA intermediate phase PAG level of 2 rem in the first year and 500 mrem annually to evaluate post-accident collective dose and offsite property costs as discussed in Appendix section C.2.13 of this cost-benefit analysis.

4.3.4.1 Population Demographic Sensitivity

Certain metrics such as property use, the number of displaced individuals (either temporarily or permanently), and the extent to which such actions may be needed are affected by the population size and the amount of economic activity in the vicinity of the postulated accident.

This examination provides a perspective on how important changes to these site demographic variables are for this cost-benefit analysis. The base case and the three additional site population densities, distributions, and economic characteristics near SFP locations are discussed in Appendix section C.2.12. It provides a basis for understanding the nature and the extent of the relationship between population densities, distributions characteristics, and property values near SFP sites.

4.3.4.2 Offsite Property Consequences beyond 50 Miles Sensitivity

Because a SFP accident under certain scenarios and environmental conditions could result in impacts to offsite property located beyond 50 miles from the postulated accident site, this case evaluates the sensitivity of offsite property cost offsets for damages occurring beyond 50 miles from the site, using the base case assumptions and the intermediate EPA PAG criterion. This is discussed in Appendix section C.2.12.

4.3.4.3 Offsite Property Costs Sensitivity to Habitability Criteria

As discussed in Section 4.3.1.2, a long-term cleanup policy for recovery after a severe nuclear power plant accident does not currently exist. The actual decisions regarding how land would

be recovered and populations relocated after an accident would be made by a number of local, State, and Federal jurisdictions and would most likely be based on a long-term cleanup strategy, which is currently being developed by the NRC, EPA, and other Federal agencies. Furthermore, a cleanup standard may not have an explicit dose level for cleanup. Instead, the cleanup strategy may give local jurisdictions the ability to develop localized cleanup goals after an accident, to allow for a number of factors that include sociopolitical, technical, and economic considerations. Given the uncertainties in which long-term habitability criterion would be used, Appendix section C.2.13 discusses this sensitivity analysis and analyze the effect on the costs for offsite property damage.

4.3.4.4 Offsite Property Cost Offset Sensitivity to Seismic Initiator Frequency Assumptions

Although the SFPS reference plant hazard exceedance frequencies curves discussed in Appendix section C.2.1 of this analysis fall close to the upper end of each SFP group in terms of hazard estimates, there are some CEUS sites that exceed those estimates. To analyze the seismic risk hazard for these CEUS sites, a high estimate using the bounding plant hazard exceedance frequency curve is used to produce the high estimate seismic bins and initiating event frequencies. This sensitivity analysis is discussed in Appendix section C.2.4 of this analysis.

4.3.4.5 Offsite Property Cost Offset Sensitivity to a Uniform Fuel Pattern during an Outage

As discussed in Section 4.3.1.4, the base case assumes that the licensee has prearranged the SFP such that discharged assemblies can be placed directly into a 1x4 arrangement for the discharges of the last two outages. This approach is consistent with Section 9.3 of the SFPS (Ref. 2). However, fuel is allowed to be stored in a less favorable configuration for some time following discharge if other considerations prevent prearrangement. To capture the effects of non-beneficial arrangement of discharged fuel, this cost-benefit analysis evaluates the situation in which the discharged spent fuel is uniformly arranged during the outage to evaluate the effect of this aspect on offsite property attribute.

For the offsite consequence analysis, the sequences with recently discharged fuel in a uniform configuration were binned in a similar manner to the low-density and high-density (1x4) loading scenarios. Because licensees are required to move their recently discharged fuel to a more favorable configuration after a certain amount of time, this sensitivity assumes that the high-density uniform case becomes identical to the high-density (1x4) case during operating cycle phase 3 (OCP3.). While the uniform case has different release categories, the situations that lead to release are largely the same as the low-density and high-density (1x4) base cases.

Table 65 in Appendix C provides a comparison of the effect on the offsite property cost offsets if a plant operator initially places discharged spent fuel in a uniform pattern and achieves the 1x4 pattern by the end of OCP2 (i.e., within 25 days) versus placing the fuel directly into the 1x4 pattern.

4.3.5 Onsite Property

This attribute measures the expected monetary effects on onsite property, including replacement power costs, decontamination, and refurbishment costs, from the proposed action.

There are two forms of onsite property costs that each alternative must disposition. The first type of onsite property costs are the cleanup and decontamination costs for the unit. The second type of onsite property costs is the cost to replace the energy from the damaged or shutdown unit(s). The cost offsets for low-density SFP storage are quantified relative to the regulatory baseline based on the probabilistic considerations provided in the SFPS (Ref. 2) and the onsite property estimates described in Appendix C.2.7.

Because many nuclear power plants have more than one reactor unit co-located on a plant site, it is assumed that a severe SFP accident that occurs at one unit would result in the cleanup and/or decommissioning costs and the loss of power generation for the affected unit. The postulated SFP accident might also result in the temporarily loss of power generation from the co-located unit. In modeling the replacement energy costs based on this scenario, it is assumed for the high estimate that replacement energy would be purchased for two units.

Based on these modeling assumptions, the onsite property results are provided in Table 8.

Table 8 Summary of Onsite Property Cost Offsets for Low-density Spent Fuel Pool Storage

Group	Case	Onsite Property Cost Offsets (2012 dollars)								
		Low Estimate			Base Case			High Estimate		
		2% NPV	3% NPV	7% NPV	2% NPV	3% NPV	7% NPV	2% NPV	3% NPV	7% NPV
1	Onsite Property - Replacement Energy	$90	$80	$50	$9,620	$8,450	$5,270	$34,680	$30,440	$19,000
	Onsite Property - Cleanup, Decontamination, Repair, & Refurbishment	$5,900	$5,200	$3,100	$57,900	$50,200	$30,200	$173,600	$150,500	$90,500
	Group 1 Total	$5,990	$5,280	$3,150	$67,520	$58,650	$35,470	$208,280	$180,940	$109,500
2	Onsite Property - Replacement Energy	$50	$40	$30	$7,500	$6,480	$3,850	$27,010	$23,340	$13,880
	Onsite Property - Cleanup, Decontamination, Repair, & Refurbishment	$3,200	$2,800	$1,600	$44,300	$37,800	$21,700	$132,800	$113,400	$65,200
	Group 2 Total	$3,250	$2,840	$1,630	$51,800	$44,280	$25,550	$159,810	$136,740	$79,080
3	Onsite Property - Replacement Energy	$80	$60	$20	$11,510	$8,530	$3,250	$41,490	$30,740	$11,700
	Onsite Property - Cleanup, Decontamination, Repair, & Refurbishment	$4,700	$3,500	$1,300	$64,400	$47,300	$17,700	$193,100	$142,000	$53,200
	Group 3 Total	$4,780	$3,560	$1,320	$75,910	$55,830	$20,950	$234,590	$172,740	$64,900
4	Onsite Property - Replacement Energy	$50	$40	$20	$6,820	$5,960	$3,670	$23,710	$20,810	$12,990
	Onsite Property - Cleanup, Decontamination, Repair, & Refurbishment	$3,000	$2,600	$1,500	$40,800	$35,200	$20,900	$122,300	$105,700	$62,800
	Group 4 Total	$3,050	$2,640	$1,520	$47,620	$41,160	$24,570	$146,010	$126,510	$75,790

As Table 8 shows, based on these calculations, the delta cost offset for the frequency-weighted onsite property base case estimate ranges from $47,620 to $75,910 per pool (2 percent net present value) to $41,160 to $55,830 per pool (3 percent net present value), and to $20,950 to

$35,470 per pool (7 percent net present value). Low and high estimates are also provided in Table 8.

4.3.6 **Industry Implementation**

Industry implementation accounts for the projected net economic effect on the affected licensees to implement the mandated changes. Costs evaluated for dry storage include upfront and incremental dry storage cask (DSC) capital and loading costs. Additional costs above the regulatory baseline are considered negative and cost savings are considered positive. The quantitative results for industry implementation are given in terms of expected costs if a policy decision is made to accelerate the transfer of spent fuel stored in SFPs to dry storage. These expected costs are not frequency weighted. Assumptions used for developing the industry implementation cost model are discussed in Appendix sections C.1.7, C.4.3, and C.4.4.

4.3.6.1 Industry Implementation Cost Summary

Table 9 provides a summary of the industry implementation costs for each SFP group and provides the number of additional DSCs that are needed to store the hotter spent fuel.

Table 9 Industry Implementation Costs for Low-Density Spent Fuel Pool Storage for a Single Spent Fuel Pool

SFP Group	No. of additional DSCs needed	Implementation Costs (2012 million dollars		
		2% NPV	3% NPV	7% NPV
1	12	$52.6	$55.2	$52.3
2	15	$51.4	$53.8	$51.3
3	10	$42.4	$35.8	$16.7
4	11	$48.8	$50.4	$46.4

Table 9 shows, the incremental costs associated with DSC upfront costs and the earlier purchasing and loading of DSCs on a periodic basis. The estimated industry implementation costs for low-density spent fuel storage relative to the regulatory baseline and calculated in accordance with the current regulatory framework, ranges from $42.4 to $52.6 million (2 percent net present value), $35.8 to $55.2 million (3 percent net present value), and $16.7 to $52.3 million (7 percent net present value).

4.3.6.2 Implementation Costs to Install Open Frame Low-Density Racks in an Existing Spent Fuel Pool

The re-racking of a SFP with open frame low-density racks is a preventive risk reduction alternative, which is intended to reduce radiological material available and promote air cooling to prevent the onset of self-sustaining clad oxidation in the event of loss of SFP water inventory. As stated in the alternative, older spent fuel assemblies are expeditiously moved from SFP storage to dry cask storage beginning in year 2014 to achieve low-density spent fuel storage and provide an opportunity to re-rack the SFP. Re-racking a SFP involves replacing the existing high-density storage rack modules with new open frame low-density racks and is estimated to take approximately 2.5 years based on a hypothetical SFP re-racking schedule to install high-density racks provided in EPRI TR-1021048 (Ref. 19). The EPRI estimated schedule is provided in Figure 2.

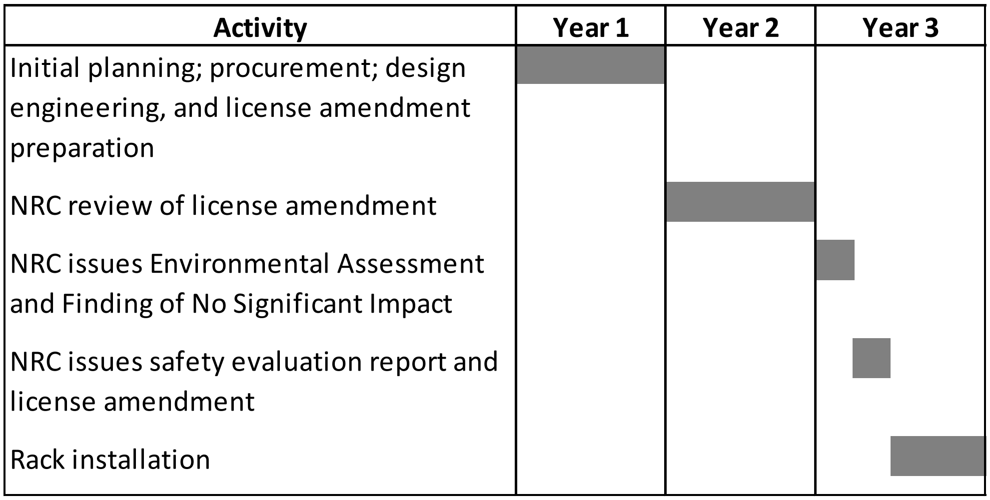

Figure 2 Estimated schedule for spent fuel pool re-racking project

The licensee would need to perform comprehensive safety analyses for the SFP re-rack project. These analyses will generally evaluate SFP criticality analysis; mechanical and structural design; seismic design; radiation protection provisions during rack removal and installation; changes to plant technical specifications; heavy loads analyses for the SFP during rack removal and installation; and SFP thermal-hydraulic; decay heat analyses; and radiological consequences of beyond-design-basis events. In addition to these design and engineering costs, other cost components include preparation of a license amendment and changes to the plant’s technical specifications; specification and procurement of low-density replacement racks; rack manufacture, rack installation, and handling and disposal of the old high-density storage racks. One licensee estimated (Ref. 20) the cost for a single unit SFP re-rack project to be \$7.5 million in 1979 which is equivalent to \$23.7 million[18] in 2012 dollars.

This cost element was not included in this alternative because it would add substantial cost and is inefficient in terms of regulatory benefit given that much of the benefit is achieved by storing less fuel in the existing high-density racks for less cost. Based on insights from the SFPS, the staff believes that within the first few months after the fuel came out of the reactor, the decay heat in the freshly unloaded spent fuel is high enough to cause a zirconium fire even in the presence of convective cooling. Therefore, reracking the SFP to install open frame racks even with channel boxes removed to allow potential crossflow, would not necessarily prevent a radiological release during this time.

4.3.7 **Industry Operation**

Industry operation accounts for the projected net economic effect caused by routine and recurring activities required by the proposed alternative. Annual operating costs for an ISFSI during reactor operation include the costs associated with NRC inspections; security; radiation monitoring; ISFSI operational monitoring; technical specification and regulatory compliance,

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18 This cost was converted from the licensee’s cost estimate of \$7.5 million in 1979 dollars using the consumer price index cost inflator. The licensee’s cost estimate includes the following: design, materials, fabrication; removal and disposal of old racks; transportation and installation of new racks; project management, licensing, quality assurance; contingency allowance; and allowances for funds used during construction.

including implementation of new certificate of compliance (CoC) amendments; personnel cost and code maintenance associated with fuel selection for dry storage; personnel costs for spent fuel management and fabrication surveillance activities; electric power usage for lighting and security systems; road maintenance to the ISFSI site; and miscellaneous expenses associated with ISFSI maintenance. NRC license fees for dry storage are included as part of the 10 CFR 50, "Domestic Licensing of Production and Utilization Facilities," operating license fees. As discussed in Appendix section C.4.4, incremental costs associated with annual ISFSI operating costs are insignificant for this analysis.

Industry operation also includes annual operating costs following reactor shutdown for decommissioning, which includes the costs associated with transporting spent fuel offsite. These costs were beyond the scope of the evaluation of expedited transfer of spent fuel to dry cask storage and are not included in this analysis.

The ability of a nuclear power plant operator to transfer spent fuel to dry storage during power operation is dependent upon what other activities are scheduled in the fuel handling area, plant-specific limitations on use of cask lifting crane or movement restrictions of heavy loads, or resource limitations if fuel handling equipment or personnel are shared between multiple reactor units. Furthermore, there could be operational impacts associated with large DSC loading campaigns as depicted in Figure 16 through Figure 19. These unintended consequences could include additional management support or attention to dry storage operations for longer periods, potential impacts on plant outage schedules or maintenance schedules because of increased staffing needs to support cask loading operations, and additional dry cask storage vendor oversight.

4.3.8 **NRC Implementation**

These costs, if calculated, would further reduce the calculated net benefit for this analysis.

4.3.9 **NRC Operation**

These costs, if calculated, would further reduce the calculated net benefit for this analysis.

4.3.10 **Other Considerations**

The other considerations are provided in relation to the regulatory baseline.

4.3.10.1 Seismic Hazard Model Uncertainties

There remain significant uncertainties in estimating the frequency of events for natural phenomena, which are postulated to challenge SFP cooling or integrity. This cost-benefit analysis uses the existing USGS 2008 model to evaluate seismic hazards at CEUS nuclear power plants. A new probabilistic seismic hazard model is currently being developed and will consist of two parts: (1) a seismic source zone characterization and (2) a ground motion prediction equation (GMPE) model. Although part (1) is now complete (Ref. 16), the GMPE update is still in progress. Furthermore, the NRC is currently developing an independent probabilistic seismic hazard assessment (PSHA) computer code to incorporate part (1) and part (2) when complete. While the USGS (2008) hazard model is not sufficiently detailed for regulatory decisions, it is used for this cost-benefit analysis because it is the most recent and readily available hazard model and was used in the SFPS.

4.3.10.2 Other Modeling Uncertainties

There are also significant uncertainties in the calculation of event consequences in terms of the dispersion and disposition of radioactive material into the site environs. This is due in part to significant uncertainties regarding the degree to which topographical features and other phenomena are modeled at distances away from the evaluated site. Estimating economic consequences also includes large uncertainties, as it is difficult to model the impact of disruptions to many different aspects of local economies and the loss of infrastructure on the general U.S. economy. An example of this is the supply chain disruptions that followed the 2011 Tohoku earthquake and subsequent tsunami on Japan or the 2004 Indian Ocean earthquake and tsunami on Thailand.

4.3.10.3 Cask Handling Risk

The NRC recognizes that there are costs and risks associated with the handling and movement of spent fuel casks. These cost and risk impacts, if included in this analysis, would further reduce the overall net benefit in relation to the regulatory baseline. These effects (e.g., the added risks of handling and moving casks) were conservatively ignored in order to calculate the maximum potential benefit by only comparing the safety of high-density fuel pool storage relative to low-density fuel pool storage and its implementation costs without consideration of cask movement risk.

4.3.10.4 Additional Repackaging Costs and Risk

Considering the uncertainty associated with the final disposal of spent fuel, there could be a potential impact of expedited transfer on the Department of Energy's (DOEs) cask standardization program and acceptance for final disposal. Should expedited transfer be required, it is expected that utilities would employ large capacity storage casks to minimize costs and handling. None of the proposed DOE repository designs were planned to accommodate the direct emplacement of large casks. Thus, the use of large canisters for storage may prove incompatible with a future repository design. There could be additional costs and risk associated with repackaging the spent fuel into canisters that are compatible with final disposal requirements. The staff is currently engaged in a significant effort with DOE and industry to address technical issues related to long term aging issues, such as canister and fuel cladding degradation. This ongoing DOE research effort could provide valuable insights with a direct impact on the potential costs and benefits of expedited spent fuel transfer to dry cask storage. These additional repackaging costs and risk were conservatively ignored to calculate the minimum implementation costs for the low-density fuel pool storage alternative.

4.3.10.5 Mitigating Strategies

The release of fission products to the environment from events that may cause the loss of SFP cooling or integrity, such as seismic events, missiles, heavy load drops, loss of cooling or make-up, inadvertent drainage or siphoning and pneumatic seal failures, are estimated to be range between 7.39×10^{-7} to 3.46×10^{-5} per year without successful mitigation. Operator diagnosis and recovery are important factors considered in the development of the event frequencies for these events and portions of this evaluation are premised on licensees having taken appropriate actions to understand the potential consequences of SFP accident events

and develop appropriate procedures and mitigating strategies to respond and mitigate the consequences.

The SFPS (Ref. 2) evaluated the potential benefits of mitigation measures required under 10 CFR 50.54(hh)(2) (Ref. 6), which were implemented following the September 11, 2001 attacks. These mitigation measures are intended to maintain SFP cooling in the event of a loss of large areas of the plant caused by explosions or fire. Neither the SFPS nor previous SFP studies considers the post-Fukushima improvements required by NRC and being implemented by the plants. These improvements are intended to increase the likelihood of restoring or maintaining power and mitigation capability during severe accidents.

The new SFP level instrumentation required under Order EA-12-051 and the mitigation strategies now required under Order EA-12-049 significantly enhance the likelihood of successful mitigation beyond that considered in this cost-benefit analysis because of the following features:

- Portable equipment with redundant sets (e.g., N+1) that is sufficient to supply all functions, simultaneously for the entire site, including equipment for the SFP. This portable equipment provides reasonable protection from seismic events, which are a dominant contributor to SFP risk.

- The mission time for this equipment is indefinite, versus the 12-hour mission time for the 50.54(hh)(2) equipment.[19]

- The new EA-12-049 mitigating strategies (Ref. 10) are capable of being deployed in all modes, which means that the new strategies can address SFP cooling issues that could occur in any operating cycle phase.

- The new SFP level instrumentation required under Order EA-12-051 (Ref. 11), ensures a reliable indication of the water level in the SFP for identification of the following pool water level conditions:
    - a level that is adequate to support operation of the normal fuel pool cooling system
    - a level that is adequate to provide substantial radiation shielding for a person standing on the SFP operating deck
    - a level where fuel remains covered and actions to implement makeup water addition should no longer be deferred

- The method of filling the SFP is via a connection to the normal SFP makeup system located away from the SFP floor, reducing the impacts on human performance because of potentially adverse environmental conditions (e.g., high temperature, humidity, and radiation) following an event.

[19] This section of the regulations deals with the development and implementation of guidance and strategies intended to maintain or restore core cooling, containment, and spent fuel pool cooling capabilities under the circumstances associated with loss of large areas of the plant resulting from explosions or fire.

This additional equipment, strategies, and features provided by Orders EA-12-049 and EA-12-051, provide additional accident mitigation capability and would further enhance the likelihood of successful mitigation, thereby further reducing the value for the conditional probability of release used in this cost-benefit analysis.

4.3.10.6 Cost Uncertainties

It is difficult to determine costs that could be incurred 50 to 100 years in the future. Changes in technology, regulation, or public policy could all have a profound effect on the actual cost. The purpose of including costs is to try to discern the benefit for the expedited transfer alternative. Of course, this analysis is based on best estimates of current spent fuel strategies and cost. If the U.S. government were to take possession of the spent fuel in order to provide storage at a non-operating plant site for extended periods, the costs could be heavily discounted, and the differences between storage alternatives in this analysis might be reduced.

4.3.10.7 Inadvertent Criticality

Design requirements and related safety analyses ensure fuel stored in the SFP will remain safely subcritical under conditions considered as part of the design basis, but rare conditions beyond the design basis may challenge some measures used to control reactivity. To maintain adequate margin to criticality in U.S. SFPs, the safety analyses credit the geometric configuration of the fuel and a combination of other measures that may include fixed neutron poison material (e.g., Boraflex) and limits on fuel reactivity. In addition, the presence of soluble boron in the coolant of PWR SFPs may be credited, but the stored fuel must remain subcritical assuming unborated water is present (10 CFR 50.68). Since these measures may be challenged by a beyond design-basis event, the NRC staff cannot rule out the potential for an inadvertent criticality event. However, the NRC staff judges that the potential consequences of a zirconium fire in the SFP and an associated hydrogen deflagration considered in this analysis would not be significantly affected by an inadvertent criticality event. The NRC staff bases this judgment on the following considerations:

- Fuel assembly geometric configuration would be maintained while water covers the fuel. Commercial reactor fuel assemblies are robust components designed to withstand the effects of design basis events, including safe shutdown earthquakes, while producing power in an operating reactor. The operating environment of a SFP is considerably less demanding than that of an operating reactor. The fuel racks are also designed to withstand design basis events, and the presence of water around the racks tends to dampen the effects of seismic events on these structures. While the earthquakes considered in this analysis are beyond what the fuel was designed to withstand, the NRC staff judges that fuel cladding and the fuel rack structure would not experience sufficient damage during a seismic event of these magnitudes to cause significant changes in the geometric configuration of the fuel.

- Potential criticality is limited by moderator availability and pool configuration. Many U.S. SFPs rely on the presence of neutron absorbing materials that are part of the storage rack structure to meet sub-criticality requirements under normal and credible abnormal events. The performance of these materials following a large beyond design basis seismic event has not been fully analyzed. It is possible that the environmental conditions after the beyond design basis seismic event could cause degradation of these materials. However, the presence of a moderator is necessary for an inadvertent

criticality event to occur, and an adequate moderator would only be present during the drain down/boil off phase or during recovery actions. While neither of these scenarios has been analyzed, the sustainable power of the inadvertent criticality event would be limited to a level significantly below the operating reactor, since the SFP is an open system and significant heat generation would create steam voids that provide inadequate moderation. Therefore, the additional fission product inventory in the fuel would not be significant. In addition, the required moderator for criticality limits the effect of any inadvertent criticality event because the water would provide shielding and reduce the fraction of radioactive material that would be released.

- Consequences of an inadvertent criticality event would be insignificant relative to consequences of a zirconium fire: Fuel assemblies that experienced zirconium cladding ignition could have sufficient cladding damage where further agitation, such as seismic aftershocks, would relocate fuel fragments in a non-uniform configuration. In this scenario, a large majority of the radioactive source term material would have already been released during the zirconium fire. The release from a subsequent inadvertent criticality event would be primarily a hazard to onsite workers with little offsite impact. The staff expects that any sustained inadvertent criticality event would be orders of magnitude lower than the power generated in the reactor with a corresponding lower production of short half-lived releasable material, making the inadvertent criticality event an insignificant contributor to the consequences of the zirconium fire. Therefore, the NRC staff judges that the consequences of a potential inadvertent criticality event following a zirconium fire fuel need not be considered. Furthermore, if a SFP criticality event did occur and generated short-lived radionuclides that are associated with offsite early fatalities, the emergency response as modeled effectively prevents any early fatality risk. This occurs in part because the modeled accident progression results in releases that are long compared with the time needed for relocation.

4.4 Presentation of Results

This section presents the analytical results, including discussion of supplemental considerations, uncertainties in estimates, and results of sensitivity analyses on the overall benefits.

4.4.1 **Cost-Benefit Analysis**

4.4.1.1 Summary Table

Table 10 provides the quantified and qualified costs and benefits for low-density SFP storage for each spent fuel group. For the quantitative analysis, the low estimate, base case, and high estimate results within 80 kilometers (50 miles) are reported.

The calculated benefits for requiring low-density SFP storage (Alternative) for the low estimate and base case are less than industry costs to achieve a low-density spent fuel loading pattern for each SFP group. As might be expected for estimates that include a compounding of the most conservative assumptions, all of the SFP group high estimate cases result in calculated benefits that are greater than the estimated costs.

Similar to the seismic event analyzed for the SFPS, no offsite early fatalities are calculated to occur. This results from the following two reasons:

(1) In comparison to reactors, SFPs have a larger proportion of longer-lived radionuclides, which are less likely to cause the significant doses required for acute health effects.

(2) Despite the large releases for certain predicted SFP accident progressions, the release from the most recently discharged fuel (which contains the shorter-lived radionuclides) is predicted to be insufficiently fast and insufficiently large to reach the acute thresholds associated with offsite early fatalities. When doses do exceed minimum levels for early fatalities, emergency response, as treated in the SFPS, effectively prevents any early fatality risk, at least in part because the modeled accident progression results in releases that are long compared with the time needed for relocation.

In addition, the predicted long-term exposure of the population, which could result in latent cancer fatality risk, is also low for the following reasons:

(1) The individual latent individual latent cancer fatality risk within 0 to 10 miles is predicted to be on the order of 2.4×10^{-10} to 1.5×10^{-8} per year, based on the linear no threshold (LNT) dose response model.

(2) The risk within 10 miles of the analyzed accident is dominated by low dose received at a low dose rate. Using truncation levels that do not quantify the effects of doses below 620 mrem/year (i.e., those arising from representative background radiation including average annual medical exposures) reduces the estimated individual LCF risk by up to a few orders of magnitude for the accident as modeled.

(3) Average individual latent cancer fatality risk is low but decreases slowly as a function of distance from the plant. Additionally, the predicted individual risks of latent cancer fatalities are dominated by long-term exposures to very lightly contaminated areas for which doses are small enough to be considered habitable.

Table 10 Summary of Totals for Alternatives

Net Monetary Savings (or Costs) – Total Present Value	Sensitivity Studies	Qualitative Benefits and (Costs)
Regulatory Baseline – Maintain the Existing Spent Fuel Storage Requirements		
$0	None	None.
Expedited Transfer Alternative – Low-density Spent Fuel Pool Storage		
Group 1 – BWR Mark I and Mark II with non-shared SFPs		
Group 1 Industry (Costs): *Base case* ($52 million) using a 7% discount rate **NRC (Costs):** Not calculated **Benefits:** *Base case* $7 million using a 7% discount rate **Group 1 Net Benefit = Benefits + (Costs)** **Base case: $7M + ($52M) = ($45M)** **Conclusion: Not cost beneficial**	**Group 1 Sensitivity Studies** **Industry (Costs) Sensitivity Studies** ($53 million) using a 2% discount rate ($55 million) using a 3% discount rate **Benefit Sensitivity Studies** *Low estimate* $0.2 million using a 2% discount rate $0.2 million using a 3% discount rate $0.1 million using a 7% discount rate *High estimate* $123 million using a 2% discount rate $109 million using a 3% discount rate $73 million using a 7% discount rate **Net Benefit Sensitivity Studies** *Low estimate* ($52.8M) using a 2% discount rate ($54.8M) using a 3% discount rate ($51.9M) using a 7% discount rate *High estimate* $70 million using a 2% discount rate $54 million using a 3% discount rate $21 million using a 7% discount rate	**Qualitative Benefits and (Costs)** **Qualitative (Costs):** Cost Uncertainties (Repackaging Costs) **Qualitative Benefits:** Modeling Uncertainties. (Cask Handling Risk) Mitigating Strategies
Group 2 – PWR and BWR Mark III with non-shared SFPs		
Group 2 Industry (Costs): *Base case* ($51 million) using a 7% discount rate **NRC (Costs):** Not calculated **Benefits:** *Base case* $6.4 million using a 7% discount rate **Group 2 Net Benefit = Benefits + (Costs)** **Base case: $6.4M + ($51M) = ($45M)** **Conclusion: Not cost beneficial**	**Group 2 Sensitivity Studies** **Industry (Costs) Sensitivity Studies** ($51 million) using a 2% discount rate ($54 million) using a 3% discount rate **Benefit Sensitivity Studies** *Low estimate* $0.3 million using a 2% discount rate $0.3 million using a 3% discount rate $0.2 million using a 7% discount rate *High estimate* $137 million using a 2% discount rate $121 million using a 3% discount rate $77 million using a 7% discount rate	**Qualitative Benefits and (Costs)** **Qualitative (Costs):** Cost Uncertainties (Repackaging Costs) **Qualitative Benefits:** Modeling Uncertainties. (Cask Handling Risk) Mitigating Strategies

Net Monetary Savings (or Costs) – Total Present Value	Sensitivity Studies	Qualitative Benefits and (Costs)
	Net Benefit Sensitivity Studies *Low estimate* ($50.7M) using a 2% discount rate ($53.7M) using a 3% discount rate ($50.8M) using a 7% discount rate *High estimate* $86 million using a 2% discount rate $67 million using a 3% discount rate $26 million using a 7% discount rate	
Group 3 – New reactor SFPs		
Group 3 Industry (Costs): *Base case* ($17 million) using a 7% discount rate **NRC (Costs):** Not calculated **Benefits:** *Base case* $4.6 million using a 7% discount rate **Group 3 Net Benefit = Benefits + (Costs)** **Base case: $4.6M + ($17M) = ($12M)** **Conclusion: Not cost beneficial**	**Group 3 Sensitivity Studies** **Industry (Costs) Sensitivity Studies** ($42 million) using a 2% discount rate ($36 million) using a 3% discount rate **Benefit Sensitivity Studies** *Low estimate* $0.3 million using a 2% discount rate $0.3 million using a 3% discount rate $0.1 million using a 7% discount rate *High estimate* $108 million using a 2% discount rate $81 million using a 3% discount rate $34 million using a 7% discount rate **Net Benefit Sensitivity Studies** *Low estimate* ($41.7M) using a 2% discount rate ($35.7M) using a 3% discount rate ($16.9M) using a 7% discount rate *High estimate* $66 million using a 2% discount rate $45 million using a 3% discount rate $17 million using a 7% discount rate	**Qualitative Benefits and (Costs)** **Qualitative (Costs):** Cost Uncertainties (Repackaging Costs) **Qualitative Benefits:** Modeling Uncertainties. (Cask Handling Risk) Mitigating Strategies
Group 4 – Reactor units with shard SFPs		
Group 4 Industry (Costs): *Base case* ($46 million) using a 7% discount rate **NRC (Costs):** Not calculated **Benefits:** *Base case* $7.3 million using a 7% discount rate **Group 4 Net Benefit = Benefits + (Costs)**	**Group 4 Sensitivity Studies** **Industry (Costs) Sensitivity Studies** ($49 million) using a 2% discount rate ($50 million) using a 3% discount rate **Benefit Sensitivity Studies** *Low estimate* $0.3 million using a 2% discount rate $0.3 million using a 3% discount rate $0.2 million using a 7% discount rate *High estimate* $205 million using a 2% discount rate	**Qualitative Benefits and (Costs)** **Qualitative (Costs):** Cost Uncertainties (Repackaging Costs) **Qualitative Benefits:** Modeling Uncertainties. (Cask Handling Risk) Mitigating Strategies

Net Monetary Savings (or Costs) – Total Present Value	Sensitivity Studies	Qualitative Benefits and (Costs)
Base case: $7.3M + ($46M) = ($39M) **Conclusion: Not cost beneficial**	$182 million using a 3% discount rate $120 million using a 7% discount rate **Net Benefit Sensitivity Studies** *Low estimate* ($48.7M) using a 2% discount rate ($49.7M) using a 3% discount rate ($48.8M) using a 7% discount rate *High estimate* $156 million using a 2% discount rate $132 million using a 3% discount rate $74 million using a 7% discount rate	

4.4.1.2 Implementation and Operation Costs–Low- Density Spent Fuel Pool Storage Alternative

4.4.1.2.1 Spent Fuel Pool Group 1 – BWR Mark I and Mark II reactors with non-shared spent fuel pool

Table 11 Summary of Total Implementation and Operation Costs for Low-Density Spent Fuel Pool Storage—Spent Fuel Pool Group 1

Attribute	Costs per SFP (2012 dollars in millions)		
	2% NPV	3% NPV	7% NPV
Occupational Health (Routine)	$0.03	$0.03	$0.03
Industry Implementation	$52.61	$55.17	$52.28
Industry Operation	nc	nc	nc
NRC Implementation	nc	nc	nc
NRC Operation	nc	nc	nc
Total per pool	$52.64	$55.20	$52.31
Total for 31 pools	$1,632	$1,711	$1,622

nc = not calculated

The low-density SFP storage alternative for BWR Mark I and Mark II reactors with a non-shared SFP total implementation and operation costs is the summation of those costs for the industry and the NRC. As shown in Table 11, the total estimated costs for a single Group 1 SFP to achieve and maintain a low-density SFP loading ranges from $52.64 million (2 percent net present value), to $55.20 million (3 percent net present value), and to $52.31 million (7 percent net present value). The total cost for all 31 SFPs in this group is approximately $1.6 billion. These costs are dominated by the capital costs for the DSCs and the loading costs for the storage systems to achieve low-density storage in the SFP than that required for the regulatory baseline.

4.4.1.2.2 Spent Fuel Pool Group 2 – PWR and BWR Mark III reactors with non-shared spent fuel pool

Table 12 Summary of Total Implementation and Operation Costs for Low-Density Spent Fuel Pool Storage—Spent Fuel Pool Group 2

Attribute	Costs per SFP (2012 dollars in millions)		
	2% NPV	3% NPV	7% NPV
Occupational Health (Routine)	$0.03	$0.03	$0.03
Industry Implementation	$51.37	$53.80	$51.33
Industry Operation	nc	nc	nc
NRC Implementation	nc	nc	nc
NRC Operation	nc	nc	nc
Total per pool	$51.40	$53.83	$51.36
Total for 49 pools	$2,519	$2,638	$2,517

nc = not calculated

The low-density SFP storage alternative for PWR and BWR Mark III reactors with a non-shared SFP total implementation and operation costs is the summation of those costs for the industry and the NRC. As shown in Table 12, the total estimated costs for a single Group 2 SFP to achieve and maintain a low-density SFP loading ranges from $51.40 million (2 percent net present value), to $53.83 million (3 percent net present value), and to $51.36 million (7 percent net present value). The total cost for all 49 SFPs in this group range is approximately $2.56 billion. These costs are dominated by the capital costs for the DSCs and the loading costs for the storage systems to achieve low-density storage in the SFP than that required for the regulatory baseline.

4.4.1.2.3 Spent Fuel Pool Group 3 – New power reactors with non-shared spent fuel pool

Table 13 Summary of Total Implementation and Operation Costs for Low-Density Spent Fuel Pool Storage—Spent Fuel Pool Group 3

Attribute	Costs per SFP (2012 dollars in millions)		
	2% NPV	3% NPV	7% NPV
Occupational Health (Routine)	$0.01	$0.01	$0.01
Industry Implementation	$42.41	$35.75	$16.74
Industry Operation	nc	nc	nc
NRC Implementation	nc	nc	nc
NRC Operation	nc	nc	nc
Total per pool	$42.42	$35.76	$16.75
Total for four pools	$169.7	$143.1	$67.0

nc = not calculated

The low-density SFP storage alternative for new reactors with a non-shared SFP total implementation and operation costs is the summation of those costs for the industry and the NRC. As shown in Table 13, the total estimated costs for a single Group 3 SFP to achieve and maintain a low-density SFP loading ranges from $42.42 million (2 percent net present value), to

$35.76 million (3 percent net present value), and to $16.75 million (7 percent net present value). The total cost for all four SFPs in this group range between $67 and $170 million. These costs are dominated by the capital costs for the DSCs, the loading costs for the storage systems to achieve low-density storage in the SFP, and the additional ISFSI annual operation and maintenance costs required for establishing and storing spent fuel at the ISFSI 15 years earlier than that required for the regulatory baseline.

4.4.1.2.4 Spent Fuel Pool Group 4—Reactor units with a shared spent fuel pool

Table 14 Summary of Total Implementation and Operation Costs for Low-Density Spent Fuel Pool Storage—Spent Fuel Pool Group 4

Attribute	Costs per SFP (2012 dollars in millions)		
	2% NPV	3% NPV	7% NPV
Occupational Health (Routine)	$0.02	$0.02	$0.03
Industry Implementation	$48.78	$50.41	$46.39
Industry Operation	nc	nc	nc
NRC Implementation	nc	nc	nc
NRC Operation	nc	nc	nc
Total per pool	$48.80	$50.43	$46.41
Total for 10 pools	$488.0	$504.3	$464.1

nc = not calculated

The low-density SFP storage alternative for reactor units with a shared SFP total implementation and operation costs is the summation of those costs for the industry and the NRC. As shown in Table 15, the total estimated costs for a single Group 4 shared SFP to achieve and maintain a low-density SFP loading ranges from $48.80 million (2 percent net present value), to $50.43 million (3 percent net present value), and to $46.41 million (7 percent net present value). The total cost for all 10 SFPs in this group range between $511 and $555 million. These costs are dominated by the capital costs for the DSCs, and the loading costs for the storage systems to achieve low-density storage in the SFP than that required for the regulatory baseline.

4.4.1.3 Total Benefits and Cost Offsets

4.4.1.3.1 Spent Fuel Pool Group 1 – BWR Mark I and Mark II reactors with non-shared spent fuel pool

Table 15 Summary of Total Benefits and Cost Offsets for Low-Density Spent Fuel Pool Storage—Spent Fuel Pool Group 1

Attribute	Benefits and Cost Offsets (2012 dollars in millions)		
	2% NPV	3% NPV	7% NPV
Public Health (Accident)	$0.05 - $35.6	$0.04 – $31.7	$0.03 – $21.2
Occupational Health (Accident)	<$0.01 – $0.1	<$0.01 – $0.09	<$0.01 – $0.06
Offsite Property	$0.17 – $85.7	$0.15 – $76.4	$0.10 – $51.1
Onsite Property	<$0.01 – $1.1	<$0.01 – $0.99	<$0.01 – $0.60
Total per pool	$0.24 – $123	$0.21 – $109	$0.15 – $73.0
Total for 31 pools	$7.4 – $3,800	$6.5 – $3,380	$4.7 – $2,260

The SFP Group 1 total benefits are shown in the Table 15. These benefits include the public health (accident) and occupational health (accident) benefits summed with the cost offsets. The cost offsets consists of the sum of the offsite property and onsite property attributes relative to the regulatory baseline. The offsite property cost offset is the largest contributor to the benefits, of which the majority of those costs occur during the long-term phase.

4.4.1.3.2 Spent Fuel Pool Group 2 – PWR and BWR Mark III reactors with non-shared spent fuel pool

Table 16 Summary of Total Benefits and Cost Offsets for Low-Density Spent Fuel Pool Storage—Spent Fuel Pool Group 2

Attribute	Benefits and Cost Offsets (2012 dollars in millions)		
	2% NPV	3% NPV	7% NPV
Public Health (Accident)	$0.06 – $38.7	$0.05 – $34.0	$0.03 – $21.8
Occupational Health (Accident)	<$0.01 – $0.11	<$0.01 – $0.96	<$0.01 – $0.06
Offsite Property	$0.27 – $97.5	$0.24 – $85.6	$0.15 – $54.8
Onsite Property	<$0.01 – $1.2	<$0.01 – $1.0	<$0.01 – $0.59
Total per pool	$0.35 – $138	$0.31 – $122	$0.20 – $77.3
Total for 49 pools	$17 – $6,760	$15 – $5,9800	$10 – $3,790

The SFP Group 2 total benefits are shown in the Table 16. These benefits include the public health (accident) and occupational health (accident) benefits summed with the cost offsets. The cost offsets consists of the sum of the offsite property and onsite property attributes relative to the regulatory baseline. The offsite property cost offset is the largest contributor to the benefits, of which the majority of those costs occur during the long-term phase.

4.4.1.3.3 Spent Fuel Pool Group 3 – AP1000 power reactors with non-shared spent fuel pool

Table 17 Summary of Total Benefits and Cost Offsets for Low-Density Spent Fuel Pool Storage—Spent Fuel Pool Group 3

Attribute	Benefits and Cost Offsets (2012 dollars in millions)		
	2% NPV	3% NPV	7% NPV
Public Health (Accident)	$0.06 – $31.9	$0.05 – $24.1	$0.02 – $10.1
Occupational Health (Accident)	<$0.01 – $0.97	<$0.01 – $0.07	<$0.01 – $0.03
Offsite Property	$0.26 – $74.5	$0.20 – $56.3	$0.08 – $23.5
Onsite Property	<$0.01 – $1.1	<$0.01 – $0.78	<$0.01 – $0.29
Total per pool	$0.34 – $108	$0.27 – $81.3	$0.12 – $33.9
Total for 4 pools	$1.4 – $430	$1.1– $330	$0.5 – $140

The SFP Group 3 total benefits are shown in the Table 17. These benefits include the public health (accident) and occupational health (accident) benefits summed with the cost offsets. The cost offsets consists of the sum of the offsite property and onsite property attributes relative to the regulatory baseline. The offsite property cost offset is the largest contributor to the benefits, of which the majority of those costs occur during the long-term phase.

4.4.1.3.4 Spent Fuel Pool Group 4 – Reactor units with a shared spent fuel pool

Table 18 Summary of Total Benefits and Cost Offsets for Low-Density Spent Fuel Pool Storage—Spent Fuel Pool Group 4

Attribute	Benefits and Cost Offsets (2012 dollars in millions)		
	2% NPV	3% NPV	7% NPV
Public Health (Accident)	$0.06 – $52.1	$0.05 – $46.3	$0.03 – $30.6
Occupational Health (Accident)	<$0.01 – $0.13	<$0.01 – $0.11	<$0.01 – $0.07
Offsite Property	$0.27 – $151.2	$0.24 – $134.3	$0.16 – $88.9
Onsite Property	<$0.01 – $1.3	<$0.01 – $1.2	<$0.01 – $0.70
Total per pool	$0.35 – $205	$0.31 – $182	$0.21 – $120
Total for 10 pools	$3.5 – $2,050	$3.1 – $1,820	$2.1 – $1, 200

The SFP Group 4 total benefits are shown in the Table 18. These benefits include the public health (accident) and occupational health (accident) benefits summed with the cost offsets. The cost offsets consists of the sum of the offsite property and onsite property attributes relative to the regulatory baseline. The offsite property cost offset is the largest contributor to the benefits, of which the majority of those costs occur during the long-term phase.

4.4.1.4 Sensitivity Analysis

This section summarizes the results of the sensitivity analyses that were performed as an additional consideration in performing safety goal screening for the evaluated alternatives. In this section, a low and high estimate is provided that combines the range of expected SFP attributes with conservative assumptions to model the range of pool accidents postulated. These high and low estimates are expected to over and under estimate the consequences from SFP accidents for any individual SFPs assigned to the group.

4.4.1.4.1 Dollar per Person-rem Conversion Factor

The NRC is currently revising the dollar per person-rem averted conversion factor based on recent information regarding the value of a statistical life. However, until the NRC completes the update to NUREG-1530 (Ref. 21) and publishes the appropriate guidance documents, the NRC performs sensitivity analysis to estimate the impact on the calculated results when more current value of a statistical life (VSL) and cancer risk factors are used. The NRC used the EPAs VSL as an interim value in the sensitivity analysis as described in Appendix section D.2. The effect of using the higher dollar per person-rem conversion factor on the calculated results is provided below. As previously discussed, the consequences calculated for the high and low estimate are expected to over and under estimate respectively the consequences if compared to plant-specific SFP analyses within this SFP grouping.

4.4.1.4.1.1 Spent Fuel Pool Group 1—BWR Mark I and Mark II reactors with non-shared spent fuel pool

Table 19 Dollar Per Person-Rem Sensitivity Analysis of Net Benefits for Low-Density Spent Fuel Pool Storage within 50 miles—Group 1 Spent Fuel Pool

Attribute	Low Estimate (2012 dollars)			Base Case (2012 dollars)			High Estimate (2012 dollars)		
	2% NPV	3% NPV	7% NPV	2% NPV	3% NPV	7% NPV	2% NPV	3% NPV	7% NPV
Public Health (Accident)	$96,000	$85,600	$57,200	$5,433,200	$4,845,800	$3,243,000	$71,176,000	$63,482,400	$42,485,000
Occupational Health (Accident)	$1,884	$1,680	$1,124	$17,158	$15,304	$10,242	$210,074	$187,367	$125,394
Offsite Property	$165,692	$147,782	$98,902	$8,959,243	$7,990,830	$5,347,787	$85,673,027	$76,412,549	$51,138,370
Onsite Property	$5,990	$5,280	$3,150	$67,520	$58,650	$35,470	$1,139,040	$989,660	$598,900
Total Benefits	***$269,600***	***$240,300***	***$160,400***	***$14,477,100***	***$12,910,600***	***$8,636,500***	***$158,198,100***	***$141,072,000***	***$94,347,700***
Occupational Health (Routine)	-$50,800	-$55,600	-$56,400	-$50,800	-$55,600	-$56,400	-$50,800	-$55,600	-$56,400
Industry Implementation	-$52,610,000	-$55,170,000	-$52,280,000	-$52,610,000	-$55,170,000	-$52,280,000	-$52,610,000	-$55,170,000	-$52,280,000
Industry Operation	nc	nc	nc	nc	nc	nc	nc	nc	nc
NRC Implementation	nc	nc	nc	nc	nc	nc	nc	nc	nc
NRC Operation	nc	nc	nc	nc	nc	nc	nc	nc	nc
Total Costs	***-$52,660,800***	***-$55,225,600***	***-$52,336,400***	***-$52,660,800***	***-$55,225,600***	***-$52,336,400***	***-$52,660,800***	***-$55,225,600***	***-$52,336,400***
Net Benefit	**-$52,391,000**	**-$54,985,000**	**-$52,176,000**	**-$38,184,000**	**-$42,315,000**	**-$43,700,000**	**$105,537,000**	**$85,846,000**	**$42,011,000**

nc = not calculated

As shown in Table 19, the dollar per person-rem sensitivity analysis does not achieve a positive net benefit for either the low estimate or base case when using a person-rem conversion factor twice as large as the conversion factor in NUREG-1530. When all the high estimates are combined, a positive net benefit is achieved. As Table 4 shows, the base case of the delta benefit for averted public health (accident) radiation exposure from a SFP accident resulting in spent fuel damage is approximately 1,740 person-rem for the Group 1 SFP. This dose represents the reduction of public health risk that results from a policy decision to transfer spent fuel from the SFP to dry storage in order to achieve low-density spent fuel loading in the pool. For a single BWR Mark I or Mark II reactor with a non-shared SFP (Group 1), the averted delta dose exposure is approximately 70 person-rem per year over a remaining licensed commercial operation of the reactor of 24 years (until year 2037). The value is based on a U.S. reactor site average population density of approximately 300 people per square mile within a 50-mile radius from the site. The calculated dose is the difference between an uncontrolled release of radionuclides from a full high-density SFP with no credit for successful mitigation to a full low-density SFP with credit for successful mitigation. The doses reflects the calculated health benefits that result if adherence to the EPA intermediate phase protective action guides that allow a dose of 2 rem in the first year and 500 mrem each year thereafter are used.

4.4.1.4.1.2 Spent Fuel Pool Group 2—PWR and BWR Mark III reactors with non-shared spent fuel pool

The effect of using the higher dollar per person-rem conversion factor on the calculated results is provided below. As previously discussed, the consequences calculated for the high and low estimate are expected to over and under estimate respectively the consequences if compared to plant-specific SFP analyses within this SFP grouping.

Table 20 Dollar Per Person-Rem Sensitivity Analysis of Net Benefits for Low-Density Spent Fuel Pool Storage within 50 miles—Group 2 Spent Fuel Pool

Attribute	Low Estimate (2012 dollars)			Base Case (2012 dollars)			High Estimate (2012 dollars)		
	2% NPV	3% NPV	7% NPV	2% NPV	3% NPV	7% NPV	2% NPV	3% NPV	7% NPV
Public Health (Accident)	$118,000	$103,600	$66,400	$4,896,800	$4,301,400	$2,752,200	$77,482,600	$68,062,000	$43,549,200
Occupational Health (Accident)	$1,000	$800	$600	$13,200	$11,600	$7,400	$218,800	$192,200	$123,000
Offsite Property	$272,584	$239,442	$153,207	$9,031,983	$7,933,837	$5,076,442	$97,457,843	$85,608,518	$54,776,349
Onsite Property	$3,250	$2,840	$1,630	$51,800	$44,280	$25,550	$1,190,370	$1,018,500	$589,050
Total Benefits	***$394,800***	***$346,700***	***$221,800***	***$13,993,800***	***$12,291,100***	***$7,861,600***	***$176,349,600***	***$154,881,200***	***$99,037,600***
Occupational Health (Routine)	-$54,400	-$58,200	-$57,800	-$54,400	-$58,200	-$57,800	-$54,400	-$58,200	-$57,800
Industry Implementation	-$51,370,000	-$53,800,000	-$51,330,000	-$51,370,000	-$53,800,000	-$51,330,000	-$51,370,000	-$53,800,000	-$51,330,000
Industry Operation	nc	nc	nc	nc	nc	nc	nc	nc	nc
NRC Implementation	nc	nc	nc	nc	nc	nc	nc	nc	nc
NRC Operation	nc	nc	nc	nc	nc	nc	nc	nc	nc
Total Costs	***-$51,424,400***	***-$53,858,200***	***-$51,387,800***	***-$51,424,400***	***-$53,858,200***	***-$51,387,800***	***-$51,424,400***	***-$53,858,200***	***-$51,387,800***
Net Benefit	**-$51,030,000**	**-$53,512,000**	**-$51,166,000**	**-$37,431,000**	**-$41,567,000**	**-$43,526,000**	**$124,925,000**	**$101,023,000**	**$47,650,000**

nc = not calculated

As shown in Table 20, the dollar per person-rem sensitivity analysis does not achieve a positive net benefit when using a person-rem conversion factor twice as large as the conversion factor in NUREG-1530 for either the low estimate or base cases. When all the high estimates are combined, a positive net benefit is achieved.

4.4.1.4.1.3 Spent Fuel Pool Group 3—New power reactors with non-shared spent fuel pool

Table 21 Dollar Per Person-Rem Sensitivity Analysis of Net Benefits for Low-Density Spent Fuel Pool Storage within 50 miles—Group 3 Spent Fuel Pool

Attribute	Low Estimate (2012 dollars)			Base Case (2012 dollars)			High Estimate (2012 dollars)		
	2% NPV	3% NPV	7% NPV	2% NPV	3% NPV	7% NPV	2% NPV	3% NPV	7% NPV
Public Health (Accident)	$129,600	$98,000	$41,000	$6,279,200	$4,748,800	$1,981,600	$63,827,600	$48,271,400	$20,143,000
Occupational Health (Accident)	$1,400	$1,200	$400	$19,000	$14,400	$6,000	$193,400	$146,200	$61,000
Offsite Property	$264,273	$199,864	$83,400	$11,451,619	$8,660,606	$3,613,942	$74,506,474	$56,347,594	$23,513,013
Onsite Property	$4,780	$3,560	$1,320	$75,910	$55,830	$20,950	$1,062,030	$781,900	$293,960
Total Benefits	***$400,100***	***$302,600***	***$126,100***	***$17,825,700***	***$13,479,600***	***$5,622,500***	***$139,589,500***	***$105,547,100***	***$44,011,000***
Occupational Health (Routine)	-$29,000	-$25,800	-$12,800	-$29,000	-$25,800	-$12,800	-$29,000	-$25,800	-$12,800
Industry Implementation	-$42,410,000	-$35,750,000	-$16,740,000	-$42,410,000	-$35,750,000	-$16,740,000	-$42,410,000	-$35,750,000	-$16,740,000
Industry Operation	nc	nc	nc	nc	nc	nc	nc	nc	nc
NRC Implementation	nc	nc	nc	nc	nc	nc	nc	nc	nc
NRC Operation	nc	nc	nc	nc	nc	nc	nc	nc	nc
Total Costs	***-$42,439,000***	***-$35,775,800***	***-$16,752,800***	***-$42,439,000***	***-$35,775,800***	***-$16,752,800***	***-$42,439,000***	***-$35,775,800***	***-$16,752,800***
Net Benefit	**-$42,039,000**	**-$35,473,000**	**-$16,627,000**	**-$24,613,000**	**-$22,296,000**	**-$11,130,000**	**$97,151,000**	**$69,771,000**	**$27,258,000**

nc = not calculated

As shown in Table 21, the dollar per person-rem sensitivity analysis does not achieve a positive net benefit when using a person-rem conversion factor twice as large as the conversion factor in NUREG-1530 for either the low estimate or base cases presented. The high estimates show a positive net benefit of between $27 and $97 million. This SFP group differs significantly from the other SFP groups analyzed in that these pools have not yet been constructed so that there is not a significant front ended DSC procurement cost difference between the two alternatives. However, in comparison to the base case, the high estimate includes additional conservative assumptions regarding seismic fragilities, release fractions, SFP inventories, long-term habitability criteria, and site population densities that are overly conservative for the four units with combined licenses.

4.4.1.4.1.4 Spent Fuel Pool Group 4—Reactor units with a shared spent fuel pool

Table 22 Dollar Per Person-Rem Sensitivity Analysis of Net Benefits for Low-Density Spent Fuel Pool Storage within 50 miles—Group 4 Spent Fuel Pool

Attribute	Low Estimate (2012 dollars)			Base Case (2012 dollars)			High Estimate (2012 dollars)		
	2% NPV	3% NPV	7% NPV	2% NPV	3% NPV	7% NPV	2% NPV	3% NPV	7% NPV
Public Health (Accident)	$114,400	$101,600	$67,200	$5,246,400	$4,661,400	$3,083,600	$104,286,600	$92,655,000	$61,292,600
Occupational Health (Accident)	$1,000	$800	$600	$12,000	$10,800	$7,200	$250,000	$222,200	$147,000
Offsite Property	$271,158	$240,914	$159,368	$9,805,063	$8,711,458	$5,762,750	$151,185,571	$134,323,136	$88,856,614
Onsite Property	$3,050	$2,640	$1,520	$47,620	$41,160	$24,570	$1,349,250	$1,168,370	$700,210
Total Benefits	***$389,600***	***$346,000***	***$228,700***	***$15,111,100***	***$13,424,800***	***$8,878,100***	***$257,071,400***	***$228,368,700***	***$150,996,400***
Occupational Health (Routine)	$45,400	$49,400	$49,600	$45,400	$49,400	$49,600	$45,400	$49,400	$49,600
Industry Implementation	$48,780,000	$50,410,000	$46,390,000	$48,780,000	$50,410,000	$46,390,000	$48,780,000	$50,410,000	$46,390,000
Industry Operation	nc	nc	nc	nc	nc	nc	nc	nc	nc
NRC Implementation	nc	nc	nc	nc	nc	nc	nc	nc	nc
NRC Operation	nc	nc	nc	nc	nc	nc	nc	nc	nc
Total Costs	***$48,825,400***	***$50,459,400***	***$46,439,600***	***$48,825,400***	***$50,459,400***	***$46,439,600***	***$48,825,400***	***$50,459,400***	***$46,439,600***
Net Benefit	**-$48,436,000**	**-$50,113,000**	**-$46,211,000**	**-$33,714,000**	**-$37,035,000**	**-$37,562,000**	**$208,246,000**	**$177,909,000**	**$104,557,000**

nc = not calculated

As shown in Table 22, the dollar per person-rem sensitivity analysis does not achieve a positive net benefit when using a person-rem conversion factor twice as large as the conversion factor in NUREG-1530 for either the low estimate or base case presented. The high estimate shows a positive net benefit of between $105 and $208 million.

4.4.1.4.2 Consequences Extending Beyond 50 Miles

The Regulatory Analysis Handbook states that in the case of nuclear power plants, changes in public health and safety from radiation exposure and offsite property impacts should be examined over a 50-mile distance from the plant site, although alternative distances from the plant may be used for sensitivity analyses. For this cost-benefit analysis, supplemental information (e.g., analyses and results) based on MACCS2 calculated results, is performed which extends the analysis to consider consequences beyond 50 miles for each SFP group.

4.4.1.4.2.1 Spent Fuel Pool Group 1 – BWR Mark I and Mark II reactors with non-shared spent fuel pool

Table 23 Consequences Extending Beyond 50 Miles Sensitivity Analysis of Net Benefits for Low-Density Spent Fuel Pool Storage—Group 1 Spent Fuel Pool

Attribute	Low Estimate (2012 dollars)			Base Case (2012 dollars)			High Estimate (2012 dollars)		
	2% NPV	3% NPV	7% NPV	2% NPV	3% NPV	7% NPV	2% NPV	3% NPV	7% NPV
Public Health (Accident)	$503,300	$448,900	$300,400	$22,835,700	$20,367,300	$13,630,700	$305,431,900	$272,417,500	$182,312,800
Occupational Health (Accident)	$942	$840	$562	$8,579	$7,652	$5,121	$105,037	$93,684	$62,697
Offsite Property	$573,290	$511,323	$342,198	$16,358,429	$14,590,231	$9,764,373	$323,691,221	$288,703,133	$193,211,821
Onsite Property	$5,990	$5,280	$3,150	$67,520	$58,650	$35,470	$1,139,040	$989,660	$598,900
Total Benefits	***$1,083,500***	***$966,300***	***$646,300***	***$39,270,200***	***$35,023,800***	***$23,435,700***	***$630,367,200***	***$562,204,000***	***$376,186,200***
Occupational Health (Routine)	-$25,400	-$27,800	-$28,200	-$25,400	-$27,800	-$28,200	-$25,400	-$27,800	-$28,200
Industry Implementation	-$52,610,000	-$55,170,000	-$52,280,000	-$52,610,000	-$55,170,000	-$52,280,000	-$52,610,000	-$55,170,000	-$52,280,000
Industry Operation	nc	nc	nc	nc	nc	nc	nc	nc	nc
NRC Implementation	nc	nc	nc	nc	nc	nc	nc	nc	nc
NRC Operation	nc	nc	nc	nc	nc	nc	nc	nc	nc
Total Costs	***-$52,635,400***	***-$55,197,800***	***-$52,308,200***	***-$52,635,400***	***-$55,197,800***	***-$52,308,200***	***-$52,635,400***	***-$55,197,800***	***-$52,308,200***
Net Benefit	**-$51,552,000**	**-$54,232,000**	**-$51,662,000**	**-$13,365,000**	**-$20,174,000**	**-$28,873,000**	**$577,732,000**	**$507,006,000**	**$323,878,000**

nc = not calculated

As shown in Table 23, calculated net benefits for requiring low-density SFP storage when considering consequences beyond 80 kilometers (50 miles) does not achieve a positive net benefit for either the low estimate or base cases presented. The high estimates show a positive

net benefit of between $324 and $578 million. In comparison to the base case, the high estimate includes additional conservative assumptions regarding seismic fragilities, release fractions, SFP inventories, long-term habitability criteria, and site population densities that when taken together result in a net beneficial result.

4.4.1.4.2.2 Spent Fuel Pool Group 2—PWR and BWR Mark III reactors with nonshared spent fuel pool

Table 24 Consequences Extending Beyond 50 Miles Sensitivity Analysis of Net Benefits for Low-Density Spent Fuel Pool Storage—Group 2 Spent Fuel Pool

Attribute	Low Estimate (2012 dollars)			Base Case (2012 dollars)			High Estimate (2012 dollars)		
	2% NPV	3% NPV	7% NPV	2% NPV	3% NPV	7% NPV	2% NPV	3% NPV	7% NPV
Public Health (Accident)	$860,600	$755,900	$483,700	$20,609,300	$18,103,500	$11,583,500	$350,842,800	$308,185,900	$197,191,800
Occupational Health (Accident)	$500	$400	$300	$6,600	$5,800	$3,700	$109,400	$96,100	$61,500
Offsite Property	$1,860,702	$1,634,470	$1,045,811	$28,788,238	$25,288,046	$16,180,479	$402,559,059	$353,614,274	$226,259,013
Onsite Property	$3,250	$2,840	$1,630	$51,800	$44,280	$25,550	$201,170	$173,800	$103,350
Total Benefits	***$2,725,100***	***$2,393,600***	***$1,531,400***	***$49,455,900***	***$43,441,600***	***$27,793,200***	***$753,712,400***	***$662,070,100***	***$423,615,700***
Occupational Health (Routine)	-$27,200	-$29,100	-$28,900	-$27,200	-$29,100	-$28,900	-$27,200	-$29,100	-$28,900
Industry Implementation	-$51,370,000	-$53,800,000	-$51,330,000	-$51,370,000	-$53,800,000	-$51,330,000	-$51,370,000	-$53,800,000	-$51,330,000
Industry Operation	nc	nc	nc	nc	nc	nc	nc	nc	nc
NRC Implementation	nc	nc	nc	nc	nc	nc	nc	nc	nc
NRC Operation	nc	nc	nc	nc	nc	nc	nc	nc	nc
Total Costs	***-$51,397,200***	***-$53,829,100***	***-$51,358,900***	***-$51,397,200***	***-$53,829,100***	***-$51,358,900***	***-$51,397,200***	***-$53,829,100***	***-$51,358,900***
Net Benefit	**-$48,672,000**	**-$51,436,000**	**-$49,828,000**	**-$1,941,000**	**-$10,388,000**	**-$23,566,000**	**$702,315,000**	**$608,241,000**	**$372,257,000**

nc = not calculated

As shown in Table 24, calculated net benefits for requiring low-density SFP storage when considering consequences beyond 80 kilometers (50 miles) does not achieve a positive net benefit for either the low estimate or base cases presented. The high estimates show a positive net benefit of between $372 and $702 million. In comparison to the base case, the high estimate includes additional conservative assumptions regarding seismic fragilities, release fractions, SFP inventories, long-term habitability criteria, and site population densities that when taken together result in a net beneficial result.

4.4.1.4.2.3 Spent Fuel Pool Group 3 – New power reactors with non-shared spent fuel pool

Table 25 Consequences Extending Beyond 50 Miles Sensitivity Analysis of Net Benefits for Low-Density Spent Fuel Pool Storage—Group 3 Spent Fuel Pool

Attribute	Low Estimate (2012 dollars)			Base Case (2012 dollars)			High Estimate (2012 dollars)		
	2% NPV	3% NPV	7% NPV	2% NPV	3% NPV	7% NPV	2% NPV	3% NPV	7% NPV
Public Health (Accident)	$844,600	$638,700	$266,500	$23,666,800	$17,898,700	$7,468,900	$263,568,800	$199,331,200	$83,178,000
Occupational Health (Accident)	$700	$600	$200	$9,500	$7,200	$3,000	$96,700	$73,100	$30,500
Offsite Property	$1,546,992	$1,169,956	$488,205	$27,166,671	$20,545,551	$8,573,353	$262,776,843	$198,732,300	$82,928,034
Onsite Property	$4,780	$3,560	$1,320	$75,910	$55,830	$20,950	$1,062,030	$781,900	$293,960
Total Benefits	***$2,397,100***	***$1,812,800***	***$756,200***	***$50,918,900***	***$38,507,300***	***$16,066,200***	***$527,504,400***	***$398,918,500***	***$166,430,500***
Occupational Health (Routine)	-$14,500	-$12,900	-$6,400	-$14,500	-$12,900	-$6,400	-$14,500	-$12,900	-$6,400
Industry Implementation	-$42,410,000	-$35,750,000	-$16,740,000	-$42,410,000	-$35,750,000	-$16,740,000	-$42,410,000	-$35,750,000	-$16,740,000
Industry Operation	nc	nc	nc	nc	nc	nc	nc	nc	nc
NRC Implementation	nc	nc	nc	nc	nc	nc	nc	nc	nc
NRC Operation	nc	nc	nc	nc	nc	nc	nc	nc	nc
Total Costs	***-$42,424,500***	***-$35,762,900***	***-$16,746,400***	***-$42,424,500***	***-$35,762,900***	***-$16,746,400***	***-$42,424,500***	***-$35,762,900***	***-$16,746,400***
Net Benefit	**-$40,027,000**	**-$33,950,000**	**-$15,990,000**	**$8,494,000**	**$2,744,000**	**-$680,000**	**$485,080,000**	**$363,156,000**	**$149,684,000**

nc = not calculated

As shown in Table 25, the dollar per person-rem sensitivity analysis does not achieve a positive net benefit when considering consequences beyond 80 kilometers (50 miles) for four of the nine cases presented. Two cases, the 2-percent and 3-percent discounted base cases and the high estimates show a positive net benefit range of between $2.7 and $485 million.

4.4.1.4.2.4 Spent Fuel Pool Group 4—Reactor units with a shared spent fuel pool

Table 26 Consequences Extending Beyond 50 Miles Sensitivity Analysis of Net Benefits for Low-Density Spent Fuel Pool Storage—Group 4 Spent Fuel Pool

Attribute	Low Estimate (2012 dollars)			Base Case (2012 dollars)			High Estimate (2012 dollars)		
	2% NPV	3% NPV	7% NPV	2% NPV	3% NPV	7% NPV	2% NPV	3% NPV	7% NPV
Public Health (Accident)	$853,200	$758,100	$501,500	$24,572,200	$21,831,600	$14,441,900	$560,905,000	$498,344,700	$329,661,900
Occupational Health (Accident)	$500	$400	$300	$6,000	$5,400	$3,600	$125,000	$111,100	$73,500
Offsite Property	$1,898,771	$1,686,992	$1,115,969	$39,619,961	$35,200,961	$23,285,923	$779,796,081	$692,821,772	$458,311,191
Onsite Property	$3,050	$2,640	$1,520	$47,620	$41,160	$24,570	$1,349,250	$1,168,370	$700,210
Total Benefits	***$2,755,500***	***$2,448,100***	***$1,619,300***	***$64,245,800***	***$57,079,100***	***$37,756,000***	***$1,342,175,300***	***$1,192,445,900***	***$788,746,800***
Occupational Health (Routine)	-$22,700	-$24,700	-$24,800	-$22,700	-$24,700	-$24,800	-$22,700	-$24,700	-$24,800
Industry Implementation	-$48,780,000	-$50,410,000	-$46,390,000	-$48,780,000	-$50,410,000	-$46,390,000	-$48,780,000	-$50,410,000	-$46,390,000
Industry Operation	nc	nc	nc	nc	nc	nc	nc	nc	nc
NRC Implementation	nc	nc	nc	nc	nc	nc	nc	nc	nc
NRC Operation	nc	nc	nc	nc	nc	nc	nc	nc	nc
Total Costs	***-$48,802,700***	***-$50,434,700***	***-$46,414,800***	***-$48,802,700***	***-$50,434,700***	***-$46,414,800***	***-$48,802,700***	***-$50,434,700***	***-$46,414,800***
Net Benefit	**-$46,047,000**	**-$47,987,000**	**-$44,796,000**	**$15,443,000**	**$6,644,000**	**-$8,659,000**	**$1,293,373,000**	**$1,142,011,000**	**$742,332,000**

nc = not calculated

As shown in Table 26, the dollar per person-rem sensitivity analysis does not achieve a positive net benefit when considering consequences beyond 80 kilometers (50 miles) for four of the nine cases presented. Two cases, the 2-percent and 3-percent discounted base cases and the high estimates show a positive net benefit range of between $6.6 and $1,293 million.

4.4.1.4.3 Combined Effect of Consequences Extending Beyond 50 Miles and Dollar per Person-Rem Conversion Factor

This sensitivity analysis considers the combined effects of extending the analysis of consequences beyond 50 miles from the site and increasing the dollar per person-rem conversion value from $2,000 to $4,000 per person-rem averted. The combined effects of these two variables on the calculated net benefits are provided below.

4.4.1.4.3.1 Spent Fuel Pool Group 1 – BWR Mark I and Mark II reactors with non-shared spent fuel pool

Table 27 Combined Sensitivity Analysis that Analyzes Consequences beyond 50 Miles Using a Revised Dollar per Person-Rem Conversion Factor on the Net Benefits for Low-Density Spent Fuel Pool Storage—Group 1 Spent Fuel Pool

Attribute	Low Estimate (2012 dollars)			Base Case (2012 dollars)			High Estimate (2012 dollars)		
	2% NPV	3% NPV	7% NPV	2% NPV	3% NPV	7% NPV	2% NPV	3% NPV	7% NPV
Public Health (Accident)	$1,006,600	$897,800	$600,800	$45,671,400	$40,734,600	$27,261,400	$610,863,800	$544,835,000	$364,625,600
Occupational Health (Accident)	$1,884	$1,680	$1,124	$17,158	$15,304	$10,242	$210,074	$187,367	$125,394
Offsite Property	$573,290	$511,323	$342,198	$16,358,429	$14,590,231	$9,764,373	$323,691,221	$288,703,133	$193,211,821
Onsite Property	$5,990	$5,280	$3,150	$67,520	$58,650	$35,470	$1,139,040	$989,660	$598,900
Total Benefits	***$1,587,800***	***$1,416,100***	***$947,300***	***$62,114,500***	***$55,398,800***	***$37,071,500***	***$935,904,100***	***$834,715,200***	***$558,561,700***
Occupational Health (Routine)	-$50,800	-$55,600	-$56,400	-$50,800	-$55,600	-$56,400	-$50,800	-$55,600	-$56,400
Industry Implementation	-$52,610,000	-$55,170,000	-$52,280,000	-$52,610,000	-$55,170,000	-$52,280,000	-$52,610,000	-$55,170,000	-$52,280,000
Industry Operation	nc	nc	nc	nc	nc	nc	nc	nc	nc
NRC Implementation	nc	nc	nc	nc	nc	nc	nc	nc	nc
NRC Operation	nc	nc	nc	nc	nc	nc	nc	nc	nc
Total Costs	***-$52,660,800***	***-$55,225,600***	***-$52,336,400***	***-$52,660,800***	***-$55,225,600***	***-$52,336,400***	***-$52,660,800***	***-$55,225,600***	***-$52,336,400***
Net Benefit	**-$51,073,000**	**-$53,810,000**	**-$51,389,000**	**$9,454,000**	**$173,000**	**-$15,265,000**	**$883,243,000**	**$779,490,000**	**$506,225,000**

nc = not calculated

As shown in Table 27, calculated net benefits for requiring low-density SFP storage when considering consequences beyond 50 miles combined with a revised dollar per person-rem

conversion factor does not achieve a positive net benefit for four of the nine cases presented. Two cases, the 2-percent and 3-percent discounted base cases and the high estimates show a positive net benefit range of between $173,000 and $883 million.

4.4.1.4.3.2 Spent Fuel Pool Group 2—PWR and BWR Mark III reactors with nonshared spent fuel pool

Table 28 Combined Sensitivity Analysis that Analyzes Consequences beyond 50 Miles Using a Revised Dollar per Person-Rem Conversion Factor on the Net Benefits for Low-Density Spent Fuel Pool Storage—Group 2 Spent Fuel Pool

Attribute	Low Estimate (2012 dollars)			Base Case (2012 dollars)			High Estimate (2012 dollars)		
	2% NPV	3% NPV	7% NPV	2% NPV	3% NPV	7% NPV	2% NPV	3% NPV	7% NPV
Public Health (Accident)	$1,721,200	$1,511,800	$967,400	$41,218,600	$36,207,000	$23,167,000	$701,685,600	$616,371,800	$394,383,600
Occupational Health (Accident)	$1,000	$800	$600	$13,200	$11,600	$7,400	$218,800	$192,200	$123,000
Offsite Property	$1,860,702	$1,634,470	$1,045,811	$28,788,238	$25,288,046	$16,180,479	$402,559,059	$353,614,274	$226,259,013
Onsite Property	$3,250	$2,840	$1,630	$51,800	$44,280	$25,550	$201,170	$173,800	$103,350
Total Benefits	***$3,586,200***	***$3,149,900***	***$2,015,400***	***$70,071,800***	***$61,550,900***	***$39,380,400***	***$1,104,664,600***	***$970,352,100***	***$620,869,000***
Occupational Health (Routine)	-$54,400	-$58,200	-$57,800	-$54,400	-$58,200	-$57,800	-$54,400	-$58,200	-$57,800
Industry Implementation	-$51,370,000	-$53,800,000	-$51,330,000	-$51,370,000	-$53,800,000	-$51,330,000	-$51,370,000	-$53,800,000	-$51,330,000
Industry Operation	nc	nc	nc	nc	nc	nc	nc	nc	nc
NRC Implementation	nc	nc	nc	nc	nc	nc	nc	nc	nc
NRC Operation	nc	nc	nc	nc	nc	nc	nc	nc	nc
Total Costs	***-$51,424,400***	***-$53,858,200***	***-$51,387,800***	***-$51,424,400***	***-$53,858,200***	***-$51,387,800***	***-$51,424,400***	***-$53,858,200***	***-$51,387,800***
Net Benefit	**-$47,838,000**	**-$50,708,000**	**-$49,372,000**	**$18,647,000**	**$7,693,000**	**-$12,007,000**	**$1,053,240,000**	**$916,494,000**	**$569,481,000**

nc = not calculated

As shown in Table 28, calculated net benefits for requiring low-density SFP storage when considering consequences beyond 50-miles combined with a revised dollar per person-rem conversion factor does not achieve a positive net benefit for four of the nine cases presented. Two cases, the 2-percent and 3-percent discounted base cases and the high estimates show a positive net benefit range of between $7.7 and $1,053 million.

4.4.1.4.3.3 Spent Fuel Pool Group 3 – AP1000 power reactors with non-shared spent fuel pool

Table 29 Combined Sensitivity Analysis that Analyzes Consequences beyond 50 Miles Using a Revised Dollar per Person-Rem Conversion Factor on the Net Benefits for Low-Density Spent Fuel Pool Storage—Group 3 Spent Fuel Pool

Attribute	Low Estimate (2012 dollars)			Base Case (2012 dollars)			High Estimate (2012 dollars)		
	2% NPV	3% NPV	7% NPV	2% NPV	3% NPV	7% NPV	2% NPV	3% NPV	7% NPV
Public Health (Accident)	$1,689,200	$1,277,400	$533,000	$47,333,600	$35,797,400	$14,937,800	$527,137,600	$398,662,400	$166,356,000
Occupational Health (Accident)	$1,400	$1,200	$400	$19,000	$14,400	$6,000	$193,400	$146,200	$61,000
Offsite Property	$1,546,992	$1,169,956	$488,205	$27,166,671	$20,545,551	$8,573,353	$262,776,843	$198,732,300	$82,928,034
Onsite Property	$4,780	$3,560	$1,320	$75,910	$55,830	$20,950	$1,062,030	$781,900	$293,960
Total Benefits	***$3,242,400***	***$2,452,100***	***$1,022,900***	***$74,595,200***	***$56,413,200***	***$23,538,100***	***$791,169,900***	***$598,322,800***	***$249,639,000***
Occupational Health (Routine)	-$29,000	-$25,800	-$12,800	-$29,000	-$25,800	-$12,800	-$29,000	-$25,800	-$12,800
Industry Implementation	-$42,410,000	-$35,750,000	-$16,740,000	-$42,410,000	-$35,750,000	-$16,740,000	-$42,410,000	-$35,750,000	-$16,740,000
Industry Operation	nc	nc	nc	nc	nc	nc	nc	nc	nc
NRC Implementation	nc	nc	nc	nc	nc	nc	nc	nc	nc
NRC Operation	nc	nc	nc	nc	nc	nc	nc	nc	nc
Total Costs	***-$42,439,000***	***-$35,775,800***	***-$16,752,800***	***-$42,439,000***	***-$35,775,800***	***-$16,752,800***	***-$42,439,000***	***-$35,775,800***	***-$16,752,800***
Net Benefit	**-$39,197,000**	**-$33,324,000**	**-$15,730,000**	**$32,156,000**	**$20,637,000**	**$6,785,000**	**$748,731,000**	**$562,547,000**	**$232,886,000**

nc = not calculated

As shown in Table 29, calculated net benefits for requiring low-density SFP storage when considering consequences beyond 50 miles combined with a revised dollar per person-rem conversion factor does not achieve a positive net benefit for the low estimate cases presented. The base cases and high estimates show a positive net benefit range of between $6.8 and $748 million.

4.4.1.4.3.4 Spent Fuel Pool Group 4 – Reactor units with a shared spent fuel pool

Table 30 Combined Sensitivity Analysis that Analyzes Consequences beyond 50 Miles Using a Revised Dollar per Person-Rem Conversion Factor on the Net Benefits for Low-Density Spent Fuel Pool Storage—Group 4 Spent Fuel Pool

Attribute	Low Estimate (2012 dollars)			Base Case (2012 dollars)			High Estimate (2012 dollars)		
	2% NPV	3% NPV	7% NPV	2% NPV	3% NPV	7% NPV	2% NPV	3% NPV	7% NPV
Public Health (Accident)	$1,706,400	$1,516,200	$1,003,000	$49,144,400	$43,663,200	$28,883,800	$1,121,810,000	$996,689,400	$659,323,800
Occupational Health (Accident)	$1,000	$800	$600	$12,000	$10,800	$7,200	$250,000	$222,200	$147,000
Offsite Property	$1,898,771	$1,686,992	$1,115,969	$39,619,961	$35,200,961	$23,285,923	$779,796,081	$692,821,772	$458,311,191
Onsite Property	$3,050	$2,640	$1,520	$47,620	$41,160	$24,570	$1,349,250	$1,168,370	$700,210
Total Benefits	***$3,609,200***	***$3,206,600***	***$2,121,100***	***$88,824,000***	***$78,916,100***	***$52,201,500***	***$1,903,205,300***	***$1,690,901,700***	***$1,118,482,200***
Occupational Health (Routine)	-$45,400	-$49,400	-$49,600	-$45,400	-$49,400	-$49,600	-$45,400	-$49,400	-$49,600
Industry Implementation	-$48,780,000	-$50,410,000	-$46,390,000	-$48,780,000	-$50,410,000	-$46,390,000	-$48,780,000	-$50,410,000	-$46,390,000
Industry Operation	nc	nc	nc	nc	nc	nc	nc	nc	nc
NRC Implementation	nc	nc	nc	nc	nc	nc	nc	nc	nc
NRC Operation	nc	nc	nc	nc	nc	nc	nc	nc	nc
Total Costs	***-$48,825,400***	***-$50,459,400***	***-$46,439,600***	***-$48,825,400***	***-$50,459,400***	***-$46,439,600***	***-$48,825,400***	***-$50,459,400***	***-$46,439,600***
Net Benefit	**-$45,216,000**	**-$47,253,000**	**-$44,319,000**	**$39,999,000**	**$28,457,000**	**$5,762,000**	**$1,854,380,000**	**$1,640,442,000**	**$1,072,043,000**

nc = not calculated

As shown in Table 30, calculated net benefits for requiring low-density SFP storage when considering consequences beyond 50-miles combined with a revised dollar per person-rem conversion factor does not achieve a positive net benefit for the low estimate cases presented. The base cases and high estimates show a positive net benefit range of between $5.8 and $1,854 million.

4.4.2 **Disaggregation**

In order to comply with the guidance provided in Section 4.3.2, “Criteria for the Treatment of Individual Requirements” of the Regulatory Analysis Guidelines (Ref. 3), the NRC conducted a screening review to ensure that the aggregate analysis does not mask the inclusion of individual requirements that are not cost-beneficial when considered individually and not necessary to meet the stated objectives. Consistent with the Regulatory Analysis Guidelines, the NRC evaluated, on a disaggregated basis, each new regulatory provision expected to result in incremental costs. Based on this screening review, the NRC did not identify any requirements needing further consideration. The NRC believes that each of these provisions described in Appendix section C.3 is necessary in the aggregate for the expedited transfer of spent fuel to DSCs. However, the NRC finds that requiring the accelerated transfer to DSCs would provide only limited safety benefits, far below the threshold that the NRC uses to inform it regulatory decisionmaking, and would not be cost-justified.

4.5 <u>Decision Rationale</u>

This section presents the decision rationale, including the basis for selection, and the decision criteria used.

Table 10 shows that the calculated benefits for requiring the low-density SFP storage alternative for the low estimate and base case are less than industry costs to achieve a low-density spent fuel loading pattern for each SFP group. As might be expected for estimates that include a compounding of the most conservative assumptions, all of the SFP group high estimate cases result in calculated benefits that are greater than the estimated costs.

Similar to the seismic event analyzed for the SFPS, no offsite early fatalities are calculated to occur. This results from the following two reasons:

(1) In comparison to reactors, SFPs have a larger proportion of longer-lived radionuclides, which are less likely to cause the significant doses required for acute health effects.

(2) Despite the large releases for certain predicted SFP accident progressions, the release from the most recently discharged fuel (which contains the shorter-lived radionuclides) is predicted to be insufficiently fast and insufficiently large to reach the acute thresholds associated with offsite early fatalities. When doses do exceed minimum levels for early fatalities, emergency response effectively prevents any early fatality risk, at least in part because the modeled accident progression results in releases that are long compared with the time needed for relocation.

In addition, the predicted long-term exposure of the population, which could result in latent cancer fatality risk, is also low for the following reasons:

(1) The individual latent individual latent cancer fatality risk within 0 to 10 miles is predicted to be on the order of 2.4×10^{-10} to 1.5×10^{-8} per year, based on the linear no threshold (LNT) dose response model.

(2) The risk within 10 miles of the analyzed accident is dominated by low dose received at a low dose rate. Using truncation levels that do not quantify the effects of doses below 620 mrem per year (i.e., those arising from representative background radiation including average annual medical exposures) reduces the estimated individual LCF risk by up to a few orders of magnitude for the accident as modeled.

(3) Average individual latent cancer fatality risk is low but decreases slowly as a function of distance from the plant. Additionally, the predicted individual risks of latent cancer fatalities are dominated by long-term exposures to very lightly contaminated areas for which doses are small enough to be considered habitable.

Sensitivity studies provided in Section 4.4.1.6 show that there are cases using conservative assumptions in each sensitivity study in which the low-density spent fuel storage alternative was cost-justified. However, after considering the analysis results, operating history, and limited safety benefits of possible plant changes, the staff finds that further study would be unlikely to support future actions requiring expedited transfer.

The NRC staff identified other considerations discussed in Section 4.3.10 that would further reduce the quantified benefits and make the proposed alternative less justifiable.

The outcome of this cost-benefit analysis indicates that undertaking additional study of the low-density SFP storage alternative is not justified. Except in those cases where action is needed to ensure adequate protection of public health and safety, the process used by the NRC when considering additional regulatory requirements is to assess the potential benefits from new regulations against a safety benefit threshold (e.g., the safety goal screening) and the costs of implementing new requirements. The potential benefits of a requirement to expedite the removal of spent fuel from storage pools could be to reduce the risk to the public from possible accidents involving SFPs. Assessments of risk and changes in risk from possible actions involve identifying what can go wrong, what are the consequences, and how likely is it to occur.

In the case of hypothetical accidents involving SFPs, the assessments have shown that impacts on public health and safety can be avoided but that the potential economic consequences can be very large. However, the assessments also show that the design and construction of SFPs, the characteristics of the spent fuel assemblies, and the availability of mitigating systems result in a very low likelihood that radioactive materials would be released because of an accident affecting a SFP. This evaluation of a low probability, high consequence event is similar to previous NRC risk assessments and related regulatory analyses for potential issues related to nuclear reactor and SFPs.

Based on the NRC's assessment of the costs and benefits, the agency has concluded that the risk of beyond-design-basis accidents in SFPs, while not negligible, is sufficiently low, far below the threshold NRC uses to inform its regulatory decisionmaking, and that the added costs involved with expediting the movement of spent fuel from the pool to achieve low-density fuel pool storage is not warranted.

5. CONCLUSION

To determine if additional studies are needed to further assess potential regulatory action on expedited transfer, the staff has conducted an analysis of expedited transfer of spent fuel to dry cask storage, in accordance with the agency's current policies and guidance.

The safety goal screening evaluation concludes that SFP accidents are a small contributor to the overall risks for public health and safety (less than one percent of the QHOs) and therefore any reductions in risk associated with expedited transfer of spent fuel would only have a marginal safety benefit. Due to the safety goal screening criterion not being satisfied, the staff finds that no further generic assessments are warranted. Although the regulatory analysis guidelines would normally stop the evaluation at this step because the risk is a small fraction of the safety goals, the staff proceeded to perform a cost-benefit analysis to provide additional information for the Commission's consideration.

The staff conducted a cost-benefit analysis, which finds that the added costs involved with expedited transfer of spent fuel to dry cask storage to achieve the low-density SFP storage alternative are not warranted in light of the marginal safety benefits from such an action. The combination of high estimates for important parameters assumed in some of the sensitivity cases presented in this analysis result in large economic consequences, such that, the calculated benefits from expedited transfer of spent fuel to dry cask storage for those cases outweigh the associated costs. However, even in these cases, there is only a marginal safety improvement in terms of public health and safety. In the staff's judgment, the assumptions made in this analysis were selected in a generally conservative manner such that the base case is the primary basis for the staff's recommendation. Based on the generic assessment and the other considerations detailed in this paper, the staff finds that additional studies are not needed to reasonably conclude that the expedited transfer of spent fuel to dry cask storage would provide only a marginal increase in the overall protection of public health and safety, and would not be warranted due to the expected implementation costs.

No further regulatory action is recommended for the resolution of this issue. The outcome of this cost-benefit analysis indicates that undertaking additional study of the low-density SFP storage alternative is not justified.

6. REFERENCES

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8. PRM-51-12, "Proposed Amendment to 10 CFR Part 51 (Rescinding finding that environmental impacts of pool storage of spent nuclear fuel are insignificant), dated March 16, 2007.

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APPENDIX A: SPENT FUEL POOL CHARACTERISTICS

A.1 Spent Fuel Pool Configurations

The configuration of spent fuel storage pools is similar for most nuclear reactor and away-from-reactor storage facilities. The pools are rectangular in cross section and approximately 12 meters (40 feet) deep. Fuel assemblies are placed vertically in storage racks that maintain an adequate spacing to prevent criticality and to promote natural convective cooling in a water medium. The pools themselves are constructed of reinforced concrete with sufficient thickness to meet radiation shielding and structural requirements, and are lined with stainless steel plates of approximately 2.5-center (1/4-inch) thickness to ensure a leak-tight system.

A.1.1 Boiling-Water Reactors with Mark I and Mark II Containments

Boiling-water reactors (BWRs) with Mark I and Mark II containments are designed with the SFP located within the reactor building as shown in Figure 3. The bottom of the SFP is usually elevated approximately 15 meters (50 feet) above grade, which places the top of the pool at the level of the operating floor. The enclosing superstructure above the pool is typically a low-leakage steel, industrial-type building designed to house cranes that are used to move reactor components, spent fuel, and spent fuel casks. For a few reactor buildings, the enclosing superstructure is a reinforced concrete structure with strength similar to the lower portions of the reactor building, as depicted in Figure 3.

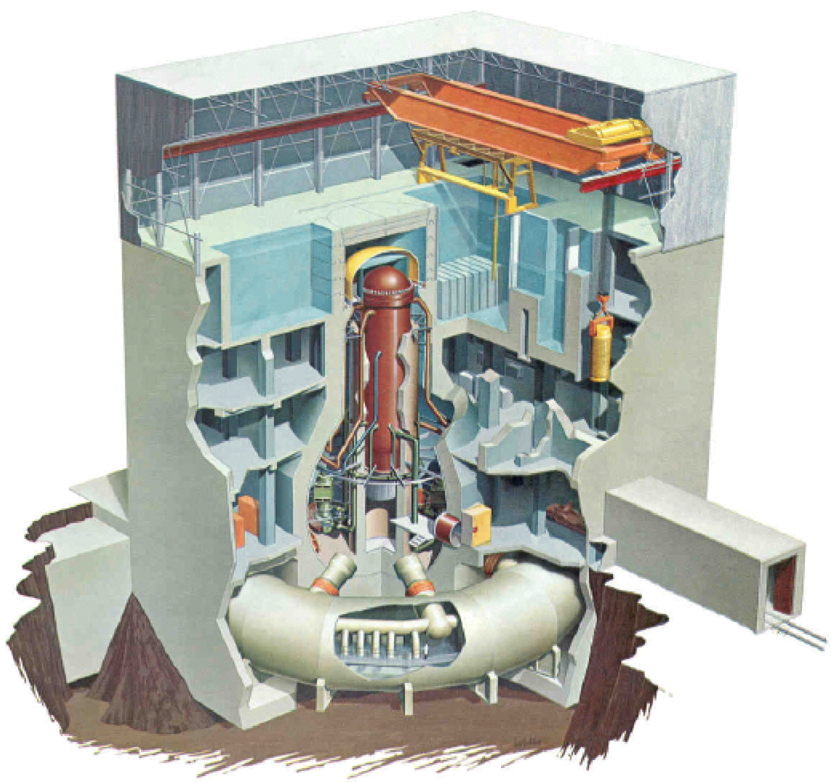

Figure 3 Schematic of a GE BWR Mark I Containment

Source: Reactor Concepts Manual: Boiling Water Reactor (BWR) Systems, p. 3–16 (Ref.A.1).

A.1.2 Pressurized-Water Reactors and Boiling-Water Reactors with Mark III Containments

Figure 4 shows the location of the SFP for the newer BWR Mark III design, which call for a ground-level storage pool to reduce seismic loads. The fuel building is located adjacent to the reactor building and is accessible for fuel servicing during plant operation. A lined fuel pool is used for the storage and servicing of spent fuel and the preparation of new fuel for insertion into the reactor. An area of the pool, separated by gates, is used for transfer of fuel to the reactor servicing pools located in the reactor building, and the receiving of spent fuel discharged from the reactor using a transfer tube. Another area of the fuel storage pool, also separated by gates, is used for the loading and decontamination of equipment and its containers for offsite shipping. Some of these SFPs are located below grade.

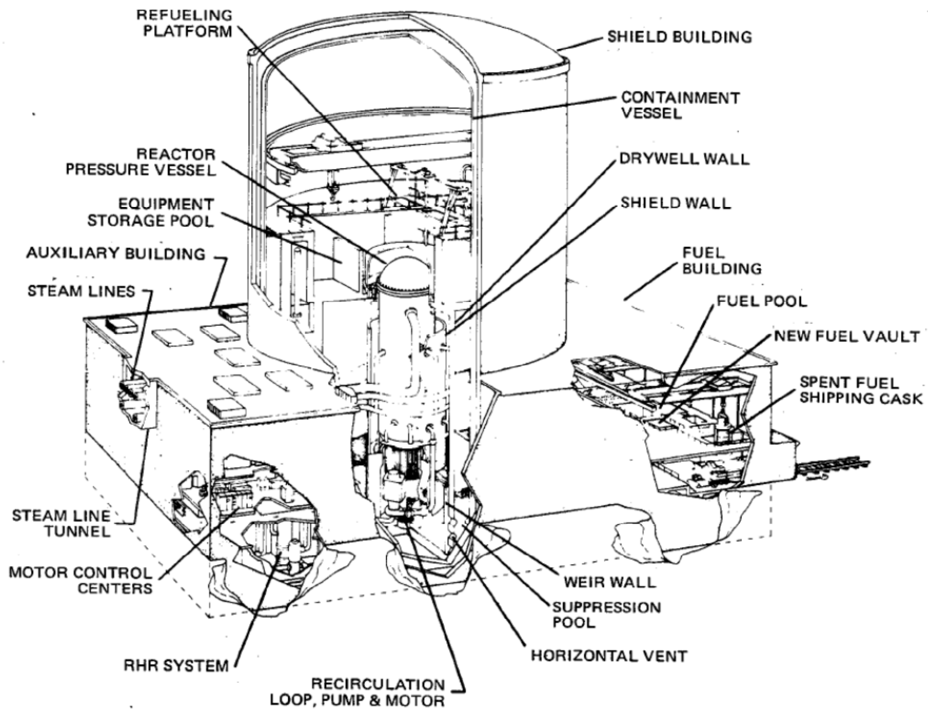

Figure 4 Schematic of a BWR Mark III reactor layout

Source: BWR/6 General Description of a Boiling Water Reactor, Figure 7-1 (Ref. A.2).

Pressurized-water reactor (PWR) designs have SFPs that are located close to grade level within the auxiliary building as shown in Figure 5. This design is typical of the fuel pool arrangement for PWRs.[20]

[20] The Shearon Harris spent fuel pools contain fuel from the Brunswick and Robinson reactors, but the BWR fuel is segregated from the PWR fuel and all transferred fuel has decayed for more than 10 years. The PWR pool reasonably represents this pool because the PWR fuel storage capacity is similar, the power and quantity of each representative refueling batch bounds the Harris conditions, and the stored BWR fuel is segregated such that it would not increase the severity of any potential release.

Nuclear power plant sites that contain two PWR reactors are usually arranged in a mirror image fashion, with the two SFPs (or a shared pool) located in a common area adjoining both reactor buildings or contained within the seismic Category I auxiliary building around or adjacent to the containment building. For single plant or two-plant arrangements, the building covering the SFP and crane structures is typically an ordinary steel industrial building.

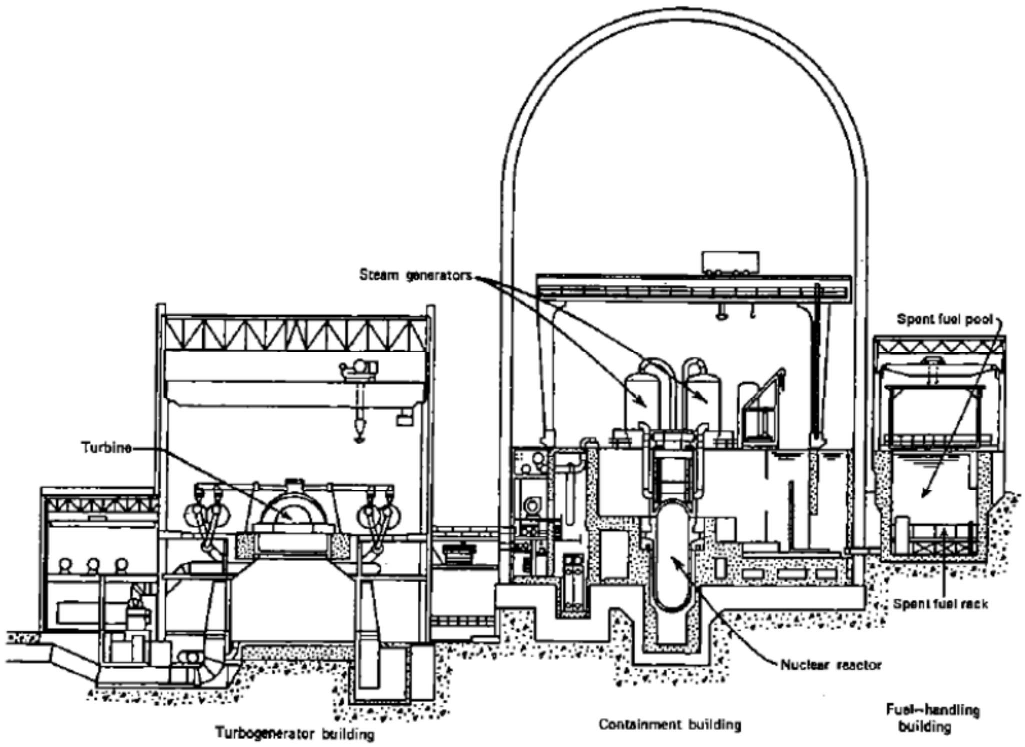

Figure 5 Schematic of a PWR layout

Source: Duderstadt and Hamilton, Figure 3-4 (Ref. A.3).

A.1.3 New Reactors

For the new reactors, the spent fuel storage facility is located within the seismic Category I auxiliary building fuel handling area. The walls of the SFP are an integral part of the seismic Category I auxiliary building structure as shown in Figure 6. The facility is protected from the effects of natural phenomena, such as earthquakes, wind and tornados, floods, and external missiles.

Figure 6 Schematic of an AP1000 reactor layout

Source: Nuclear Street (Ref. A.4).

A.1.4 Spent Fuel Pools at Non-Operating Plants

A SFP at non-operating plants is a special situation in which the reactor unit is no longer operating and spent fuel is stored in the unit's SFP for safe storage until it is placed in an ISFSI or shipped to a long-term Federal repository.

This grouping of pools was not evaluated due to its much lower potential for runaway zirconium oxidation. No further analysis is performed in this analysis for this grouping.

A.1.5 Decommissioned Plant Spent Fuel

A decommissioned plant spent fuel is a special situation in which the licensee requested a license for an independent spent fuel storage installation (ISFSI) to store the reactor unit's spent fuel. The spent fuel was relocated from wet storage in a SFP to dry storage containers at the ISFSI. The spent fuel will be held at the ISFSI until the U.S. Department of Energy is prepared to take possession of the spent fuel and transport it to a long-term repository.

This grouping also includes the GE–Hitachi Morris ISFSI, which is a wet pool storage design and is the only wet ‘‘away from reactor’’ ISFSI of its kind in the U.S. The major components of the Morris ISFSI include the stainless steel lined concrete storage basins, the pool structure, the spent fuel storage grid structure and fuel storage baskets that can store BWR spent fuel assemblies or PWR spent fuel assemblies, ancillary equipment necessary for the movement of spent nuclear fuel, e.g., cranes and basket grappling devices, and equipment necessary for the maintenance of the pool water quality and level (Ref. A.5). Because of the length of time that the discharged spent fuel stored at the Morris ISFSI has cooled, the licensee estimates that, based on evaporation rates, it will take approximately 140 days for the water level to expose the top of the stored fuel bundles (Ref. A.6). Furthermore, there is not sufficient energy in the stored fuel assemblies to ignite the fuel from either a partial or total loss of water.

Based on the characteristics of the spent fuel storage in this grouping, no further analysis is performed in this analysis for this grouping.

A.2 Spent Fuel Storage Options

The technologies available for spent fuel storage fall broadly into two categories—wet and dry—distinguished according to the cooling medium used. The wet option has historically been used for temporary storage in anticipation of the next step in the fuel cycle. More recently, a variety of dry storage options have been developed and applied in the U.S. and international markets.

A.2.1 Wet Storage

The majority of U.S. nuclear power plant spent fuel is stored in water pool storage (i.e., SFPs). SFPs have been used for storage of spent fuel as an established practice since the early days of nuclear power, due among other things, to the excellent properties of water for heat removal and shielding. The majority of reactor SFPs has been re-racked once, and some several times, to increase in-pool storage capacity. These pools are designed to the following principles as discussed in NUREG-0800, "Standard Review Plan," Section 9.1.2, "New and Spent Fuel Storage," (Ref. A.7):

- the capability to withstand and protect against natural phenomena (e.g., safe shutdown earthquake, design-basis tornado)
- the effectiveness of natural circulation of water through the spent fuel storage racks
- the ability to retain water and minimize leakage, which should be detectable, collectable, and quantifiable
- the configuration of the new fuel vault, the spent fuel storage pool, and their handling areas to preclude accidental falls of heavy objects on the new and spent fuel
- the ability to provide both radiological shielding for personnel by maintaining adequate water levels in the SFP
- the use of design features to maintain an adequate water inventory in the SFP under accident condition (e.g., weirs and gates, absence of unnecessary drains, and proper piping penetration levels)
- the use of appropriate monitoring systems to detect SFP water levels, pool temperature, building radiation levels, and to ensure an adequate degree of subcriticality

While there are many common features between SFPs, there are design differences.

A.2.1.1 Location

A.2.1.1.1 At-reactor pool located above grade

For boiling water reactor (BWR) Mark I and II designs, the SFP structures are located in the reactor building at an elevation several stories above grade.

A.2.1.1.2 At-reactor pool located near or below grade

The SFPs at pressurized water reactors (PWRs) and BWR Mark III operating reactors in the U.S. are located with the bottom of the pool at or below plant grade level. Because of the lower elevation, the seismic response is relatively low in comparison to the elevated pools in the BWR Mark I and Mark II plants. Some pools are located below grade, often in bedrock, such that even if a hole in the pool formed, it cannot rapidly drain this pool.

A.2.1.1.3 Away-from reactor or non-operating reactor pool

Away-from-reactor pools are used to provide interim spent fuel storage. Typically, they are divided into pools at the reactor site and pools away from the reactor site or offsite although this distinction is not important to this analysis. True away-from-reactor pools are independent of the reactor and all its services and can continue to operate after the reactor has been finally shut down and decommissioned. There are pools, however, that are located at reactors that are shut down but rely extensively on reactor services such as cooling water and water treatment, ventilation and electrical supplies. When reactors are shut down, special arrangements are usually taken because it could be impractical or uneconomic to continue to operate costly reactor-derived services if the spent fuel must remain in storage onsite for long periods. Dry storage facilities generally remove decay heat by passive cooling and have lower operating costs.

A.2.1.2 Functional Configuration

A.2.1.2.1 Dedicated pool

This is the simplest layout adopted for nuclear power plants in which a SFP supports a single nuclear power plant unit.

A.2.1.2.2 Shared pool

There are cases in which nuclear power plant units may be connected by water gates to share a SFP.

A.2.2 Dry Storage

Numerous companies supply dry storage technologies to U.S. commercial nuclear power plants, as shown in Table 70 located in Appendix F to this document. These dry storage cask systems[21] (DSCs) are certified by the NRC for storage of high burnup spent fuel (i.e., burnups greater than 45 GWd/MTU), using both regional and uniform loading of spent fuel in the packages. Although the dry storage design differs in design details, capacity, and loading steps, the scope of this analysis is limited to generic dry storage technologies, in order to develop a context for the cost-benefit analysis described in subsequent sections of this document.

A.3 Rack Designs

[21] The term dry storage cask system (DSC) includes dual-purpose canister based systems, dual-purpose casks, and storage-only dry storage casks and canister systems.

The design of storage racks and fuel element holder configurations varies considerably from facility to facility, both in general appearance and in details. In March 1979, the NRC issued NUREG/CR-0649, "Spent Fuel Heatup following Loss of Water during Storage" (Ref. A.8), which provided an analysis of spent fuel heatup following a hypothetical accident involving drainage of the storage pool. The report included analysis to assess the effect of decay time, fuel element design, storage rack design, packing density, room ventilation, drainage level, and other variables on the heatup characteristics of spent fuel stored in a SFP to predict the conditions under which clad failure would occur. The report concluded that the likelihood of clad failure caused by rupture or melting following a complete drainage is extremely dependent on the storage configuration and the spent fuel decay period. Furthermore, the minimum prerequisite decay time to preclude clad failures may vary from less than 10 days for some storage configurations to several years for others. The potential for reducing this critical decay time either by making reasonable design modifications or by providing effective emergency countermeasures was found to be significant. The NUREG/CR-0649 analysis assumed in most cases that a 41-centimeter (16-inch) open space is maintained between the baseplate and the bottom of the pool and between the sidewalls and the outermost basket or holder. The rack designs evaluated had center-to-center fuel element spacing that ranged from 21.6 centimeters (8.5 inches) to 53 centimeters (21 inches).

NUREG-1353, "Regulatory Analysis for the Resolution of Generic Issue 82, Beyond-Design-Basis Accidents in Spent Fuel Pools," which draws from the preceding report, concludes that if the decay heat level is high enough to heat the fuel rod cladding to about 900 degrees Celsius (C), the oxidation becomes self-sustaining, resulting in a Zircaloy cladding fire. NUREG-1353 used a conservative and bounding conditional probability of a Zircaloy cladding fire given a complete loss of water. The conservative and bounding values used were 1.0 for PWRs and 0.25 for BWRs in high-density configurations based on differences in assumed rack geometry.

NUREG/CR-6441, "Analysis of Spent Fuel Heatup following Loss of Water in a Spent Fuel Pool: A Users' Manual for the Computer Code SHARP" (Ref. A.9), was issued in 2002. This report included an analysis of spent fuel heatup, using representative design parameters and fuel loading assumptions. Sensitivity calculations were also performed in this NUREG to study the effect of fuel burnup, building ventilation rate, baseplate hole size, partial filling of the racks, and the amount of available space to the edge of the pool. The spent fuel heatup was found to be strongly affected by the total decay heat production in the pool, the availability of open spaces for airflow, and the building ventilation rate. SFP analyses performed by the NRC after this time do not use the SHARP computer code. Rather, the NRC uses the MELCOR computer code (owing to its mechanistic treatment of severe accident phenomena), with supporting analysis using the COBRA-SFS, FLOW3D, and Fluent codes, along with confirmatory experiments at Sandia National Laboratories.

The SFPS (draft) evaluated a BWR reference plant rack geometry with a cell pitch of 16 centimeters (6.3 inches); a closed rack design that inhibited or prevented cross-flow, while being relatively open at the top and bottom for axial flow; and a distance between the pool floor liner and the bottom of the rack baseplate of approximately 26 centimeters (10.2 inches).

A.4 REFERENCES

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APPENDIX B: SPENT FUEL STORAGE STRATEGIES

B.1 Interim Storage Options to Expand Onsite Storage

The delay in the construction of the geologic repository mandated by Congress has caused nuclear power plants to store used fuel on site for longer than originally intended. The result is that many nuclear plants are running out of existing storage capacity. When a plant's used fuel pool nears its designed capacity, a company has two options:

- **Re-racking**. The first choice is to re-rack the used SFP, moving the fuel assemblies closer together. Eventually, even re-racked pools reach their capacity.

- **Dry Containers**. Many U.S. nuclear power plants are storing used spent fuel in large, rugged containers made of steel or steel-reinforced concrete. Depending on the design, a container can hold up to 37 PWR fuel assemblies or 87 BWR fuel assemblies. The containers have a 20-year license. After 20 years, with NRC approval, the license could be extended for up to 40 years.

Building a dry storage facility at a plant site requires an initial investment of approximately $10 million to $20 million. Once the facility is operational, it may cost $5 million to $7 million a year for the maintenance and security of the facility and for adding more containers as storage needs grow (Ref. B.1).

While re-racking is the most used method for expanding at-reactor spent fuel storage capacity over the past 40 years, utility experience with dry storage applications has grown significantly. In addition to the implementation and continued operation of dry storage at operating plant sites, numerous nuclear power plants that have permanently ceased operation have offloaded spent fuel from storage pools to at-reactor ISFSIs to facilitate decommissioning of the SFPs.

B.2 Cask Loading Strategies

Two cask loading strategies used to manage cask loading are 1) full core reserve (FCR) margin, and 2) SFP inventories. The first strategy is just-in-time cask loading, in which casks are loaded with a goal of maintaining FCR in the SFP. The second type of cask loading strategy employs larger loading campaigns with a goal of achieving additional space above that required for FCR in order to space cask loading campaigns further apart. When implementing this cask loading strategy, a plant might load 10 to 12 casks following every other refueling rather than five to six casks following every refueling outage.

The benefits of just-in-time cask loading are that:

- It minimizes near-term capital and operating expenditures since only enough casks to maintain FCR are loaded.

- Cask loading crews also do not have long periods of time between cask loading campaigns and may result in shorter learning curves for the next cask loading campaign.

The risks associated with a just-in-time loading strategy include:

- unexpected maintenance that requires offloading the reactor core at a time when the SFP has less than one FCR

- unexpected delays in delivery of storage casks caused by licensing issues or fabrication delays that might affect FCR capability

- increased outage times because of space limitations in the SFP

Benefits associated with larger loading campaigns include:

- There are fewer cask loading campaigns over the life of the plant (although the same number of casks would be loaded over the life of the plant) resulting in cost savings associated with mobilization/demobilization for cask loading, training, and dry runs.

- If a company owns multiple sites with operating ISFSIs and cask loading equipment is shared between sites, this results in fewer shipments of cask handling equipment between sites and possible cost savings.

- Larger loading campaigns would also provide more margin in SFPs over FCR, such that unexpected maintenance requiring off-loading of the reactor core can be accomplished and unexpected delays in delivery of storage casks are more likely to be accommodated.

- A negative benefit is that costs associated with large loading campaigns include increases in near-term capital and operating budgets because of purchasing and loading casks sooner than in a just-in-time loading scenario.

Risks associated with larger loading campaigns include:

- Longer cask loading cycles (months rather than weeks) to complete a loading campaign and possible impacts on plant maintenance activities or other SFP activities.

- Impacts on workers involved in cask loading operations. Shutdown nuclear operating plants have loaded between 15 and 60 casks in extended campaigns with reasonable schedules.

B.3 References

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APPENDIX C: ANALYSIS MODEL INFORMATION

C.1 <u>Economic Modeling and Representative Plant Assumptions</u>

C.1.1 Compliance with Existing NRC Requirements

The regulatory baseline assumes full compliance with existing NRC requirements, including current regulations and relevant orders. This is consistent with NUREG/BR-0058, "Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission," Rev. 4 (Ref. C.1), which states that "in evaluating a new requirement…, the staff should assume that all existing NRC and Agreement State requirements have been implemented." For the purpose of evaluating the potential benefits of expedited transfer of spent fuel to dry cask storage, this analysis used a conservative approach by crediting successful application of post-9/11 and post-Fukushima mitigation capabilities for the low-density SFP alternative and assumed no successful mitigation for the high-density SFP storage regulatory baseline.

The data and assumptions used in analyzing the quantifiable impacts associated with each proposed alternative are discussed in this section. Information on attributes affected by the proposed regulatory framework alternatives is obtained from experienced NRC staff and other sources as referenced. The NRC considers the potential differences between the new requirements and the current requirements and incorporates the proposed incremental changes into this cost-benefit analysis.

C.1.2 Base Year

All monetized costs are expressed in 2012 dollars. Ongoing costs of operation related to the alternatives are assumed to begin in 2014 unless otherwise stated, and are modeled on an annual cost basis.

Estimates are made for one-time implementation costs. The NRC assumes that these costs will be incurred in the first year of the analysis unless otherwise noted.

Estimates are made for recurring annual operating expenses. The values for annual operating expenses are modeled as a constant expense for each year of the analysis horizon. An annuity calculation was performed to discount these annual expenses to 2012 dollar values.

C.1.3 Discount Rates

In accordance with guidance from the Office of Management and Budget (OMB) Circular No. A-4 (Ref. C.2) and NUREG/BR-0058, Revision 4 (Ref. C.1), present-worth calculations are used to determine how much society would need to invest today to ensure that the designated dollar amount is available in a given year in the future. By using present-worth, costs and benefits, regardless of when the cost or benefit is incurred in time, are valued to a reference year for comparison. The choice of a discount rate, and its associated conceptual basis, is a topic of ongoing discussion within the Federal government. Based on OMB Circular No. A-4, present-worth calculations are presented using 3 percent and 7 percent real discount rates. A 3 percent discount rate approximates the real rate of return on long-term government debt, which serves as a proxy for the real rate of return on savings to reflect reliance on a social rate of time preference discounting concept. A 7 percent rate approximates the marginal pretax real rate of return on an average investment in the private sector, and is the appropriate discount rate whenever the main effect of a regulation is to displace or alter the use of capital in the

private sector. A 7 percent rate is consistent with an opportunity cost of capital concept to reflect the time value of resources directed to meet regulatory requirements.

C.1.4 Cost/Benefit Inflators

The consequences for some attributes are estimated based on the values published in the NRC Regulatory Analysis Handbook. Within the NRC Regulatory Analysis Handbook, the information in relation to severe reactor accident consequences is provided in previous year dollars.
To evaluate the costs and benefits consistently, the consequences are inflated. The most common inflator is the Consumer Price Index for all urban consumers (CPI-U), developed by the U.S. Department of Labor, Bureau of Labor Statistics. Using the CPI-U, the previous year dollars were converted to the year 2012. The formula to determine the amount in 2012 dollars is

$$\frac{\mathrm{CPIU}_{2012}}{\mathrm{CPIU}_{\mathrm{Base\ Year}}} * \mathrm{Consequence}_{\mathrm{Base\ Year}} = \mathrm{Consequence}_{2012}$$

Values of CPI-U used in this cost-benefit analysis are summarized in Table 31.

Table 31 Consumer Price Index—All Urban Consumers Inflator

Base Year	CPI-U Inflator for Year 2012
2005	1.1756
2006	1.1389
2007	1.1073
2008	1.0664
2009	1.0702
2010	1.0529
2011	1.0207

Source: U.S. Department of Labor, Bureau of Labor Statistics, “Databases, Tables & Calculators by Subject: CPI Inflation Calculator (Ref. C.3).

C.1.5 Description of Representative Plants

Representative BWR Mark I and Mark II (Group 1)

The representative Group 1 plant is a single unit boiling-water reactor (BWR) Mark I or Mark II reactor with a rated capacity of approximately 3,500 megawatts thermal (MW_t) and a unit dedicated SFP. The representative BWR reactor began operating in the 1970s and will reach the end of its renewed operating license by year 2037. The NRC assumes the reactor core contains 764 assemblies and the SFP has a capacity of approximately 3,055 assemblies in a high-density 1x4 loading configuration. This number is based on a pool capacity of 3,819 assemblies, reduced by 764 assemblies to accommodate a full core offload capability using the existing high-density racking. In a low-density configuration, the SFP stores 852 assemblies in which the newly discharged spent fuel is arranged in a 1x4 configuration and the remaining fuel assemblies arranged in a checkerboard pattern. The unit operates on 24-month cycles, discharging approximately 284 assemblies per cycle. The representative BWR has already implemented dry storage.

Representative PWR or BWR Mark III (Group 2)

The representative Group 2 plant is a single unit pressurized-water reactor (PWR) with a rated capacity of approximately 3,400 MW_t and a unit dedicated SFP. The representative Group 2 reactor began operating in the 1970s and will reach the end of its extended operating license by year 2040. The NRC assumes the reactor core contains 193 assemblies and the SFP has a capacity of approximately 1,220 assemblies in a high-density 1x4 loading configuration. This number is based on a pool capacity of 1,414 assemblies, reduced by 193 assemblies to accommodate a full core offload capability using the existing high-density racking. In a low-density 1x4 with empties configuration, the SFP stores 312 assemblies. The unit operates on 18-month cycles, discharging approximately 78–84 assemblies per cycle. The representative PWR has already implemented dry storage.

Representative New Nuclear Plant (Group 3)

The representative new plant is an AP1000 PWR with a rated capacity of approximately 3,400 MW_t and a unit dedicated SFP. The representative Group 3 reactor begins operating in the year 2018 and will reach the end of its extended operating license by year 2078. The NRC assumes the reactor core contains 157 assemblies and the SFP has a capacity of approximately 1,000 assemblies in a high-density 1x4 loading configuration. This number is based on a pool capacity of 1,160 assemblies, reduced by 157 assemblies to accommodate a full core offload capability using the existing high-density racking. In a low-density 1x4 with empties configuration, the SFP stores 340 assemblies. The unit operates on either 18-month or 24-month cycles, discharging an estimated 69 assemblies per 18-month cycle or 77 assemblies per 24-month cycle (Ref. C.4, Section 9.1). The representative new nuclear plant is expected to begin dry storage in 2038 if high-density pool storage is allowed and will load a sufficient number of casks to maintain its full core offload capability.

Representative SFP Shared Between Units (Group 4)

This representative SFP is shared between two PWR units, each with a rated capacity of approximately 3,400 MW_t. The SFP, designed in two halves, is located outside the containment in the Auxiliary Building and provides underwater storage of spent fuel assemblies after their

removal from the reactor vessel of either reactor unit. The associated Group 4 reactor unit began operating in the 1970s and will reach the end of its extended operating license by year 2038. The NRC assumes each reactor core contains 193 assemblies and the SFP has a capacity of 1637 assemblies in a high-density 1x4 loading configuration. This number is based on a pool capacity of 1,830 assemblies, reduced by 193 assemblies to accommodate one unit's full core offload capability using the existing high-density racking. In a low-density 1x4 with empties configuration, the SFP stores 468 assemblies. The units operate on 24-month cycles, discharging approximately 78–84 assemblies per cycle on a 1-year staggered cycle. The representative shared SFP has already implemented dry storage.

C.1.6 Projected Number of Outages and Spent Fuel Assemblies

The spent fuel assembly inventory at a SFP is plant specific based on initial inventory, projected spent fuel discharged during each refueling outage, and operating cycle length. Additional spent fuel storage requirements are calculated using the SFP capacity and the cumulative spent fuel discharges. The cumulative number of fuel assemblies discharged is subtracted from the spent fuel pool capacity, assuming that each spent fuel pool retains space in the SFP to discharge one full core of fuel. During years in which no spent fuel is discharged at plants operating on 18-month or 24-month operating cycles, there would be no change in the SFP inventory. If there are more assemblies requiring storage than there is space in the SFP (including space to discharge one full core of fuel), these additional storage needs are assumed to be met using at-reactor dry storage rather than expansion of SFP capacity. The number of spent fuel assemblies required up to operating license expiration is calculated for each group based on the existing high-density SFP inventory, the number added from refueling outages, and the full reactor core inventory. These results are provided in Table 32.

Table 32 Number of Spent Fuel Assemblies Remaining through Operating License Expiration

Group No.	Category	Inventory	Number of Inventories	No. of spent fuel assemblies	Total
1	Current SFP inventory	3,055	1	3,055	
	refueling	284	12	3,408	
	reactor core	764	1	764	7,227
2	Current SFP inventory	1,220	1	1,220	
	refueling	78	18	1,404	
	reactor core	193	1	193	2,817
3a	Current SFP inventory	0	1	0	
	Refueling (18-month cycle)	69	40	2,760	
	reactor core	157	1	157	2,917
3b	Current SFP inventory	0	1	0	
	Refueling (24-month cycle)	77	30	2,310	
	reactor core	157	1	157	2,467
4	Current SFP inventory	1,637	1	1,637	
	refueling	78	24	1,872	
	reactor core	193	2	386	3,895

C.1.7 Dry Storage Capacity

Three companies supply most of the dry storage technologies to U.S. commercial nuclear power plants. These companies are Holtec International, Inc. (Holtec), NAC International, Inc. (NAC), and Transnuclear, Inc. (Transnuclear). The dry storage cask systems[22] (DSCs) for all three companies are certified by the NRC for storage of high burnup spent fuel (i.e., burnups greater than 45 GWd/MTU), using both regional and uniform loading of spent fuel in the packages. A summary of a representative sampling of dry storage canisters commercially available for spent fuel storage is provided in Table 33.

Table 33 Representative Sampling of Commercially Available BWR Spent Fuel Dry Storage Technology

Vendor Package	Fuel Type	Canister Type	Capacity (Assemblies)	Maximum Decay Heat Per Package[1] (kW)
Holtec HI-STORM 100	PWR	MPC-24	24	34
	PWR	MPC-32	32	34
Holtec HI-STORM FW	PWR	MPC-37	37	47
NAC UMS	PWR	24P	24	23
NAC MAGNASTOR	PWR	37P	37	35.5
Transnuclear NUHOMS	PWR	24PTH	24	40.8
	PWR	32PTH1	32	40.8
Transnuclear TN-40HT	PWR	Bolted	40	32
Holtec HI-STORM	BWR	MPC-68	68	34
Holtec HI-STORM FW	BWR	MPC-89	89	46.36
NAC MAGNASTOR	BWR	87B	87	33
Transnuclear NUHOMS	BWR	61BTH	61	31.2
Transnuclear TN-68	BWR	Bolted	68	30

The maximum decay heat per assembly for uniform loading is estimated by dividing the package decay heat by the number of assemblies. The maximum decay heat per assembly under regional loading schemes will generally be higher than the maximum decay heat per assembly assuming uniform loading for a smaller number of assemblies. Cask certificates of compliance provide the specific maximum assembly decay heat limits for each storage location in the basket.
Source: EPRI TR-1025206, p. 2-11 (Ref. C.5).

C.1.8 Discharged Spent Fuel Assemblies

The number of spent fuel assemblies in units of metric tons of uranium (MTU) that is discharged by a reactor unit during each refueling outage is estimated based on the unit's licensed thermal rating (megawatts thermal, MW_t, discharge burnup (BUP in MWd/MTU), capacity factor (CF in percent), and operating cycle length (CYL in years) as shown below.

$$MTU = \frac{MW_t \; x \; CYL \; x \; \frac{CF}{100} \; x \; \frac{365 \; days}{year}}{BUP}$$

Using the above formula, a 3,514 MW_t BWR reactor with a 24-month operating cycle operating at a 90 percent capacity factor and an average spent fuel assembly burnup of 45,000 MWd/MTU would discharge 51.3 MTU during each refueling cycle. The number of discharged

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22 The term dry storage cask system (DSC) includes dual-purpose canister based systems, dual-purpose casks, and storage-only dry storage casks and canister systems.

assemblies (ASSY) is estimated by dividing the MTU discharge value by the fuel assembly unit weight. Based on an average BWR fuel assembly unit weight of 0.18 MTU per assembly the equation yields approximately 285 assemblies.

C.1.9 Spent Fuel Assembly Decay Heat as a Function of Burnup and Cooling Time

As fuel assembly burnups increase, the decay heat of the fuel assembly (watts per assembly) and the Cesium-137 inventory in the spent fuel increase. Decay heat also can vary significantly with initial enrichment and assembly irradiation parameters. Spent fuel burnups have gradually increased since the 1990s with average BWR burnups about 43 GWd/MTU and range between 40 and 53 GWd/MTU and with average PWR burnups range between 40 and 55 GWd/MTU.

As shown in Figure 7, average decay heat for a 40 GWd/MTU PWR spent fuel assembly that has cooled for 5 years is approximately 1,100 watts per assembly based on approximately 2.3 kW/MTU times 0.45 MTU per assembly and a cesium-137 inventory of approximately 6.8×10^4 Ci per assembly. The average decay heat for a 55 GWd/MTU assembly that has cooled for five years is approximately 1,500 watts per assembly with a cesium-137 inventory of 9.6×10^4 Ci per assembly (Ref. C.5, p. 2-6). In comparison, a 40 GWd/MTU PWR spent fuel assembly that has cooled for 10 year has a decay heat of approximately 700 watts per assembly and a 55 GWd/MTU PWR spent fuel assembly has a decay heat of approximately 1,000 watts per assembly.

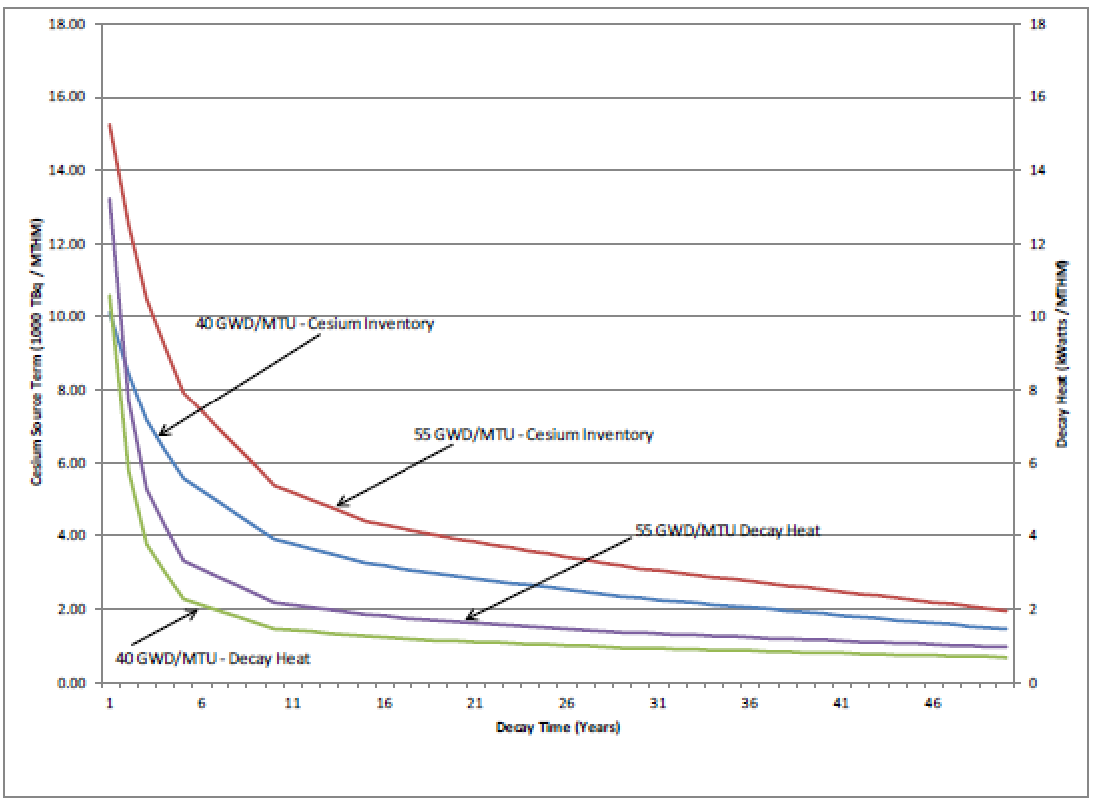

Figure 7 PWR spent fuel assembly decay heat and cesium inventory as a function of burnup and cooling time

Source: EPRI TR-1025206, p. 2-6 (Ref.C.5).

Average decay heat for a 40 GWd/MTU BWR spent fuel assembly that has cooled for five years is approximately 360 watts/assembly based on approximately 2.0 kW/MTU (from Figure 8) times 0.18 MTU per BWR assembly and a cesium-137 inventory of approximately 3.0×10^4 curies per assembly. The average decay heat for a 50 GWd/MTU assembly that has cooled for 5 years is approximately 520 watts per assembly with a cesium-137 inventory of 3.4×10^4 curies per assembly (Ref. C.5, p. 2-8). In comparison, a 40 GWd/MTU BWR spent fuel assembly that has cooled for 10 years has a decay heat of approximately 250 watts per assembly and a 50 GWd/MTU BWR spent fuel assembly has a decay heat of approximately 350 watts per assembly.

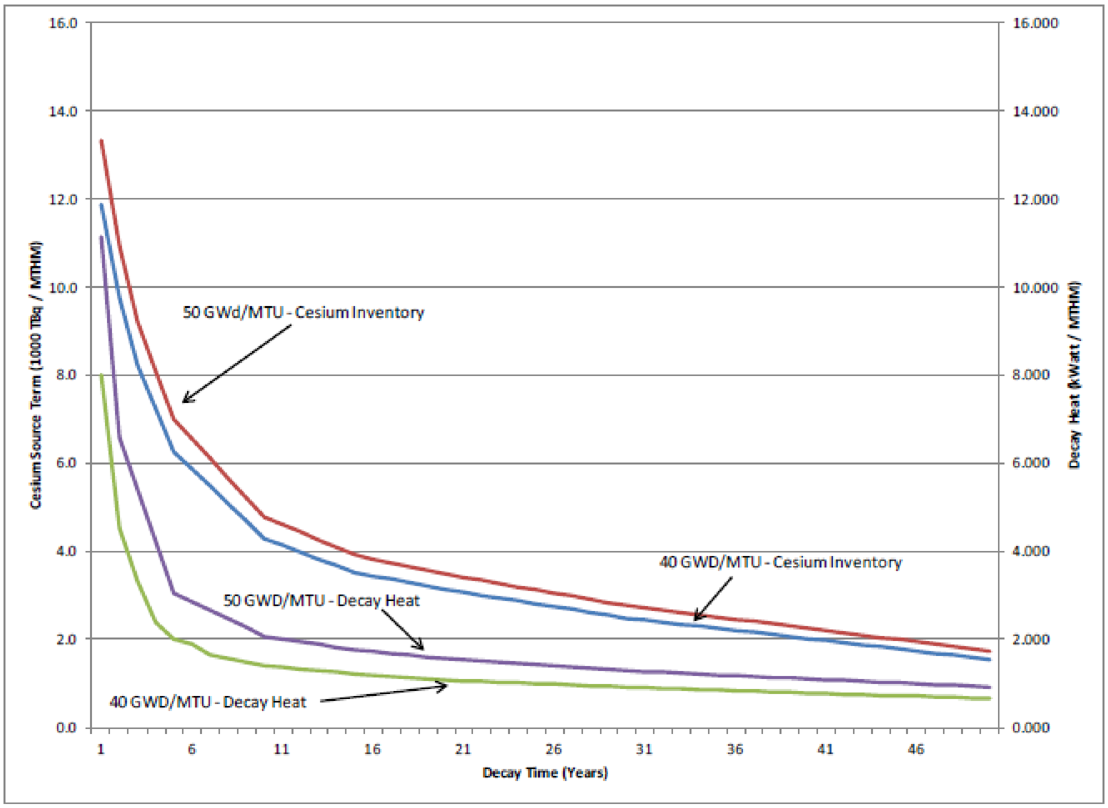

Figure 8 BWR spent fuel assembly decay heat and cesium inventory as a function of burnup and cooling time

Source: EPRI TR-1025206, p. 2-9 (Ref. C.5).

Based on an average PWR spent fuel assembly that emits 1,100 watts or an average BWR spent fuel assembly that emits 360 watts, Table 34 shows the number of spent fuel assemblies that could be stored assuming uniform fuel assembly burnup of 40 GWd/MTU and a five year decay time. Table cells that are not shaded identify those dry storage canisters that can be filled to capacity without exceeding the maximum decay heat per package rating, subject to restrictions on loading pattern. Shaded table cells identify those casks whose capacity loading is limited by the spent fuel assembly decay heat. For 55 GWd/MTU PWR assemblies that emit approximately 1,500 watts after they have cooled for five years or 50 GWD/MTU BWR assemblies that emit approximately 520 watts, fewer assemblies can be stored in the DSC than its design capacity due to decay heat limitations. The number of additional dry storage casks

that would be required for spent fuel cooled for five years depends on the vendor package selected and ranges between no additional canisters to almost twice as many additional canisters. Additional DSCs, which are required because of high heat load, are estimated in this cost-benefit analysis. For a sensitivity analysis, the maximum capacity based on decay heat limitations was also calculated if the spent fuel was allowed to cool for 10 years. As shown in Table 6, all of the lower heat rate fuel and most of the higher heat rate fuel could be loaded into casks without any decay heat limitations.

For this analysis, the Transnuclear TN-68 dry casks are selected as representative DSCs for the BWR spent fuel for Group 1. For Groups 2, 3, and 4, the Holtec Hi-Storm FW DSC is modeled as representative DSCs for the PWR spent fuel.

Table 34 Canister Storage Capacity Based on Decay Heat Limitations

Vendor Package	Fuel Type	Capacity (Assemblies)	Maximum Decay Heat Per Package (kW)	Maximum capacity based on decay heat			
				5 year cooling		10 year cooling	
				1100w (PWR) 360w (BWR) per assembly	1500w (PWR) 520w (BWR) per assembly	700w (PWR) 250w (BWR) per assembly	1000w (PWR) 350w (BWR) per assembly
Holtec HI-STORM 100	PWR	24	34	24.00	22.67	24.00	24.00
	PWR	32	34	30.91	22.67	32.00	32.00
Holtec HI-STORM FW	PWR	37	47	37.00	31.33	37.00	37.00
NAC UMS	PWR	24	23	20.91	15.33	24.00	23.00
NAC MAGNASTOR	PWR	37	35.5	32.27	23.67	37.00	35.50
Transnuclear NUHOMS	PWR	24	40.8	24.00	24.00	24.00	24.00
	PWR	32	40.8	32.00	27.20	32.00	32.00
Transnuclear TN-40HT	PWR	40	32	29.09	21.33	40.00	32.00
Holtec HI-STORM	BWR	68	34	68.00	65.38	68.00	68.00
Holtec HI-STORM FW	BWR	89	46.36	89.00	89.00	89.00	89.00
NAC MAGNASTOR	BWR	87	33	87.00	63.46	87.00	87.00
Transnuclear NUHOMS	BWR	61	31.2	61.00	60.00	61.00	61.00
Transnuclear TN-68	BWR	68	30	68.00	57.69	68.00	68.00

1. Shaded values identify where cask loading capacity is limited by the spent fuel decay heat.

The currently approved minimum cooling time for fuel stored in dry casks is seven years (10 years for some fuel types). Cask vendors would need to demonstrate, in an amendment request, that spent fuel that was cooled for a shorter period can be stored safely. The costs to prepare such an amendment request and for the NRC review are not included in this analysis. Furthermore, fuel selected must meet cask design specific fuel selection parameters that limit the maximum enrichment, maximum burnup, minimum cooling time, and maximum decay heat. The methodology used to estimate the capacity of the DSCs for spent fuel is subject to uncertainties resulting from decay heat and loading pattern restrictions. As a result, the actual DSC capacity may be higher or lower than those estimated.

C.1.10 Facility Life Cycle

Spent fuel storage involves a series of phases over the life cycle of the nuclear power plant for which it supports. The plant operational phases will have variable time requirements depending on the plant's refueling schedule, the capacity of the SFP, the term of the operating license, and the forecast schedule of removal of spent fuel from the SFP to the ISFSI.

At the expiration of a nuclear power plant's operating license, the full core is offloaded into the SFP. The licensee continues to store spent fuel in the pool following commercial operation[23] to allow the spent fuel to cool sufficiently before placing into dry storage.

C.1.11 Spent Fuel Pool Capacities

SFPs for all reactor types typically range from 9 to 18 meters (30 to 60 feet) in length and 6 to 12 meters (20 to 40 feet) in width, with a spent fuel capacity that ranges from 544 to 4,117 spent fuel assemblies for dedicated SFPs as shown in Table 72 in Appendix F. SFPs that are shared between units have capacities up to 4,628 fuel assemblies. This analysis assumes that plants with SFPs that are shared by multiple units reserve space for only one full core in the SFP.

For new reactors, spent fuel is stored in high-density racks which include integral neutron absorbing material to maintain the required degree of subcriticality. The SFP rack layout contains both Region 1 rack modules and Region 2 rack modules. The racks are designed to store fuel of the maximum design basis enrichment. Each rack in the SFP consists of an array of cells interconnected to each other at several elevations and to a thick base plate at the bottom elevation. These rack modules are free-standing, neither anchored to the pool floor nor braced to the pool wall. For the AP1000 reactors, the spent fuel storage racks include storage locations for 884 fuel assemblies and five defective fuel assemblies.

C.1.12 Spent Fuel Pool Cesium Inventory

The amount of cesium inventory in a SFP varies based on the number of spent fuel assemblies, the type of fuel stored, the discharge burnup, and the amount of time since the fuel was removed from the reactor core. The specific activity, $\frac{A}{M}$, in megacuries per metric tons of uranium (MCi/MTU) is relatively invariant and the assembly mass (in initial MTUs) is a reasonable scaling factor account for variations between different SFPs. This scaling factor is derived as follows assuming the two pools have similar distributions of burnup and cooling periods:

$$\frac{A_1}{M_1} \sim \frac{A_2}{M_2}$$

Where A_x is the absolute activity in megacuries (MCi) of SFP_x and M_x is the total amount of uranium in metric tons (MTU) stored in spent fuel pool x. The total amount of uranium, M_x, is estimated based on the number of spent fuel assemblies, N, and the average fuel assembly unit weight, m in MTU per assembly in the pool. A burnup scaling factor (BUP in MWd/MTU) can also be used in the above equation to yield:

$$\frac{A_1}{N_1 \; x \; BUP_1 \; x \; m_1} = \frac{A_2}{N_2 \; x \; BUP_2 \; x \; m_2}$$

Solving for the SFP absolute activity of the second pool yields:

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23 Decommissioning of the unit must be completed within 60 years of permanent cessation of operations under 10 CFR 50.82, "Termination of License." Completion of decommissioning beyond 60 years will be approved by the Commission only when necessary to protect public health and safety.

$$A_2 = \frac{A_1 \, x \, N_2 \, x \, BUP_2 \, x \, m_2}{N_1 \, x \, BUP_1 \, x \, m_1}$$

Using the above formula, a 3,514 MW_t BWR reactor with a SFP with absolute activity of 60 MCi from the storage of 3,055 BWR fuel assemblies with an average spent fuel assembly burnup of 45,000 MWd/MTU and with an average BWR fuel assembly unit weight of 0.18 MTU per assembly can be equated to a 3,400 MW_t PWR reactor with a SFP with an unknown absolute activity from the storage of 1,220 PWR fuel assemblies with an average spent fuel assembly burnup of 45,000 MWd/MTU and with an average PWR fuel assembly unit weight of 0.46 MTU per assembly as shown below.

$$A_{PWR} = \frac{59 \, x \, 1220 \, x \, 45000 \, x \, 0.46}{3055 \, x \, 45000 \, x \, 0.18} = \frac{1.49 \, x \, 10^9}{2.47 \, x \, 10^7} = 60.2 \, MCi$$

To test the accuracy of this estimate for high-density SFP scaling, the high-density Peach Bottom, Unit 3 SFP cesium inventories from 2001 and 2011 were used. The results showed that there is less than 1 percent error by using the scaling method described above.

Error is introduced when attempting to estimate a pool with a significantly different average cooling period for the spent fuel. To eliminate this source of error, the low-density loaded SFP inventory is estimated based on the low-density SFP characteristics evaluated in the SFPS and using the actual Cs-137 inventory of 22 MCi for all low-density SFPs and the formula above.

Table 72 located in Appendix F provides the estimated Cs-137 inventory for each SFP in a high-density loading configuration using the scaling factor discussed above. Cesium inventories used to analyze each SFP group are summarized in Table 35.

Table 35 Spent Fuel Pool Group Cesium Inventory

SFP Group	Pool Storage Case	Pool Cesium Inventory (MCi)		
		Sensitivity (Low Estimate)	Base Case	Sensitivity (High Estimate)
1	High-density	40.6	52.7	63.3
	Low-density	19.8	22.0	26.4
2	High-density	57.4	67.9	78.2
	Low-density	15.7	17.4	20.9
3	High-density	33.7	44.4	54.2
	Low-density	15.7	17.4	20.9
4	High-density	63.6	101.1	142.2
	Low-density	31.4	34.8	41.8

C.2 Spent fuel Pool Accident Modeling and Evaluation Assumptions

C.2.1 Seismic Hazard Model

This cost-benefit analysis uses the existing U.S. Geological Survey (USGS) 2008 model to evaluate seismic hazards at central and eastern United States (CEUS) nuclear power plants. A new probabilistic seismic hazard model is currently being developed and will consist of two parts: (1) a seismic source zone characterization and (2) a ground motion prediction equation (GMPE) model. Although part (1) is now complete (Ref. C.6), the GMPE update is still in

progress. Furthermore, the NRC is currently developing an independent probabilistic seismic hazard assessment (PSHA) computer code to incorporate part (1) and part (2) when complete. While the USGS (2008) hazard model is not sufficiently detailed for regulatory decisions, it is used for this analysis because it is the most recent and readily available hazard model and was used in the SFPS (Ref. C.7). Although the USGS 2008 model considers western U.S. sites (e.g., Columbia, Diablo Canyon, Palo Verde, and San Onofre), these sites are not addressed in Generic Issue 199 (Ref. C.8), which focused on the CEUS and, therefore, are not included in this analysis. Western sites will be considered on a site-specific basis in response to licensee requested information related to Recommendations 2.1 (Seismic Hazards Evaluations) and 2.3 (Seismic Walkdowns) of the Post-Fukushima Near-Term Task Force.

A comparison of the annual frequency of exceeding a given PGA for BWR Mark I and II sites (see Figure 9) shows that Peach Bottom (i.e., the reference plant) falls close to the upper end of the group located in the CEUS in terms of hazard estimates.

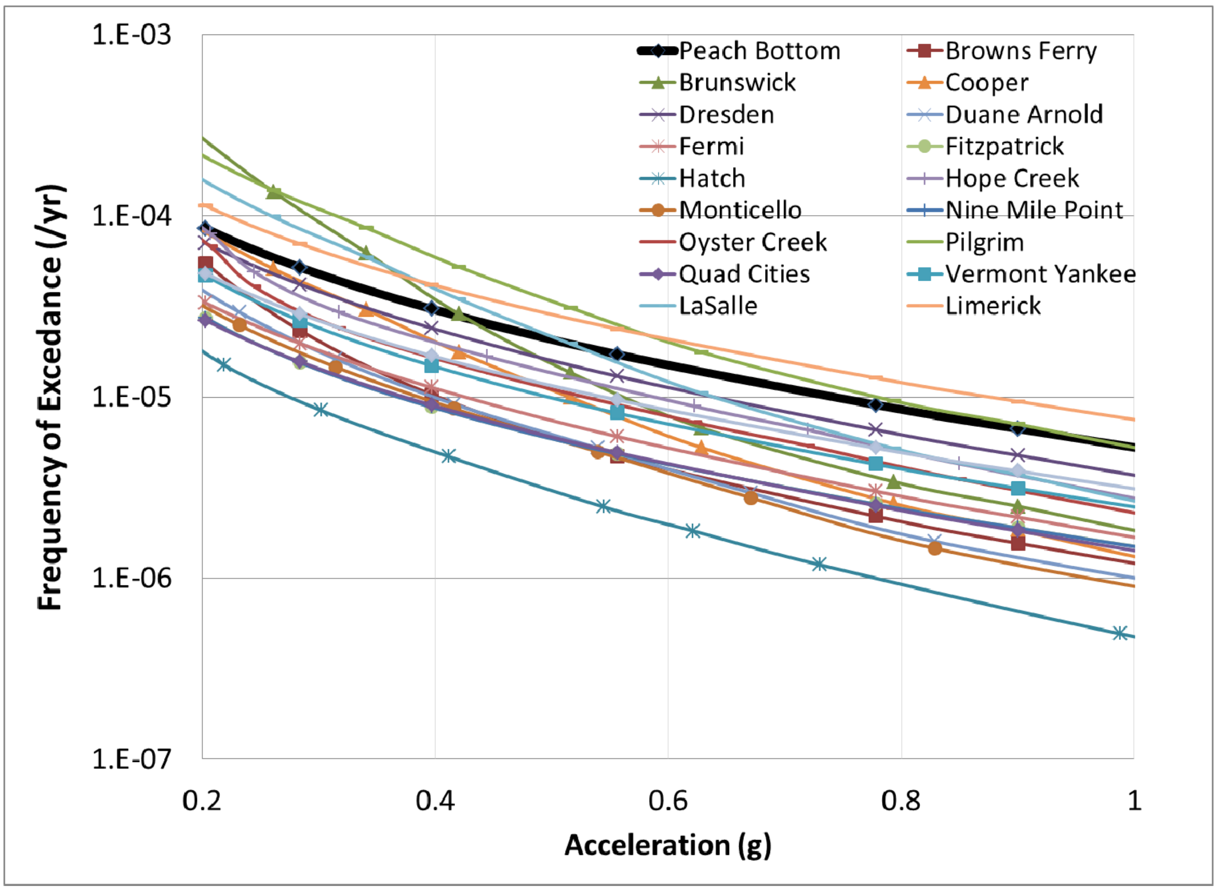


Figure 9 Comparison of annual PGA exceedance frequencies for U.S. BWR Mark I and Mark II reactors (USGS 2008 model)

A similar comparison of the annual frequency of exceeding a given PGA for PWR and BWR Mark III sites (Figure 10), for new reactors (Figure 11), and for reactors units with a shared SFP (Figure 12) shows that Peach Bottom falls close to the upper end of the group in terms of hazard estimates.

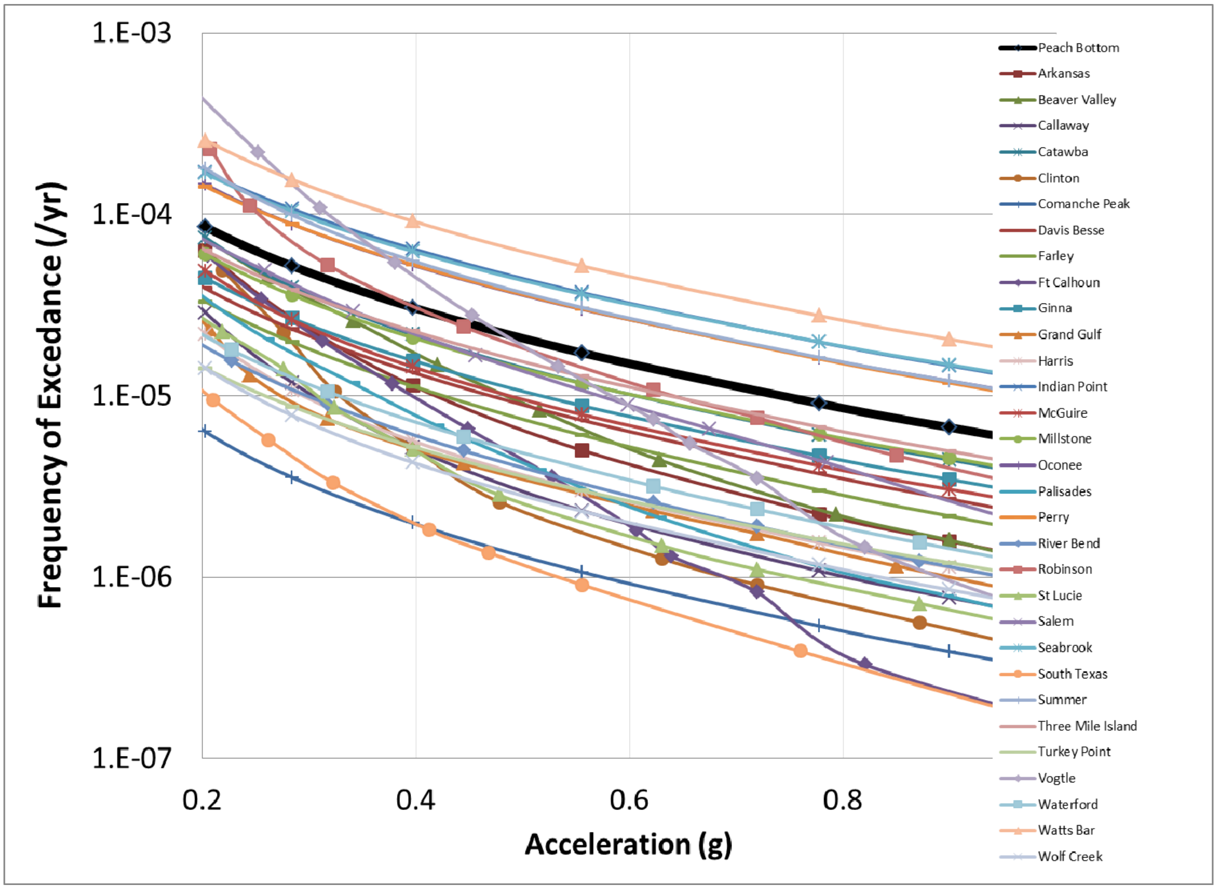


Figure 10 Comparison of annual PGA exceedance frequencies for U.S. PWR and BWR Mark III reactors (USGS 2008 model)

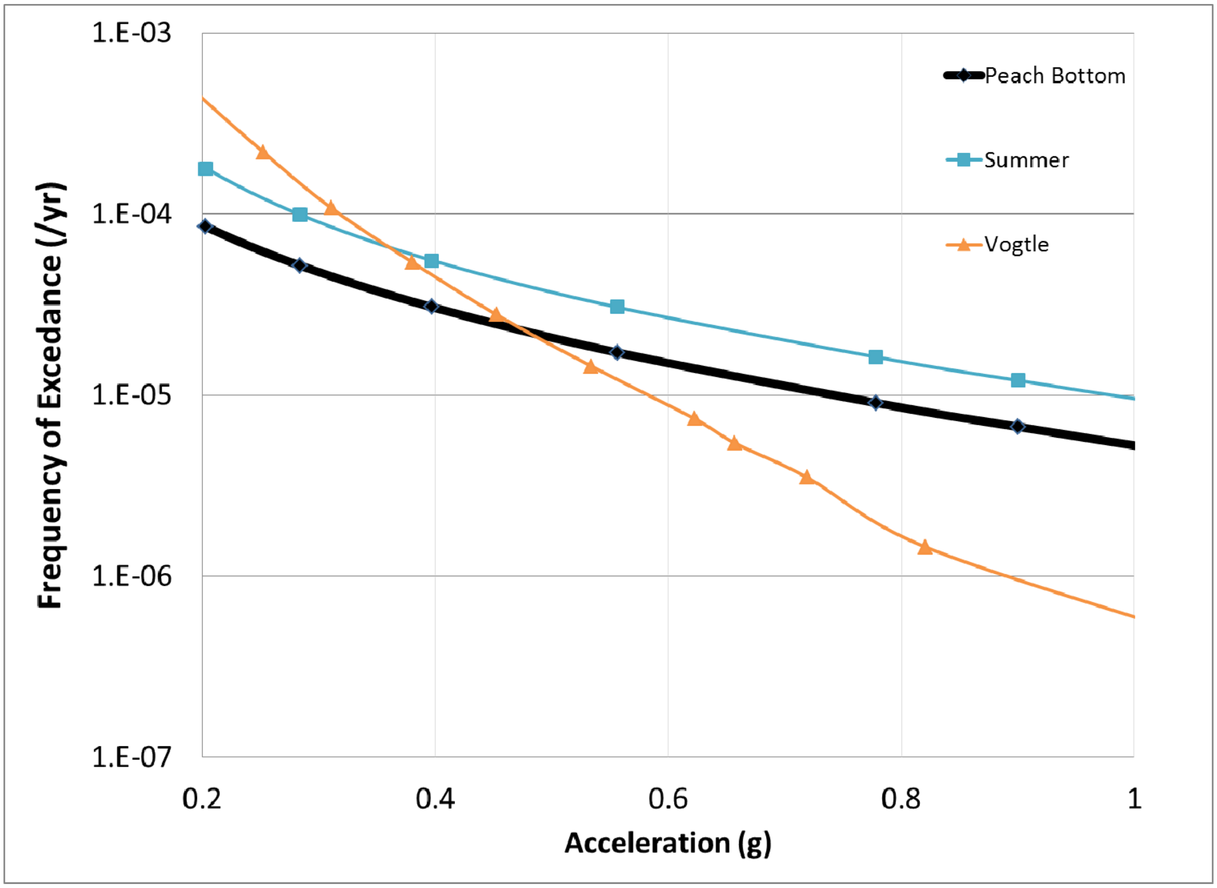


Figure 11 Comparison of annual PGA exceedance frequencies for new U.S. reactors (USGS 2008 model)

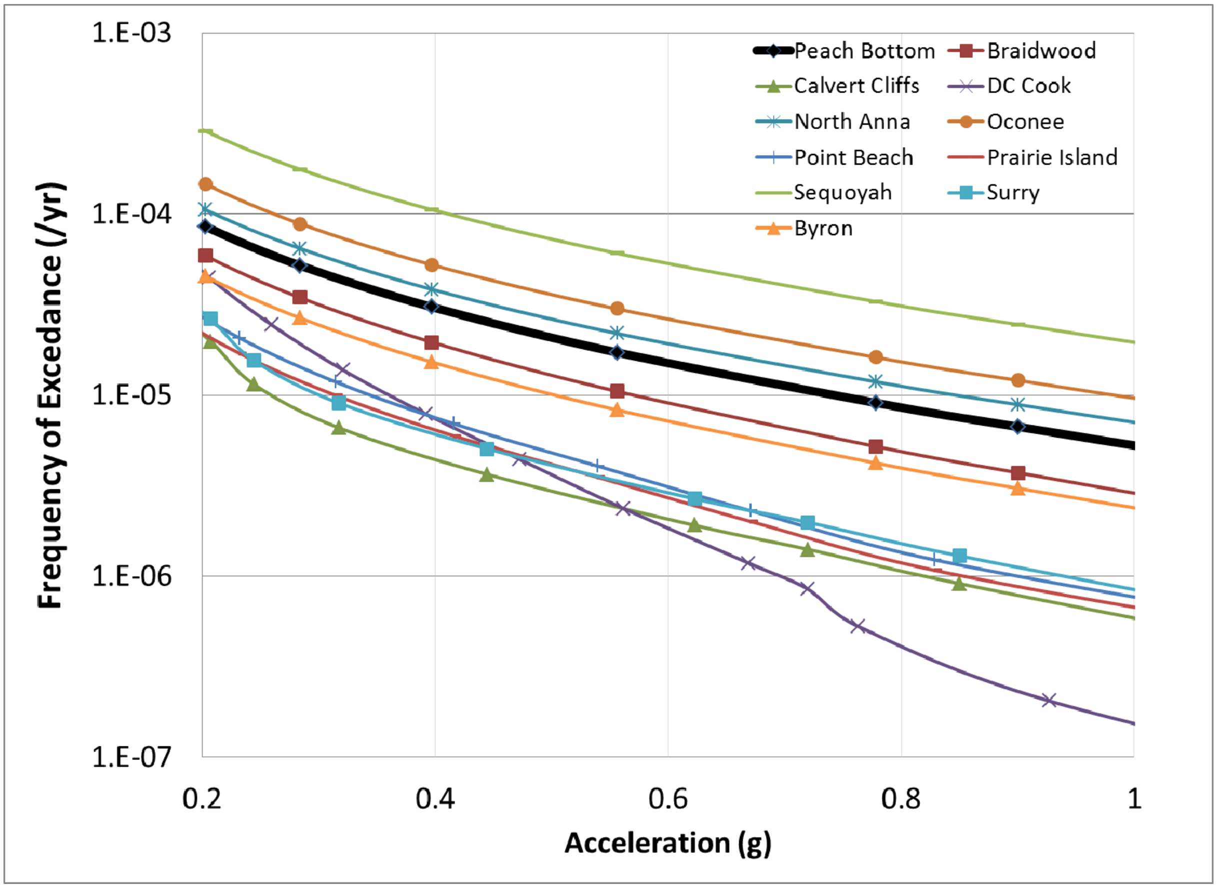


Figure 12 Comparison of annual PGA exceedance frequencies for U.S. reactors with a shared spent fuel pool (USGS 2008 model)

C.2.2 Characterization of Seismic Event Likelihood

As described in Section 3.2 of the SFPS (Ref. C.7), the hazard exceedance frequencies can be translated into initiating event frequencies by partitioning the PGA range into a number of discrete categories (bins) defined in terms of PGA intervals. These bins define a discrete number of seismic event scenarios with increasing intensity (PGA). Revision 1.01 of the NRC handbook entitled, "Risk Assessment of Operational Events, Volume 2—External Events," issued January 2008 (Ref. C.9), recommends the use of at least three bins unless plant-specific considerations require more bins. The SFPS used four bins.

Table 4 of the SFPS, reproduced in this analysis as Table 36, shows the resulting bins, along with the tabulated frequencies for various spectral and peak accelerations for Peach Bottom, the reference plant evaluated in that study. Note that for bin 4, the representative bin PGA has been set to 1.2g by convention, whereas for the other bins, it is the geometric mean of the interval endpoints.

Table 36 Seismic Bin Initiating Event Frequencies (Base Case)

Bin No.	Bin Range (g)	Bin PGA (g)	Approximate Initiating Event Frequency (USGS 2008 model) (/yr)
1	0.05 - 0.3	0.12	5.2×10^{-4}
2	0.3 - 0.5	0.4	2.7×10^{-5}
3	0.5 - 1.0	0.7	1.7×10^{-5}
4	> 1.0	1.2[1]	4.9×10^{-6}

1. Assumed based on PRA modeling convention.

Although the Peach Bottom hazard exceedance frequencies curves shown in Figures 7 through 10 fall close to the upper end of each group in terms of hazard estimates, there are some CEUS sites that exceed those estimates. For each SFP group, the site with the highest plant hazard exceedance frequency for peak ground accelerations greater than 0.6g was selected to produce the high estimate seismic bins and initiating event frequencies provided in Table 37.

Table 37 Seismic Bin Initiating Event Frequencies (High Estimate sensitivity)

SFP Group (Site Name)	Bin No.	Bin Range (g)	Bin PGA (g)	Approximate Initiating Event Frequency (USGS 2008 model) (/yr)
SFP Group 1 (Limerick)	1	0.05 - 0.3	0.12	6.8E-04
	2	0.3 - 0.5	0.4	3.6E-05
	3	0.5 - 1.0	0.7	2.2E-05
	4	> 1.0	1.2	7.1E-06
SFP Group 2 (Watts Bar)	1	0.05 - 0.3	0.12	1.7E-03
	2	0.3 - 0.5	0.4	8.1E-05
	3	0.5 - 1.0	0.7	4.9E-05
	4	> 1.0	1.2	1.5E-05
SFP Group 3 (Summer)	1	0.05 - 0.3	0.12	1.8E-03
	2	0.3 - 0.5	0.4	5.4E-05
	3	0.5 - 1.0	0.7	2.9E-05
	4	> 1.0	1.2	9.1E-06
SFP Group 4 (Sequoyah)	1	0.05 - 0.3	0.12	1.79E-03
	2	0.3 - 0.5	0.4	8.98E-05
	3	0.5 - 1.0	0.7	5.64E-05
	4	> 1.0	1.2	2.00E-05

The information above coupled with the review of previous studies (Ref. C.10) suggests that the base case frequency of a seismic event that could challenge the integrity of a SFP is on the order of 1.7×10^{-5} per year (i.e., approximately one event in 60,000 years) or less. Table 38 contrasts this frequency against other sources of information.

Table 38 Comparison of Seismic Frequencies from Various Sources

Source	Estimated initiating event frequency of a large seismic event	Notes
USGS 2008—Cost-benefit analysis base case	$1.7x10^{-5}$ /year[2] (one event in 60,000 years)	Frequency of seismic bin 3 (0.5 to 1.0 g) of 4 bins
USGS 2008—Cost-benefit analysis high estimate sensitivity	$5.6x10^{-5}$/year[3] (one event in 18,000 years)	
NUREG-1738[1]	$1.1x10^{-5}$/year (one event in 90,000 years)	Frequency of seismic hazard between 0.51g to 1.02g

1. Initiating event frequency reported is based on the LLNL models (Ref. C.11).
2. This value is from Table 36 for Bin No. 3.
3. This value is the SFP group 4 Bin No. 3 value from Table 37 and is the greatest magnitude for any of the SFP groups.

C.2.3 Spent Fuel Pool Initiator Release Frequency

Section 1.5 of the SFPS (Ref. C.7) provides an overview of contributors to SFP risk. The majority of SFP risk emanates from a loss of water from a sizeable leak in the SFP or a boil off in which operator action to inject water into the pool for an extended period is precluded. The release frequency from the SFP can then be characterized as the frequency of the initiator causing fuel uncovery multiplied by the probability of a release given fuel uncovery for the specific initiating event. The total release frequency is the sum of the frequency of releases from cask drops, seismic events, and other initiators. This value is given by:

$$F_{release} = \sum_i F_{initiator_i} \, x \, P_{release_i}$$

Where $F_{initiator}$ includes:

- F_{drop} = frequency of spent fuel uncovery from cask drops
- $F_{seismic\text{-}bin\ 3}$ = frequency of spent fuel uncovery from seismic bin 3 event
- $F_{seismic\text{-}bin\ 4}$ = frequency of spent fuel uncovery from seismic bin 4 event
- F_{other} = frequency of spent fuel uncovery from sources other than cask drops and seismic
- $P_{release}$ = probability of release given spent fuel uncovery for specific initiators

Source: Derived from SFPS, Section B.4 (Ref. C.7).

The SFPS provides a detailed analysis of the consequences, for a particular site and a calculation of $F_{seismic}$ for seismic bin 3, depicted as a hazard exceedance frequency range provided in Table 36.

The SFPS did not analyze initiators that contribute to SFP risk other than for seismic events defined by seismic bin no. 3. However past studies, such as NUREG-1353 (Ref. C.12) and NUREG-1738 (Ref. C.10), evaluated additional events that could contribute to risk and consequences from SFP accidents. Table 42 summarizes these initiating-event-class fuel uncovery frequencies. Uncovery frequencies taken from past studies depend on the assumptions stated in those studies. Additionally, seismic bin no. 4 is included by extrapolating the results of the SFPS. For seismic bin no. 3 and bin no. 4 events, the uncovery frequency is the product of the initiating event frequency, ac power fragility, and the liner fragility.

The SFPS (Ref. C.7) uses an alternating current (ac) power fragility value of 0.84 taken from NUREG-1150 (Ref. C.13) as a surrogate for the conditional probability of normal SFP cooling

and makeup not being available following a 0.7g earthquake. This simplifying assumption was made in light of the fact that the SFPS is not a probabilistic risk assessment but rather a consequence analysis with probabilistic considerations.

In reality, the availability of normal SFP cooling and makeup would be a combination of the ac power fragility, the fragility of the actual equipment and its support equipment, and operator actions to recover SFP cooling capabilities using additional mitigation equipment and strategies implemented in response to Order EA-12-049 (Ref. C.14). The modeling and consideration of these guidance and strategies to maintain or restore SFP cooling capabilities following a beyond-design-basis external event on a plant-specific basis may result in a value for SFP cooling and makeup failure conditional probability that may differ from the NUREG-1150. Because a documented ac power fragility analysis that covers U.S. SFPs is not readily available, a conservative bounding value of 1.0 is used in this analysis.

Section 4.1.5 of the SFPS (Ref. C.7) describes the results from the nonlinear finite element analysis to estimate the likelihood of leakage from concrete cracking and related SFP liner failure for the 0.7g earthquake. Figure 27 from the SFPS shows that the maximum membrane effective strain is about 3.7 percent. Based on this calculated liner strain for the 0.7g earthquake, a structural analysis of the pool estimates that the SFP in this study has a 90 percent probability of surviving the 0.7g earthquake with no liner leakage (or conversely, a 10 percent probability of damaging the liner such that leakage will occur). As a result, a liner fragility value of 0.1 is used in the SFPS for the seismic bin No. 3 initiating event. NUREG/CR-5176 (Ref. C.15) provides the fragility for the walls of a PWR located in the CEUS as having a 98 percent probability of surviving the 0.7g earthquake with no liner leakage (or conversely, a 2 percent probability of damaging the liner such that leakage will occur).

For the seismic bin 4 initiating event (i.e., 1.2g earthquake), a comparable structural analysis is not performed in the SFPS to determine the liner fragility value for the reference BWR Mark I plant. As a result, a bounding value of 1.00 for the seismic bin no. 4 earthquake is used in this analysis for Group 1 liner fragility high estimate, even though a detailed analysis may be able to justify a value a factor of 2 or more lower. NUREG/CR-5176 provides the fragility for the walls of a PWR located in the CEUS as having an 84 percent probability of surviving the 1.2g earthquake with no liner leakage (or conversely, a 16 percent probability of damaging the liner such that leakage will occur). As a result, a value of 0.16 is used for the seismic bin no. 4 earthquake low estimate in this analysis for Groups 2, 3, and 4 liner fragility. A summary of these liner fragility values is provided in Table 39.

Table 39 Liner Fragility Values as a Function of Spent Fuel Pool Group and Seismic Bin

SFP Group	Seismic Bin	Liner Fragility		
		Low Est.	Base Case	High Est.
1	Bin 3	10%	10%	100%
	Bin 4	50%	100%	100%
2, 3, & 4	Bin 3	2%	5%	25%
	Bin 4	16%	50%	100%

Past studies have reached generally similar conclusions about the relative contribution to risk from the seismic initiating events considered. Table 40 summarizes the impact of the above modeling assumptions when comparing the seismic initiating event fuel uncovery frequencies from previous SFP accident regulatory analyses.

Table 40 Frequency of Spent Fuel Pool Fuel Uncovery for Seismic Events

Reference	Reactor Type / SFP Grouping	Seismic Event Contribution to SFP Fuel Uncovery (per 10^6 reactor-years)	
		Base case	High estimate sensitivity
NUREG-1353 (Ref. C.12) (best estimate)	BWR[1]	6.7	N/A
	PWR	1.8	N/A
NUREG-1738[2] (Ref. C.10)	All	2.0	N/A
This analysis[3]	SFP Group 1	6.6	29
	SFP Group 2	3.3	27
	SFP Group 3	3.3	16
	SFP Group 4	3.3	34

1. The NUREG-1353 BWR seismic structural failure value was not multiplied by the stated conditional probability of having a zirconium fire of 0.25.
2. NUREG-1738 presented results for the two different seismic hazard models in wide use at the time (the Electric Power Research Institute and Lawrence Livermore National Labs models). The larger of the two values is listed above.
3. The base case initiating event frequency value is from Table 36. The high estimate sensitivity initiating event frequency value is from Table 37. The likelihood of fuel uncovery is a product of initiating event frequency, ac power fragility (1.0), and liner fragility (value depends on case being evaluated as displayed in Table 39). A value of 1.0 for ac power or pool liner failure mean represents a 100 percent likelihood of failure.

The SFPS evaluated a specific BWR Mark I reference site for a specific initiating event. When spent fuel in a pool becomes uncovered, it may still be coolable from natural circulation of air once the water level clears the baseplate of the racks, depending on the amount of decay heat during the operating cycle. In Section 12.1 of the SFPS, the fuel is estimated to be air coolable for all but roughly 10 percent of the operating cycle. Factors affecting this value include the amount of fuel in the pool, its configuration, geometry of the fuel racks, etc. A partial draindown event with channeled fuel or solid-walled high-density racks could impede airflow. In this case with no natural circulation of air through the racks, the fuel could only be cooled by steam generated by the fuel itself or through the application of water spray. For these mechanisms to be effective, a substantial fraction of the decay heat must be absorbed by the remaining water to generate adequate steam flow or adequate spray flow must be applied. Distributed fuel assemblies late in the operating cycle may lose a significant portion of the remaining decay heat to radiation heat transfer and limited convective heat transfer at temperatures below the runaway oxidation threshold, and therefore, the assemblies would not reach a self-sustaining oxidation condition.

The spent fuel is expected to retain an air coolable geometry following a seismic event that causes a moderate to large crack in the pool, and information provided in NUREG/CR-5176 (Ref. C.15), which concludes that there is high confidence that SFP racks are sufficiently robust to remain generally intact with their fuel channels open supports this assumption. Furthermore, prior studies conclude that severe earthquakes are not expected to result in catastrophic failure of SFP structural walls and floor or fuel racks. However, there is considerable variability in U.S. SFP size, capacity, rack type, and geometry as well as the amount and age of the fuel in the pool and its burnup. Because plant-specific analyses is not available to verify that U.S. SFPs

and racks retain their structural integrity and air-coolable geometry following a beyond-design basis seismic event for U.S. SFPs, a bounding approach was used to evaluate the sensitivity of assuming the spent fuel is not air-coolable following a seismic bin 3 or seismic bin 4 earthquake. For bin 3, this modeling represents the scenario in which the seismic event results in a partial draindown condition (i.e., liner tearing at the walls) with some water remaining at the bottom of the SFP. In the SFPS, the fuel is estimated to not be air coolable for 10 percent of the operating cycle following a Bin 3 seismic event based on the SFP configuration and other factors. This value was used for the base case of SFP Group 1. For stronger seismic events for SFP Group 1 , the other SFP Group base cases, and for all high estimates, a bounding value of 100 percent for the conditional probability of release was assumed as shown in Table 41.

Table 41 Fraction of Time Either Excessive Heat or a Partial Spent Fuel Pool Draindown Prevents Natural Circulation Cooling of the Spent Fuel

<table>
<tr><th rowspan="2">SFP Group</th><th rowspan="2">Seismic Bin</th><th colspan="3">Inadequate Spent Fuel Cooling Fraction</th></tr>
<tr><th>Low Est.</th><th>Base Case</th><th>High Est.</th></tr>
<tr><td rowspan="2">1</td><td>Bin 3</td><td>10%</td><td>10%</td><td rowspan="2">100%</td></tr>
<tr><td>Bin 4</td><td>30%</td><td>100%</td></tr>
<tr><td rowspan="2">2, 3, & 4</td><td>Bin 3</td><td>10%</td><td rowspan="2">100%</td><td rowspan="2">100%</td></tr>
<tr><td>Bin 4</td><td>30%</td></tr>
</table>

For the postulated cask drop event, the spent fuel is expected to retain an air coolable geometry because a cask drop accident would most likely affect the fuel pool floor in the cask loading area. Typically overhead cranes used to move casks are designed to meet single failure proof criteria, and have interlocks and administrative controls that limit the motion of the crane over the SFP to the cask loading area, where no fuel is stored. Although improbable, crane failure is more likely to occur during hoisting operations when many components contribute to holding the cask than during translational motion when the hoist holding brakes are set. The hoisting activities occur over the cask loading area, and, in that location, the cask, if dropped, could have sufficient potential energy to damage the SFP floor. However, a structural analysis to evaluate all U.S. SFPs was not performed to verify that spent fuel and racks retain their structural integrity and air-coolable geometry following a cask drop event. Given the uncertainties and plant-specific variabilities involved, a bounding approach was used by assuming the spent fuel is not air-coolable following a cask drop accident. This was done by assigning a bounding value of 1.0 for the conditional probability of release for the cask drop unsuccessful mitigation event.

To calculate the total release frequency, the uncovery frequencies are multiplied by the conditional probability of release for each initiating event class. The conditional probability of release depends on the fraction of the operating cycle where the fuel is not air-coolable. As previously discussed in this section, given the uncertainties and plant-specific variability involved, a bounding approach was used. For SFP draindown events (e.g., seismic events and cask drops) the bounding approach used in this analysis assumes these events are not air-coolable. For the nonseismic and noncask drop events taken from previous studies, the nature of the events may lead to a situation similar to a partial draindown where the rack baseplate is not cleared and airflow is impeded. For these events, the spent fuel is not air-coolable and the conditional release probability is assumed to be 100 percent.

When mitigation is credited, the SFPS found that successful deployment of mitigation decreased the conditional probability by a factor of 19 for the seismic bin no. 3 event analyzed at the reference plant using mitigation measures required under 10 CFR50.54(hh)(2)

(Ref. C.16). The SFPS does not consider the post-Fukushima SFP instrumentation required under Order EA-12-051 (Ref. C.17) and severe accident mitigation equipment and mitigation strategies (Ref. C.18) required under Order EA-12-049 (Ref. C.14), which is being implemented by the plants and is intended to increase the likelihood of restoring or maintaining power and mitigation capability during severe accidents. In reality, the effectiveness of post-Fukushima improvements to severe accident mitigation measures will depend on a variety of factors, which the SFPS did not consider but are expected to increase the likelihood that deployment of mitigation measures is successful. Each plant has developed a plant-specific analysis and strategies for coping with the effects of the beyond-design-basis natural events that may challenge its SFP cooling and makeup capabilities. For the purposes of this analysis, it was estimated that mitigation if successfully deployed in time decreased the conditional probability by a factor of 19 for all initiating events as determined in the SFPS. This analysis used a conservative approach by crediting successful mitigation for the low-density SFP alternative and assumed no successful mitigation for the high-density SFP storage regulatory baseline.

Table 42 summarizes the non-seismic initiating event fuel uncovery frequencies, the conditional probability of release, and the total release frequency without mitigation.

Table 42 Release Frequencies for Spent Fuel Pool Initiators for Nonseismic Events

Initiating Event Class	Initiating Event Fuel Uncovery Frequency (per r-yr)	Conditional Probability of Release (Unsuccessful mitigation)	Release Frequency (Unsuccessful mitigation) (per r-yr)
Cask / heavy load drop	$2x10^{-7}$ (2)	8.2% - 100%	$1.64x10^{-8}$ – $2.00x10^{-7}$
LOOP – severe weather	$1x10^{-7}$ (2)	100%	$1.00x10^{-7}$
LOOP – other	$3x10^{-8}$ (2)	100%	$3.00x10^{-8}$
Internal fire	$2x10^{-8}$ (2)	100%	$2.00x10^{-8}$
Loss of pool cooling	$6x10^{-8}$ (1)	100%	$6.00x10^{-8}$
Loss of water inventory	$1x10^{-8}$ (2)	100%	$1.00x10^{-8}$
Inadvertent aircraft impacts	$6x10^{-9}$ (2)	100%	$6.00x10^{-9}$
Missiles – general	$1x10^{-8}$ (1)	100%	$1.00x10^{-8}$
Missiles - tornado	$1x10^{-9}$ (2)	100%	$1.00x10^{-9}$
Pneumatic seal failures	0 – $3x10^{-8}$ (1,4)	100%	0 – $3.00x10^{-8}$
Total			$2.53x10^{-7}$ – $4.37x10^{-7}$

1. Values from NUREG-1353 (Ref. C.12). These numbers are applicable to all reactors and were not adjusted by the stated conditional probability of having a zirconium fire of 0.25 for BWR reactors.
2. Values from NUREG-1738 (Ref. C.10).
3. The operating cycle phase is equal to 8.2% (e.g., 60/730) for 2-year refueling cycles and 11.0% (e.g., 60/547.5) for 18-month refueling cycles.
4. Although many plants use gates with mechanical seals that are kept under pressure by passive mechanical means (i.e., do not depend on air pressure, ac power, or dc power) to prevent leakage, there may be some plants that continue to use pneumatic seals. This analysis conservatively includes the pneumatic seal failures as an initiating event for U.S. PWR SFPs.

Table 43 provides the total release frequency by SFP group for all SFP event initiators.

Table 43 Total Release Frequency by Spent Fuel Pool Group

SFP Group	Seismic Bin	Bin Frequency (per year)	Liner Fragility	Fraction Not Air Coolable	Seismic Release Frequency (per year)	Non-Seismic Release Frequency (per year)	Total Release Frequency per Group (per year)
Low Estimate							
1	3	$1.65x10^{-5}$	10%	8%	$1.35x10^{-7}$	$2.53x10^{-7}$	$1.12x10^{-6}$
	4	$4.90x10^{-6}$	50%	30%	$7.35x10^{-7}$		
2,3,4	3	$1.65x10^{-5}$	2%	8%	$3.30x10^{-8}$	$2.83x10^{-7}$	$5.51x10^{-7}$
	4	$4.90x10^{-6}$	16%	30%	$2.35x10^{-7}$		
Base Case							
1	3	$1.65x10^{-5}$	10%	8%	$1.35x10^{-7}$	$4.37x10^{-7}$	$5.47x10^{-6}$
	4	$4.90x10^{-6}$	100%	100%	$4.90x10^{-6}$		
2,3,4	3	$1.65x10^{-5}$	5%	100%	$8.25x10^{-7}$	$4.67x10^{-7}$	$3.74x10^{-6}$
	4	$4.90x10^{-6}$	50%	100%	$2.45x10^{-6}$		
High Estimate							
1	3	$2.24x10^{-5}$	100%	100%	$2.24x10^{-5}$	$4.37x10^{-7}$	$2.99x10^{-5}$
	4	$7.09x10^{-6}$	100%	100%	$7.09x10^{-6}$		
2	3	$4.92x10^{-5}$	25%	100%	$1.23x10^{-5}$	$4.67x10^{-7}$	$2.79x10^{-5}$
	4	$1.51x10^{-5}$	100%	100%	$1.51x10^{-5}$		
3	3	$2.95x10^{-5}$	25%	100%	$7.38x10^{-6}$	$4.67x10^{-7}$	$1.69x10^{-5}$
	4	$9.10x10^{-6}$	100%	100%	$9.10x10^{-6}$		
4	3	$5.64x10^{-5}$	25%	100%	$1.41x10^{-5}$	$4.67x10^{-7}$	$3.46x10^{-5}$
	4	$2.00x10^{-5}$	100%	100%	$2.00x10^{-5}$		

C.2.1 Seismic Initiator Frequency Assumptions Sensitivity

As illustrated in Table 44, the combination of conservative seismic initiator modeling assumptions with the bounding seismic source zone characterization for any spent fuel pool located in the CEUS results in public health (accident) benefit values increasing by a factor between 4.5 and 9.3 times the averted public health (accident) dose calculated for the base case.

Table 44 Sensitivity of Public Health (Accident) Benefits within 50 Miles to Changes in Seismic Initiator Frequency Assumptions

SFP Group	Seismic Initiator Case	Dose (averted person-rem per pool)	Benefits (2012 million dollars)		
			2% NPV	3% NPV	7% NPV
1	Base Case	1,740	\$2.72	\$2.42	\$1.62
	High Estimate	9,510	\$14.86	\$13.25	\$8.87
2	Base Case	1,630	\$2.45	\$2.15	\$1.38
	High Estimate	12,100	\$18.23	\$16.02	\$10.25
3	Base Case	3,020	\$3.14	\$2.37	\$0.99
	High Estimate	13,650	\$14.21	\$10.75	\$4.49
4	Base Case	1,690	\$2.62	\$2.33	\$1.54
	High Estimate	15,660	\$24.23	\$21.53	\$14.24

Offsite Property Cost Offset Sensitivity to Seismic Initiator Frequency Assumptions

Although the SFPS reference plant hazard exceedance frequencies curves discussed in Appendix section C.2.1 of this analysis fall close to the upper end of each SFP group in terms of hazard estimates, there are some CEUS sites that exceed those estimates. To analyze the seismic risk hazard for these CEUS sites, a high estimate using the bounding plant hazard exceedance frequency curve is used to produce the high estimate seismic bins and initiating event frequencies. These seismic frequencies are provided in Table 37. Several other bounding assumptions are also made to arrive at the bounding SFP release frequency provided in Table 43 These include the loss of all ac power for all SFP initiators, a conservative liner fragility value (see Table 39) even though a realistic analysis may be able to justify a value that is lower by factor of 2 or more, and assuming a bounding value of 1.0 for the conditional probability for failure to successfully mitigate the high-density storage spent fuel accident. These conservative (bounding) assumptions were used to calculate the offsite property cost offset estimate sensitivity to the seismic initiating frequency assumptions provided in Table 45.

Table 45 Sensitivity of Offsite Property Cost Offset within 50 Miles to Changes in Seismic Initiator Frequency Assumptions

SFP Group	Seismic Initiator Case	Offsite Property Cost Offsets (2012 million dollars)		
		2% NPV	3% NPV	7% NPV
1	Base Case	7.65	6.83	4.57
	High Estimate	41.85	37.32	24.98
2	Base Case	11.50	10.10	6.46
	High Estimate	85.65	75.24	48.14
3	Base Case	12.07	9.13	3.81
	High Estimate	54.65	41.33	17.25
4	Base Case	14.35	12.75	8.44
	High Estimate	132.58	117.80	77.92

C.2.5 Duration of Onsite Spent Fuel Storage Risk

For this cost-benefit analysis, it is assumed that the each nuclear power plant operates through the term of its operating license and that the licensee continues to store spent fuel in the plant's SFP following commercial operation[24] to allow the spent fuel to cool sufficiently before placing into dry storage. Other than for operating reactors that have indicated they would not seek a license renewal, this analysis assumes that remaining operating reactors' operation expectancy will include a 20-year license renewal period, unless stated otherwise.[25] As a result, the average license will expire in 2039. Table 1 summarizes the average reactor operation expectancy by the identified SFP groupings.

24 Decommissioning of the unit must be completed within 60 years of permanent cessation of operations under 10 CFR 50.82, "Termination of License." Completion of decommissioning beyond 60 years will be approved by the Commission only when necessary to protect public health and safety.

25 Six U.S. nuclear power plant units have announced early retirements (with year of closure in parentheses) are Crystal River 3 (2013), Kewaunee (2013), San Onofre Units 2 and 3 (2013), Vermont Yankee (2014), and Oyster Creek (2019).

C.2.6 Dollar per Person-Rem Conversion Factor

Using the dollar value of the health detriment and a risk factor that establishes the nominal probability for stochastic health effects attributable to radiological exposure (fatal and nonfatal cancers and hereditary effects) provides a dollar per person-rem of $2,000, rounded to the nearest thousand, according to NUREG-1530, "Reassessment of NRC's Dollar per Person-Rem Conversion Factor Policy," dated December 1995 (Ref. C.19).

The NRC currently uses a value of statistical life (VSL)[26] of $3 million based on NUREG-1530, and a cancer risk factor of 7.0×10^{-4}, which is a reduction to the closest significant digit of a recommendation by the International Commission on Radiation Protection (ICRP) in Publication No. 60. Therefore, the dollar per person-rem is equal to $3 million times 7.0×10^{-4} rounded to the nearest thousand (because of uncertainties) or $2,000.

C.2.7 Onsite Property Decontamination, Repair, and Refurbishment Costs

SFP accident risks have significant contributions from onsite property monetary losses (e.g., repair and refurbishment) and plant decontamination. The risk dominant accident sequences involve the failure of the pool because of seismic or load drop events resulting in the loss of pool integrity. This scenario results in loss of SFP water inventory, Zircaloy cladding fire initiation with propagation through the spent fuel assemblies stored in the pool, and an uncontrolled radiological release from the reactor building. The NRC assumes that, based on the current regulatory framework, with insights from the Fukushima Dai-ichi accident, that onsite property would be radiologically affected in the following way. The consequences of a spent fuel fire are expected to be similar to the severe reactor accidents resulting in core damage and possible fuel melting as defined in NUREG/CR-5281, Section 3.2.4 (Ref. C.20). Based on this reference, the cleanup and decontamination costs are estimated to be approximately $165 million (1983 dollars) and the cost for permanent disposal of the damaged fuel is $26 million (1983 dollars). Using Table C.95 from the RA Handbook (Ref. C.21), the pool repair is expected to cost $72 million (1983 dollars). Adjusting these estimated costs using the CPI-U inflator formula and using a multiplier of three to model the high estimate and a divider of two to model the low estimate results in the values provided in Table 46.

Table 46 Onsite Property Decontamination, Repair, and Refurbishment Costs

Onsite Property Cost Element	1983 dollars			2012 dollars		
	Best Estimate	High Estimate	Low Estimate	Best Estimate	High Estimate	Low Estimate
Cleanup and decontamination	$165,000,000	$495,000,000	$82,500,000	$380,358,000	$1,141,074,000	$190,179,000
Repair Pool	$72,000,000	$216,000,000	$36,000,000	$165,974,000	$497,922,000	$82,987,000
Disposal of damaged fuel	$26,000,000	$78,000,000	$13,000,000	$59,935,000	$179,805,000	$29,968,000
Total	$263,000,000	$789,000,000	$131,500,000	$606,267,000	$1,818,801,000	$303,134,000

26 The value of a statistical life (VSL) is the monetary value of a mortality risk reduction that would prevent one statistical (as opposed to an identified) death (Ref. C.22). The VSL is a key component in the calculation of the dollar per person-rem value, which is the product of the VSL multiplied by a risk coefficient.

C.2.8 Replacement Energy Costs

Replacement energy costs are the costs for replacing the energy from the nuclear power plant because of a plant shutdown to install required equipment or because of an accident.[27] The NRC assumes that replacement energy costs would be required until onsite decontamination and repair efforts are completed or the unit is retired. The NRC assumes that the cost per year of replacement energy would be about $2.3 million (2012 dollars).

The NRC assumes that licensees engage in power purchase agreements (PPA)[28] to economically purchase replacement power. A PPA is a legal contract between an electricity generator (licensee) and a power purchaser. The NRC assumes that a licensee will not be able to replace the power through other generation for 7 years and would have to buy power from the market. Although not all licensees may have PPAs, the licensee will still replace the lost energy any time that the nuclear power plant is not operating to meet its electrical power supply obligations. The NRC assumes that after 7 years, the onsite decontamination and repair efforts are completed or the unit is retired and other power sources will be developed to replace the unit's lost electrical generation capability. Therefore, the NRC assumes that the undiscounted cost of replacement energy would be $15.9 million.

C.2.9 Occupational Worker Exposure (Accident)

There are two types of occupational exposure related to accidents: short-term and long-term. The first occurs at the time of the accident and during the immediate management of the emergency. The second is a long-term exposure, presumably at significantly lower individual rates, associated with the cleanup and refurbishment or decommissioning of the damaged facility. The value gained in the avoidance of both types of exposure is conditioned on the change in frequency of the accident's occurrence.

The experiences at the Three Mile Island Unit 2 (TMI-2), the Chernobyl, and the Fukushima nuclear power plants illustrated that significant occupational exposures could result from performing activities outside the control room during a power reactor accident. At TMI-2, the average occupational exposure related to the incident was approximately 1.0 rem, with a collective dose of 1,000 person-rem occurring over a 4-month span, after which time occupational exposure approached pre-accident levels. For Chernobyl, the average dose for persons closest to the plant was 3.3 person-rem (Ref. C.21, p. 5.30), yielding a collective dose of 3,300 person-rem.

The accident at Fukushima involved release of both short-lived and long-lived radionuclides from the reactor cores within Units 1, 2, and 3, and no release from the fuel stored in the SFPs. Significant changes in the release of radioactivity occurred following changes in the status of the

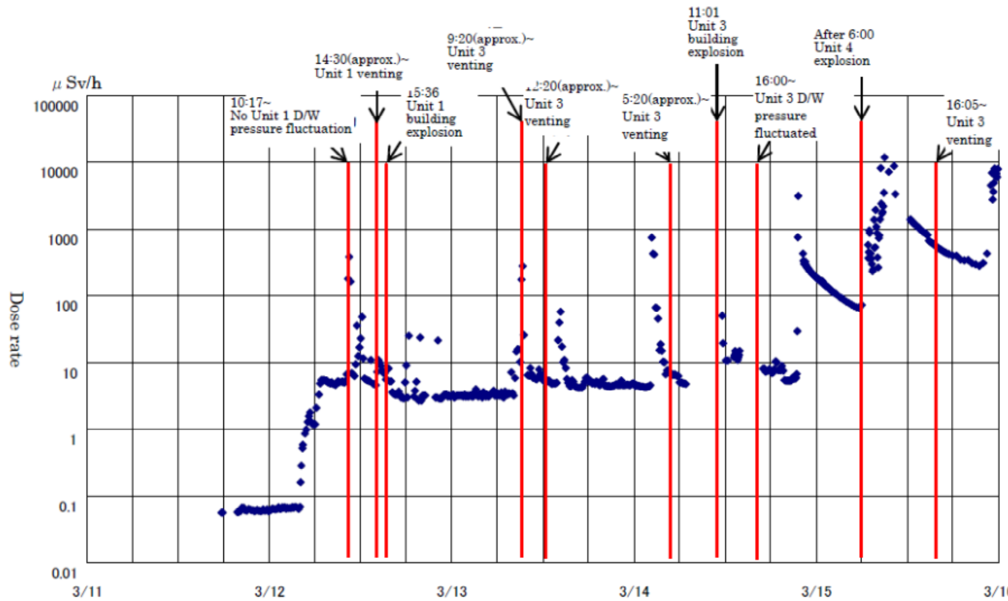
[27] The replacement energy cost is only the cost to buy the energy for production on the market. Therefore, the cost would be the cost of buying the cheapest energy. These estimates do not include transmission or distribution costs.

[28] A power purchase agreement is a contract between two parties, one who generates electricity for the purpose of sale (the seller) and one who is looking to purchase electricity (the buyer). The PPA defines all of the commercial terms for the sale of electricity between the two parties, including when the project will begin commercial operation, schedule for delivery of electricity, penalties for under delivery, payment terms, and termination.

core, primary containment, and secondary containment. After the Fukushima unit 1 building explosion on March 12, 2011, the unit 3 building explosion on March 14, and the unit 4 building explosion, which released radioactivity from Unit 3 because of a shared ventilation system, and the exposure of the unit 2 reactor fuel rods on March 15, radioactive materials were released into the environment and surrounding areas of the Fukushima Dai-ichi nuclear power plant. Measurement and evaluation of radiation exposure levels for workers engaged in emergency work at the Fukushima Dai-ichi NPS have been implemented continuously since the Tohoku earthquake.

As shown in Figure 13, the dose rate in the vicinity of the main gate at the Fukushima Dai-ichi site near the time of the Unit 4 explosion varied between 20 mrem and 1.0 rem per hour (between 200 and 10,000 μSv per hour).

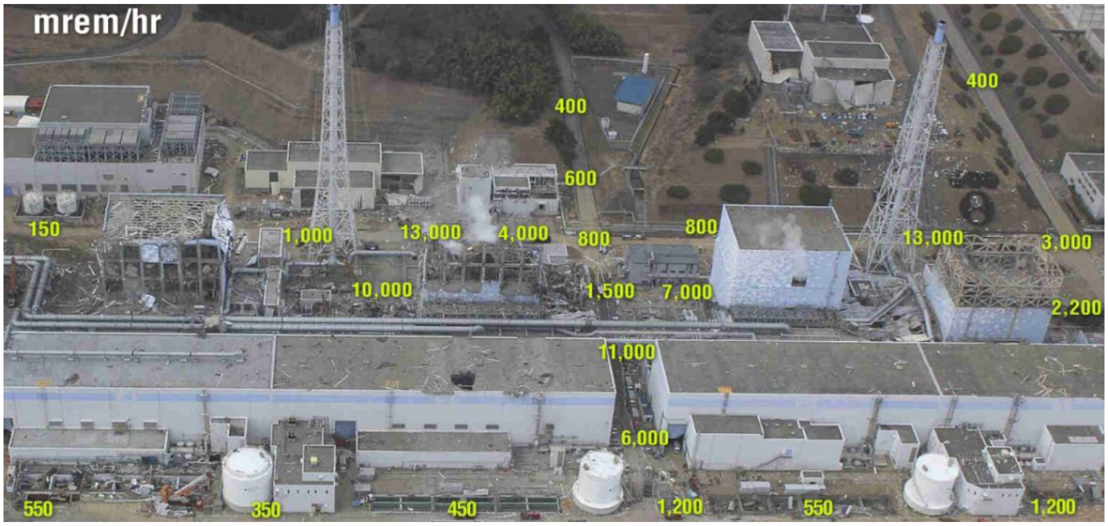
Figure 13: Dose rate in vicinity of Fukushima Dai-ichi nuclear plant site main gate between March 11 and March 16, 2011



Source: Fukushima Nuclear Accident Analysis Report p. 371 (Ref. C.24).

On March 22 and 23, surveys of the airborne radioactivity and dose rates around the Fukushima Dai-ichi site were collected and documented. The dose rates are shown on Figure 14.

Figure 14: Fukushima Dai-ichi site dose rates between March 22 and March 23, 2011



Source: INPO 11-005, p 41 (Ref. C.23).

The distribution of total monthly exposure for workers engaged in radiation work at the Fukushima Dai-ichi nuclear plant site for the first 3 months following the March 2011 accident is provided in Table 47.

Table 47 Average Accident Occupational Exposure at Fukushima Dai-ichi Nuclear Power Plant from March to May 2011

Total Radiation Exposure (mSv)	Number of Plant Workers Exposed		
	March 2011[1]	April 2011[2]	May 2011[3]
≥ 250	6	0	0
200 - 249	2	0	0
150 - 199	14	0	0
100 - 149	77	0	0
50 - 99	309	3	0
20 - 49	859	81	19
10 - 19	1041	310	144
< 10	1434	3214	2854
Total number of workers	3742	3608	3017

Notes:
1. Maximum March 2011 occupational exposure was 670.4 mSv.
2. Maximum April 2011 occupational exposure was 69.3 mSv.
3. Maximum May 2011 occupational exposure was 41.6 mSv.
4. One mSv is equal to 0.1 rem.

Source: Wada et al, Occupational and Environmental Medicine, 2012 August; 69(8): p. 600 (Ref. C.26).

To estimate the monthly total occupational radiation exposure received by all workers, a high estimate, base case, and low estimate were calculated based on the maximum category value, the midpoint category value, and the first quartile category value. The results are tabulated in Table 48.

Table 48 Estimated Immediate Accident Occupational Monthly Exposure at Fukushima

Radiation Exposure (mSv)	Best Estimate Category Radiation Exposure (mSv)			High Estimate Category Radiation Exposure (mSv)			Low Estimate Category Radiation Exposure (mSv)		
	March 2011	April 2011	May 2011	March 2011	April 2011	May 2011	March 2011	April 2011	May 2011
≥ 250	460.2			670.4			355.1		
200 - 249	224.5			249			212.25		
150 - 199	174.5			199			162.25		
100 - 149	124.5			149			112.25		
50 - 99	74.5	69.3		99	69.3		62.25	62.25	
20 - 49	34.5	34.5	34.5	49	49	41.6	27.25	27.25	27.25
10 - 19	14.5	14.5	14.5	19	19	19	12.25	12.25	12.25
< 10	5	5	5	10	10	10	2.5	2.5	2.5
Total Monthly Dose	90,200	23,600	17,000	125,600	42,200	32,100	72,500	14,200	9,400
Avg Worker Dose	24.1	6.5	5.6	33.6	11.7	10.6	19.4	3.9	3.1

The immediate accident occupational exposure for a SFP accident shown in Table 49 is estimated based on the Fukushima data and the following assumptions:

- The immediate accident period lasts for 1 year.
- The workforce during the immediate accident period is 3,700 workers.
- The average worker radiation exposure remains constant at the May 2011 value from May 2011 through February 2012.

Table 49 Immediate Accident Occupational Exposure for a Spent Fuel Pool Fire

Case	Immediate Accident Occupational Exposure (averted person-rem)
Low Estimate	18,070
Best Estimate	28,380
High Estimate	48,880

After the immediate response to a SFP accident, a long process of cleanup and refurbishment or decommissioning will follow. The Fukushima Nuclear Accident Analysis Report states, "The average value for 5,128 people in April of 2012 was 1.07 mSv per worker because of decreasing trends in environment dose rates (Ref. C.24, p 415). The NRC assumes that the process of cleanup and refurbishment or decommissioning will begin 1 year after the accident and will take 7 years to complete. During those 7 years, the NRC assumes that each occupational worker at the damaged reactor site will be exposed to 1.07 mSv per month (0.107 rem per month) for the duration of the cleanup and refurbishment or decommissioning. Assuming the average value for 5,128 workers would remain for the duration yields a cumulative long-term occupational dose of 46,000 person-rem.

In NUREG/CR-5281 (Ref. C.20), Jo et al. (1989) conducted what essentially amounted to a regulatory analysis of a non-reactor nuclear fuel cycle facility using Heaberlin, et al 1983 Handbook (Ref. C.27) as guidance. The accidental occupational exposure was assumed to be similar to that from TMI-2, which is 4,580 person-rem.

As described in the RA Handbook (Ref. C.21, p 5.30), the DOE (1987) summarized results on the collective dose received by the populace surrounding the Chernobyl accident. Average dose equivalents of 3.3 rem per person, 45 rem per person, and 5.3 rem per person were estimated for residents within 3 km, between 3 km and 15 km, and between 15 km and 30 km of Chernobyl, respectively (Ref. C.28, p. A-5). Assuming 1,000 workers and a 4.2 multiplier, an estimate radiation exposure of 14,000 person-rem results.

Site worker exposures following a SFP accident could be greater than that of a reactor core melt accident. This is because a SFP stores significantly more fuel assemblies than a reactor core. Given the uncertainties in existing data and variability in severe accident parameters and worker response, Table 50 provides the long-term occupational dose used in this analysis to analyze SFP accidents.

Table 50 Long-Term Accident Occupational Exposure for a Spent Fuel Pool Fire

Case	Long-Term Accident Occupational Exposure (averted person-rem)
Low Estimate	4,580
Best Estimate	14,000
High Estimate	46,000

C.2.10 Spent Fuel Pool Release Fractions

The SFP release fractions used in this analysis is based on the results of the SFPS for Group 1 as well as previous SFP studies. Table 51 shows a comparison of the release fractions between the SFPS and previous studies that demonstrates that cesium release fractions are generally less in the SFPS when compared to previous studies, and the timing of the release is generally longer.

The range of release fractions for this analysis is shown in Table 52. The Group 1 high SFP loading release fractions are based on the high-density cases in the SFPS with the low estimate representing cases where the reactor building remains intact, the base case reflects cases with significant air oxidation as a result of substantial damage to the refueling bay, and the high estimate represents a bounding case with large scale damage and relocation of the spent fuel assemblies and subsequent interaction of the fuel debris with the concrete floor. The Group 1 low SFP loading release fractions represent the low-density cases from the SFPS. For the other groups, the range of release fractions is consistent with past studies, but the high estimate is 90 percent based on insights from the SFPS regarding the molten core concrete interaction sensitivity study. The low SFP loading release fractions in Groups 2, 3, & 4 are assumed the same as in Group 1 since the releases are dominated by the recently discharged fuel.

Table 51 Comparison of Release Fractions from Current and Previous Spent Fuel Pool Analyses

Resolution of GI-82: NUREG-1353 (Ref. C.12), NUREG/CR-4982 (Ref. C.29), NUREG/CR-5281 (Ref. C.20)	NUREG-1738 (Ref. C.10)	Spent Fuel Pool Study (Ref. C.7)
• 10 to 100% cesium release (100% assumed for cases 1 and 2) • Release over 8 hours for a propagating SFP zirconium fire (assumed) • 0.25 (BWR) or 1.0 (PWR) conditional probability if fuel becomes uncovered	• 75% cesium release (assumed from NUREG-1465 (Ref. C.30) • Instantaneous draindown for large seismic event • 2 to 14 hour heatup depending on fuel age (see Ref. C.10, Table A1-1)	• Less than 1% to 49% cesium release • Draindown to uncovery ranges from 2.5 to 43 hours (when leak exists) • Start of release ranges between 8 hours to greater than 72 hours

Table 52 Estimated Cumulative Cesium Inventory Release Fraction Given a Spent Fuel Pool Fire

SFP Group	SFP loading	Low Est.	Base Case	High Est.
Group 1	High-density	3%	40%	90%
	Low-density	0.5%	3%	5%
Group 2, 3 & 4	High-density	10%	75%	90%
	Low-density	0.5%	3%	5%

C.2.11 Atmospheric Modeling and Meteorology

The atmospheric transport and dispersion model used in this analysis are based on the Peach Bottom MACCS2 results described in Section 7.1.2 of the SFPS (Ref. C.7), which uses a straight-line Gaussian plume segment dispersion model. As described in this study, the atmospheric release of radionuclides is discretized into (at longest) 1-hour plume segments. This accounts for variations in the release rate, as well as for changes in wind direction. More plume segments increase the resolution of the dispersion modeling to the point the resolution corresponds to the time resolution of the weather data, because each segment can travel in a compass direction representative of the actual weather data at the time the plume segment is released.

Two important parameters and variables required to model a SFP site are 1) the population density and distribution and 2) the site meteorology. The radionuclide inventory, source term (i.e., release fraction, release start time, and release duration), initial plume dimensions (related to the system geometry), and plume heat content were described.

C.2.12 Population and Economic Data

Population densities and distributions characteristics for SFP sites are examined to provide perspective on site demographic characteristics important to this cost-benefit analysis. Based on the review performed, site population densities near SFPs have the following statistical characteristics:

Table 53 Population Density within a 50 Mile Radius of U.S. Nuclear Power Plant Sites

Case	Statistical Parameter	Average Population Density within 50 miles (No. of people per square mile)	Representative Site Demographics
High estimate	90th percentile	722	Peach Bottom
Mean estimate	Mean	303	Surry
Median estimate	Median	183	Palisades
Low estimate	20th percentile	102	Point Beach

Source: 2010 census. Population density calculations do not correct the area within the radius that is water

Representative site demographics were selected to represent the 90th percentile, the mean, the median, and the 20th percentiles. For each representative site, the site population and economic data was created for 16 compass sectors and then interpolated onto a 64 compass-sector grid for better spatial resolution for the consequence analysis. Site population data is projected to the year 2011 using the latest version of the computer code SECPOP2000 (Ref. C.31). SECPOP2000 uses 2000 census data and applies a multiplier to account for population growth and an economic multiplier to account for the value of the dollar to create site data for the MELCOR Accident Consequence Code System (MACCS2). A multiplier value of 1.1051 from the U.S. Census Bureau was used to account for the average population growth in the U.S. from 2000 to 2011. Consistent with the approach used in the SFPS, the economic values from the database in SECPOP2000 (which uses an economic database based on the year 2002) were scaled to account for price escalation between the years 2002 and 2011. A scaling factor of 1.250 was derived based on the Consumer Price Index.

<u>Population Demographic Sensitivity</u>

The base case and the three additional site population densities and distributions near spent fuel pool locations discussed above were used as additional inputs into the MACCS2 calculations. Although the results provided in Appendix section C.2.12 provides insight into the analysis sensitivity to site population demographics in the United States, the results are not representative of any specific site because site specific meteorology for these additional sites is not used.

Table 54 Sensitivity of Public Health (Accident) Base Case Results to Population Demographics within 50 Miles

SFP Group	Site Population	Dose (averted person-rem per pool)	Benefits (2012 million dollars)		
			2% NPV	3% NPV	7% NPV
1	Low	469	$0.73	$0.65	$0.44
	Median	1097	$1.71	$1.53	$1.02
	Average (base case)	1739	$2.72	$2.42	$1.62
	High	2172	$3.39	$3.03	$2.02
2	Low	652	$0.98	$0.86	$0.55
	Median	1421	$2.14	$1.88	$1.20
	Average (base case)	2109	$3.18	$2.79	$1.79
	High	2684	$4.04	$3.55	$2.27
3	Low	1046	$1.09	$0.82	$0.34
	Median	2360	$2.46	$1.86	$0.78
	Average (base case)	3616	$3.77	$2.85	$1.19
	High	4560	$4.75	$3.59	$1.50
4	Low	751	$1.16	$1.03	$0.68
	Median	1586	$2.46	$2.18	$1.44
	Average (base case)	2284	$3.54	$3.14	$2.08
	High	2933	$4.54	$4.03	$2.67

Variations in population densities given the underlying assumptions stated above have the following net change on the averted public health (accident) attribute as summarized in Table 55.

Table 55 Net Percent Change in Public Health (Accident) Base Case Results for Variations in Population Densities within 50 Miles

Site Population Case	Statistical Parameter	Average Population Density within 50 miles (No. of people per square mile)	Net Percent Change in Public Health (Accident) Base Case (within 50 miles)
High estimate	90^{th} percentile	722	25% – 28%increase
Mean estimate	Mean	303	No change
Median estimate	Median	183	21% - 37% decrease
Low estimate	20^{th} percentile	102	67% - 73% decrease

Because a spent fuel pool fire could result in impacts to public health that extend beyond 50 miles, this case evaluates the sensitivity of averted public health exposures extending beyond 50 miles from the site, using the base case assumptions and the standard and sensitivity value for the person-rem conversion factor. Table 56 shows the sensitivity on public health (accident) benefits of extending the consequence analysis beyond 50 miles for the base case.

Table 56 Sensitivity of Public Health (Accident) Benefits for Expedited Transfer Alternative–Low-density Spent Fuel Pool Storage extending beyond 50 miles (Base case with $2,000 and $4,000 per person-rem)

SFP Group	Case	Dose conversion factor ($/person-rem)	Dose (averted person-rem per pool	Benefits (2012 million dollars)		
				2% NPV	3% NPV	7% NPV
1	Alternative 2 - Low-density storage	$2,000	11,120	$17.37	$15.49	$10.37
		$4,000		$34.73	$30.98	$20.73
2	Alternative 2 - Low-density storage	$2,000	13,680	$20.61	$18.10	$11.58
		$4,000		$41.22	$36.21	$23.17
3	Alternative 2 - Low-density storage	$2,000	22,730	$23.67	$17.90	$7.47
		$4,000		$47.33	$35.80	$14.94
4	Alternative 2 - Low-density storage	$2,000	15,880	$24.57	$21.83	$14.44
		$4,000		$49.14	$43.66	$28.88

Sensitivity of Offsite Property Cost Offset Results to Population Demographics

Certain metrics such as property use, the number of displaced individuals (either temporarily or permanently), and the extent to which such actions may be needed are affected by the population size and the amount of economic activity in the vicinity of the postulated accident.

This section provides a basis for understanding the nature and the extent of the relationship between population densities, distributions characteristics, and property values near spent fuel pool sites. This examination provides a perspective on how important changes to these site demographic variables are for this regulatory analysis. The base case and the three additional site population densities, distributions, and economic characteristics near spent fuel pool locations are discussed above. These population and economic characteristics were used as additional inputs into the MACCS2 calculations that otherwise still used the SFPS reference plant specific values. Although the results provided in Table 57 provide insight into the analysis sensitivity to site population demographics in the U.S., the results are not representative of any specific site because site specific meteorology for these additional sites is not used. These measures are also subject to large uncertainties, as it is difficult to model the impact of disruptions to many different aspects of local economies, the loss of infrastructure on the general U.S. economy, or the details of how long-term protective actions would be performed.

Table 57 Sensitivity of Offsite Property Cost Offset Results to Population Demographics within 50 Miles (Base Case using EPA Intermediate PAG Criterion)

SFP Group	Site Population	Offsite Property Cost Offsets (2012 million dollars)		
		2% NPV	3% NPV	7% NPV
1	Low	$1.29	$1.15	$0.77
	Median	$4.19	$3.73	$2.50
	Average (base case)	$7.65	$6.83	$4.57
	High	$12.55	$11.19	$7.49
2	Low	$2.04	$1.79	$1.14
	Median	$6.75	$5.93	$3.79
	Average (base case)	$11.50	$10.10	$6.46
	High	$13.43	$11.80	$7.55
3	Low	$2.09	$1.58	$0.66
	Median	$6.84	$5.18	$2.16
	Average (base case)	$12.07	$9.13	$3.81
	High	$17.08	$12.91	$5.39
4	Low	$2.60	$2.31	$1.53
	Median	$8.69	$7.72	$5.11
	Average (base case)	$14.35	$12.75	$8.44
	High	$16.14	$14.34	$9.48

Because a spent fuel pool fire under certain scenarios and environmental conditions could result in impacts to offsite property located beyond 50 miles from the postulated accident site, this case evaluates the sensitivity of offsite property cost offsets for damages occurring beyond 50 miles from the site, using the base case assumptions and the intermediate EPA PAG criterion. Table 58 shows the sensitivity on offsite property cost offsets of extending the consequence analysis beyond 50 miles for the base case.

Table 58 Sensitivity of Offsite Property Cost Offset Results to Consequences beyond 50 Miles (Base Case using EPA Intermediate PAG Criterion)

SFP Group	Case	Offsite Property Cost Offsets (2012 million dollars)			
		2% NPV	3% NPV	7% NPV	% increase
1	Base case - within 50 miles	$8.96	$7.99	$5.35	
	Sensitivity - beyond 50 miles	$16.36	$14.59	$9.76	83%
2	Base case - within 50 miles	$9.03	$7.93	$5.08	
	Sensitivity - beyond 50 miles	$28.79	$25.29	$16.18	219%
3	Base case - within 50 miles	$11.45	$8.66	$3.61	
	Sensitivity - beyond 50 miles	$27.17	$20.55	$8.57	137%
4	Base case - within 50 miles	$9.81	$8.71	$5.76	
	Sensitivity - beyond 50 miles	$39.62	$35.20	$23.29	304%

C.2.13 Long-Term Habitability Criteria

The long-term phase is the period following the 7-day emergency phase and is modeled for 50 years to calculate consequences from exposure of the average person. Radiation exposure during this phase is mainly from external radiation from trace contaminants that remain after the land is decontaminated, or in lightly contaminated areas where no decontamination was required. Internal radiation exposures may also occur during this period, including inhalation of resuspended radionuclides and ingestion of food and water with trace contaminants. Depending on the relevant protective action guides (PAGs) and the level of radiation, food, and water below a certain limit could be considered adequately safe for ingestion, and lightly contaminated areas could be considered habitable.

A long-term cleanup policy for recovery after a severe nuclear power plant accident does not currently exist. The actual decisions regarding how land would be recovered and populations relocated after an accident would be made by a number of local, State, and Federal jurisdictions and would most likely be based on a long-term cleanup strategy, which is currently being developed by the NRC, U.S. Environmental Protection Agency (EPA), and other Federal agencies. Furthermore, a cleanup standard may not have an explicit dose level for cleanup. Instead, the cleanup strategy may give local jurisdictions the ability to develop localized cleanup goals after an accident, to allow for a number of factors that include sociopolitical, technical, and economic considerations.

Site-specific values are used to determine long-term habitability. For habitability, most States adhere to EPA intermediate phase protective action guides that allow a dose of 2 rem in the first year and 500 mrem each year thereafter (Ref. C.32). This habitability criterion was used in previous SFP studies, which used 4 rem in 5 years to represent these PAG levels (e.g., 2 rem in year one, followed by 0.5 rem each successive year). The nationally and internationally recommended upper bound for dose in a single year from man-made sources, excluding medical radiation, is 500 mrem per year to the whole body of individuals in the general population. The EPA states “these recommendations were not developed for nuclear incidents … [and] also not appropriate for chronic exposure” (Ref. C.32, p. E-12). However, some States, such as the State of Pennsylvania, has adopted a habitability criterion of 500 mrem beginning in the first year (and each following year) as determined by the Pennsylvania Code Title 25 Section 219.51 (Ref. C.33). The use of this long-term habitability criterion reduces the predicted long-term population doses and health effects and increases the costs associated with interdiction, decontamination, and condemnation.[29]

Given the uncertainties in which long-term habitability criterion would be used, Table 60 provides the long-term phase habitability criterion used in this analysis to analyze the consequences of SFP accidents on public health (accident).

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[29] Interdiction and condemnation refer to the relocation of people from contaminated areas according to the habitability criterion. Interdiction is the temporary relocation of the affected population while decontamination, natural weathering, and radioactive decay reduce the contamination levels. Condemnation is the permanent relocation of the affected population if decontamination, natural weathering, and radioactive decay cannot adequately reduce contamination levels to habitability limits within 30 years.

Table 59 Long-Term Habitability Criterion

Case[30]	Long-Term Habitability Criterion	Protective Action Basis
Low Estimate	500 mrem annually	Pennsylvania dose limit to the public
Base Case	2 rem in the first year and 500 mrem each year thereafter	EPA intermediate phase PAGs
High Estimate	2 rem annually	EPA intermediate phase PAG: first year

MACCS2 computer runs were run for each of the protective action levels listed in Table 59 to calculate averted dose and offsite property damage using the representative plant site demographics listed in Table 53.

Different habitability criteria given the underlying assumptions stated above has the following net change on the averted public health (accident) attribute as summarized in Table 60.

Table 60: Sensitivity of Public Health (Accident) Benefits to Habitability Criteria (within 50 Miles)

SFP Group	Habitability Criteria	Dose (averted person-rem per pool)	Benefits (2012 million dollars)		
			2% NPV	3% NPV	7% NPV
1	Low (500 mrem annually)	770	$1.21	$1.08	$0.72
	Base Case (4rem / 5years)	1,740	$2.72	$2.42	$1.62
	High (2 rem annually)	1,980	$3.09	$2.75	$1.84
2	Low (500 mrem annually)	900	$1.36	$1.20	$0.77
	Base Case (4rem / 5years)	1,630	$2.45	$2.15	$1.38
	High (2 rem annually)	2,480	$3.74	$3.29	$2.10
3	Low (500 mrem annually)	1,580	$1.64	$1.24	$0.52
	Base Case (4rem / 5years)	3,020	$3.14	$2.37	$0.99
	High (2 rem annually)	4,180	$4.36	$3.29	$1.37
4	Low (500 mrem annually)	960	$1.49	$1.33	$0.88
	Base Case (4rem / 5years)	1,690	$2.62	$2.33	$1.54
	High (2 rem annually)	2,730	$4.23	$3.76	$2.49

The use of these habitability criteria also affects the values of offsite property damage used in this analysis. Certain metrics such as offsite property damage, the number of displaced individuals (either temporarily or permanently) and the extents to which such actions may be needed are inversely proportional to changes in collective dose resulting from changes in habitability criteria.

30 Cases are defined as low and high estimate based on the effect that different long-term habitability criteria have on averted radiation exposure.

These criteria provide a benchmark for understanding the nature and the extent of the relationship between collective dose, economic consequences, and habitability criteria following a severe SFP accident. These measures are subject to large uncertainties, as it is difficult to model the impact of disruptions to many different aspects of local economies, the loss of infrastructure on the general U.S. economy, or the details of how long-term protective actions would be performed.

Table 61 Net Percent Change in Public Health (Accident) Base Case Results for Variations in Population Densities within 50 Miles

Habitability Criterion Case	Habitability Criterion	Net Percent Change in Public Health (Accident) Base Case (within 50 miles)
High estimate	2 rem annually	14% – 20% increase
Base case	2 rem first year, 500 mrem thereafter (4 rem / 5 years)	No change
Low estimate	500 mrem annually	56% – 58% decrease

Offsite Property Costs Sensitivity to Habitability Criteria

A long-term cleanup policy for recovery after a severe nuclear power plant accident does not currently exist. The actual decisions regarding how land would be recovered and populations relocated after an accident would be made by a number of local, State, and Federal jurisdictions and would most likely be based on a long-term cleanup strategy, which is currently being developed by the NRC, EPA, and other Federal agencies. Furthermore, a cleanup standard may not have an explicit dose level for cleanup. Instead, the cleanup strategy may give local jurisdictions the ability to develop localized cleanup goals after an accident, to allow for a number of factors that include sociopolitical, technical, and economic considerations. Given the uncertainties in which long-term habitability criterion would be used, Table 62 provides a low and high value for the long-term phase habitability criterion for use in a sensitivity analysis to analyze the effect on the costs for offsite property damage.

Table 62: Sensitivity of Offsite Property Damage Cost Offsets within 50 Miles to Different Habitability Criteria

SFP Group	Habitability Criteria	Offsite Property Cost Offsets (2012 million dollars)		
		2% NPV	3% NPV	7% NPV
1	Low Est. (500 mrem annually)	$12.83	$11.44	$7.66
	Base Case (4rem / 5years)	$7.65	$6.83	$4.57
	High Est. (2 rem annually)	$7.19	$6.41	$4.29
2	Low Est. (500 mrem annually)	$16.56	$14.54	$9.31
	Base Case (4rem / 5years)	$11.50	$10.10	$6.46
	High Est. (2 rem annually)	$11.10	$9.75	$6.24
3	Low Est. (500 mrem annually)	$18.71	$14.15	$5.90
	Base Case (4rem / 5years)	$12.07	$9.13	$3.81
	High Est. (2 rem annually)	$11.50	$8.70	$3.63
4	Low Est. (500 mrem annually)	$19.28	$17.13	$11.33
	Base Case (4rem / 5years)	$14.35	$12.75	$8.44
	High Est. (2 rem annually)	$14.02	$12.45	$8.24

This sensitivity analysis uses three protective action levels—the Pennsylvania PAG of 500 mrem annually for the low estimate, the EPA intermediate phase PAG level of 2 rem in the first year, and 500 mrem annually thereafter for the base case, and 2 rem annually for the high estimate—to evaluate post-accident collective dose and offsite property costs. As discussed in Appendix section C.2.12, offsite property costs are inversely proportional to changes in collective dose resulting from changes in habitability criteria (i.e., lower PAG guidelines result in lower collective dose value and higher offsite property costs). These results show the cost offsets increase by up to 67 percent (7 percent net present value) than those in the Group 1 base case result when the 500 mrem annual limit is used. Conversely, offsite property damage cost offsets decrease by up to 6 percent (7 percent net present value) than those in the Group 1 base case result when the 2 rem annual limit is used.

C.2.14 Emergency Response Modeling

This cost-benefit analysis uses the emergency response model contained in the Reference Plant-specific MACCS2 results described in Section 7.1.2 and Appendix A of the SFPS. The extended loss of ac power is assumed to be limited to the plume exposure pathway emergency planning zone (EPZ) (approximately 16 kilometers or 10 miles) because of the assumption that the strength of the seismic event is from the proximity of the seismic event to the site, rather than being a wider impact from a larger magnitude. See Section 7.1.4 of the SFPS for additional details.

A summary of the evacuation timing and speeds for each cohort modeled in the SFPS and reproduced here is provided in Table 63. This evacuation timing and speeds is used to produce the consequence analyses results for this analysis.

Table 63 Evacuation Model 1: Plume Exposure Pathway EPZ Evacuation

Population			Response Delays (hours)				Phase Duration (hours)		Evacuation Travel Speeds (mph)		
Cohort		Population Fraction	Siren (OALARM)	Delay to Shelter	Delay to Evacuation	Total (Depart time)	Early (DURBEG)	Middle (DURMID)	Early (ESPEED)	Middle (ESPEED)	Late (ESPEED)
1	0 to 10 miles Early Evacuees	0.3	1	0	0	1	1	0.5	20	15	5
	10 to 20 miles Shadow			2	1	4					
2	0 to 10 miles General Public	0.417	1	1	1	3	0.25	3	5	2	20
3	0 to 10 miles Special Facilities	0.006	1	0	4	5	0.5	0.5	2	15	20
4	0 to 10 miles Evacuation Tail	0.1	1	2	3	6	0.5	0.5	2	15	20
5	0 to 10 miles Schools	0.172	1	0	0.5	1.5	1	0.5	20	15	20
6	0 to 10 miles Nonevacuating Public	0.005	1	-	-	-	-	-	-	-	-

Meteorological data used to calculate offsite consequences for this analysis consisted of 1 year of hourly meteorological data (8,760 data points for each meteorological parameter) for the Peach Bottom site evaluated in the SFPS (Ref. C.7) and in NUREG-1935 (Ref. C.34). The Peach Bottom site provided 2 years of weather data, including directly measured hourly

precipitation data. Stability class data were derived from temperature measurements at two elevations on the site meteorological towers. The specific year of meteorological data chosen for the Peach Bottom site was 2006, which was based on data recovery (greater than 99 percent being desirable) as documented in NUREG/CR-7009 (Ref. C.35). Different trends (e.g., wind rose pattern and hours of precipitation) between the years were estimated to have a relatively minor (less than 25 percent) effect on the results. More specific details of the weather data can be found in NUREG/CR-7009.

The wind rose shown in Figure 15 shows the Peach Bottom site wind direction (direction the wind blows toward) data that were used in the consequence analyses for this analysis. The wind rose in the figure below suggests that the predominant wind direction is to the south and east and a secondary direction in terms of likelihood is to the northwest to north.

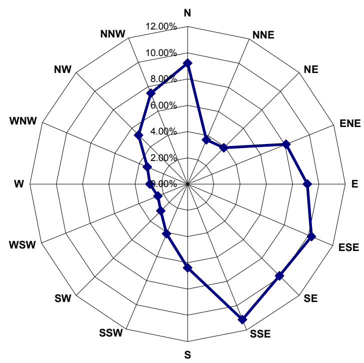

Figure 15 Reference plant wind rose
Source: SFPS (Ref. C.7, p. A-3)

Although using a single plant's emergency response modeling and consequence analyses introduces uncertainty, the conditional individual risk measures near the site are expected to be relatively insensitive to site-specific characteristics (i.e., emergency response measures). This is because the relatively delayed and prolonged releases as predicted by the SFPS and the lack of short-lived radionuclides allow time for effective protective actions, in both the early and long term phases, to limit exposures to the public particularly in the event of large releases. This is consistent with previous studies in which individual early and latent fatality risks were projected to be low. Therefore, the resulting individual risk measures near the site can used for comparisons to the quantitative health objectives represent risk to the average individual within 1.6 and 16 kilometers (1 and 10 miles) of the plant.

C.2.15 Uniform Fuel Pattern during an Outage Sensitivity

The base case of this regulatory analysis assumes that each licensee has prearranged the spent fuel pool such that discharged assemblies can be placed directly into a 1x4 arrangement for the discharges of the last two outages. This approach is consistent with the requirements discussed in Section 9.3 of the SFPS. However, those requirements do allow for the fuel to be stored in a less favorable configuration for some time following discharge if other considerations prevent prearrangement. To capture the effects of nonbeneficial arrangement of discharged fuel, this regulatory analysis evaluates the situation in which the discharged spent fuel is uniformly arranged during the outage to evaluate the effect of this aspect on the public health (accident) attribute.

For the offsite consequence analysis, the sequences with recently discharged fuel in a uniform configuration were binned in a similar manner to the low-density and high-density (1x4) loading scenarios. Because licensees are required to move their recently discharged fuel to a more favorable configuration after a certain amount of time, this sensitivity assumes that the high-density uniform case becomes identical to the high-density (1x4) case by the end of operating cycle phase 2 (OCP 2) or within 25 days.

Table 64 provides a comparison of the effect on the public health (accident) attribute if a plant operator initially places discharged spent fuel in a uniform pattern and achieves the 1x4 pattern by the end of OCP 2 (i.e., within 25 days) versus placing the fuel directly into the 1x4 pattern.

Table 64: Sensitivity of Public Health (Accident) Benefits (within 50 Miles) to Initial Loading Pattern of Discharged Fuel

SFP Group	Initial Loading Pattern of Discharged Fuel	Dose (averted person-rem per pool)	Benefits (2012 million dollars)		
			2% NPV	3% NPV	7% NPV
1	Base Case - 1x4	1,740	$2.72	$2.42	$1.62
	Uniform fuel pattern	2,040	$3.18	$2.84	$1.90
2	Base Case - 1x4	1,630	$2.45	$2.15	$1.38
	Uniform fuel pattern	1,840	$2.77	$2.44	$1.56
3	Base Case - 1x4	3,020	$3.14	$2.37	$0.99
	Uniform fuel pattern	3,310	$3.45	$2.61	$1.09
4	Base Case - 1x4	1,690	$2.62	$2.33	$1.54
	Uniform fuel pattern	1,980	$3.07	$2.73	$1.80

The placement of the discharged fuel directly into a 1x4 pattern reduces the estimated averted dose within 50 miles of the site between 10 percent and 17 percent discounted at 7 percent compared to the cases when achieving this fuel pattern is delayed for up to 25 days at the end of OCP 2. These effects are bounded by the assumption of the unavailability of natural circulation air cooling for the base case and high estimate.

Offsite Property Cost Offset Sensitivity

Table 65 provides a comparison of the effect on the offsite property cost offsets if a plant operator initially places discharged spent fuel in a uniform pattern and achieves the 1x4 pattern by the end of OCP 2 (i.e., within 25 days) versus placing the fuel directly into the 1x4 pattern.

Table 65 Sensitivity of Offsite Property Cost Offsets within 50 Miles to Initial Loading Pattern of Discharged Fuel

SFP Group	Initial Loading Pattern of Discharged Fuel	Offsite Property Cost Offsets (2012 million dollars)		
		2% NPV	3% NPV	7% NPV
1	Base Case - 1x4	8.96	7.99	5.35
	Uniform fuel pattern	9.86	8.80	5.89
2	Base Case - 1x4	9.03	7.93	5.08
	Uniform fuel pattern	14.82	13.02	8.33
3	Base Case - 1x4	11.45	8.66	3.61
	Uniform fuel pattern	15.56	11.77	4.91
4	Base Case - 1x4	9.81	8.71	5.76
	Uniform fuel pattern	18.50	16.44	10.87

C.3 Implementation Assumptions

C.3.1 Dry Storage Occupational Exposure (Routine)

Routine occupational exposure associated with dry storage of spent fuel includes worker dose associated with additional DSC loading, unloading and handling activities; additional ISFSI operations, maintenance, and surveillance activities; additional DSC storage at an ISFSI; and additional transportation cask loading, unloading, and handling activities.

Worker dose associated with DSC loading operations vary depending upon the cask technology being loaded, the characteristics of the fuel being loaded (e.g., fuel age and burnup), and fuel loading patterns in the DSC (e.g., the location of short-cooled, high burnup spent fuel or colder spent fuel within DSC baskets using regional loading). For the regulatory baseline, a worker dose of 400 person-mrem per DSC loaded was assumed. This radiation dose is consistent with the exposure value used in EPRI TR-1021049 (Ref. C.36) and in EPRI TR-1018058 (Ref. C.37), which analyzed worker impacts associated with loading spent fuel for transport to the proposed Yucca Mountain repository. Some sites achieve per package dose ranges in the range of 200 to 300 person-mrem per package loaded, while other sites experience higher per package dose rates. For the low-density storage case, each cask loaded in addition to the number required by the regulatory baseline is estimated to result in an incremental 400 person-mrem dose.

There is routine occupational dose associated with ISFSI annual operation and maintenance activities (i.e., inspection, surveillance, and security operations). The regulatory baseline assumes an annual dose of 120 person-mrem per site per year for inspection, surveillance, and security activities and 1,500 person-mrem per site per year for ISFSI operations and maintenance. These estimated radiation doses are consistent with assumptions used by EPRI in EPRI TR-1021049 (Ref. C.36) and TR-1018058 (Ref. C.37). Because additional shielding is assumed to be provided by concrete overpacks, the worker dose associated with ISFSI operations and maintenance is not expected to increase. Therefore, no incremental occupational dose is predicted for performing annual ISFSI operation and maintenance.

There is routine occupational dose associated with the storage of each DSC at an operational ISFSI. The regulatory baseline assumes a worker dose of 170 person-mrem for each additional DSC loaded at an ISFSI site. This estimated radiation dose is consistent with assumptions used by EPRI in EPRI TR-1021049 (Ref. C.36) and TR-1018058 (Ref. C.37). Because additional shielding is assumed to be provided by concrete overpacks, the worker dose associated with each DSC stored at an operational ISFSI is not expected to increase. For the low-density SFP storage case, each cask stored in addition to the number required by the regulatory baseline is estimated to result in an incremental 170 person-mrem dose.

Table 66 summarizes the occupational dose estimates for each activity.

Table 66 Incremental Occupational Dose (Routine) Estimates

Activity	Incremental Occupational Dose (Routine) (person-mrem per activity)
Load a DSC	400
ISFSI Operation and maintenance	0
Loading a DSC at an ISFSI	170
Total	570

C.3.2 Number of Dry Storage Casks

In 2013, the representative Group 1 plant has 3,055 fuel assemblies stored in the SFP in a high-density 1x4 loading configuration. During each refueling outage, 284 assemblies are offloaded from the reactor vessel to the SFP. For the regulatory baseline, the plant is expected to load the required number of DSCs with a 68-assembly capacity each refueling outage to retain sufficient space in the SFP to discharge one full core of fuel (full core reserve). The estimated DSC inventory is shown in Figure 16.

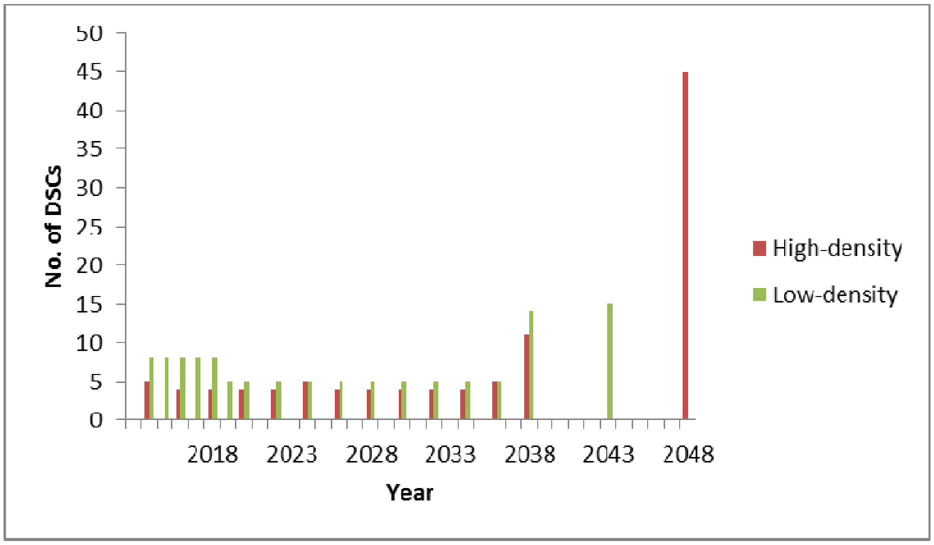

Figure 16 Timing of dry storage cask loading for the representative Group 1 plant

At the expiration of the operating license in 2038, the full core is offloaded into the SFP. The analysis further assumes that the entire SFP inventory will be placed into dry storage by 2048, 10 years after termination of unit commercial operation.

For the low-density SFP storage case, it is assumed that there is an NRC policy decision that requires licensees to offload the spent fuel inventory to dry storage to obtain a low-density configuration within 5 years (e.g., by end of 2019). In this configuration, the representative Group 1 plant SFP stores 852 assemblies, which is equivalent to the discharge from the last three refueling outages. Using the same initial conditions as above, and using the DSC with a 57-assembly derated capacity beginning in year 2020, the inventory model is provided as the low-density chart in Figure 16.

At the expiration of the operating license in 2034, the full core is offloaded into the SFP. The analysis further assumes that the entire SFP inventory will be placed into dry storage by 2048. Additionally, in year 2048, the spent fuel has cooled for a sufficient length of time that the DSC is no longer derated.

Similar calculations were performed for Groups 2, 3 and 4 using the Holtec Hi-Storm FW DSC system for PWR spent fuel. The dry storage cask loading for the representative Group 2 plant is shown in Figure 17.

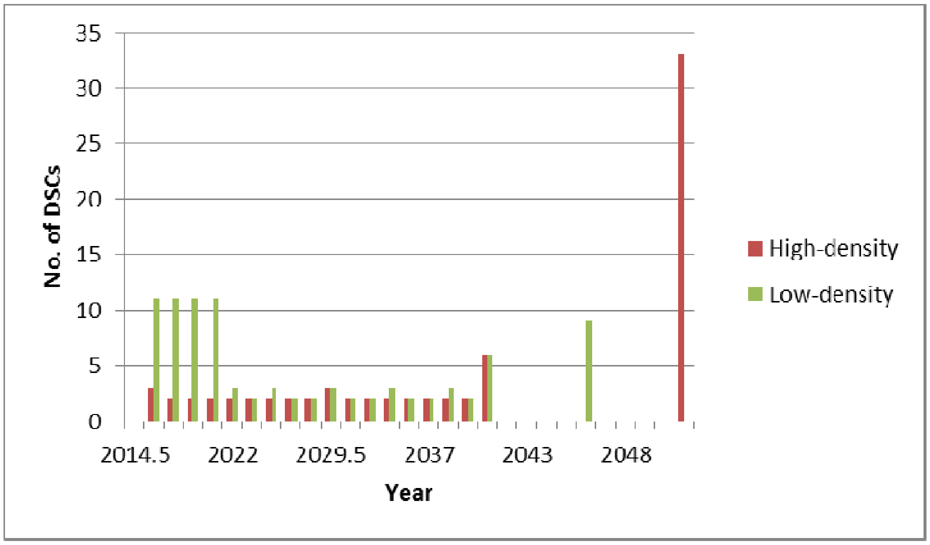

Figure 17 Timing of dry storage cask loading for the representative Group 2 plant

In 2018, the representative Group 3 plant is assumed to begin commercial operation. At this time, there is no spent fuel assemblies stored in the SFP. The unit is assumed to operate on an 18 month refueling cycle, discharging an estimated 69 assemblies per cycle (Ref. C.4, Section 9.1). For the regulatory baseline, the representative new nuclear plant is expected to begin dry storage in 2038 and will load a sufficient number of Holtec Hi-Storm FW casks to maintain its full core offload capability. The estimated timing for DSC loading is shown in Figure 18.

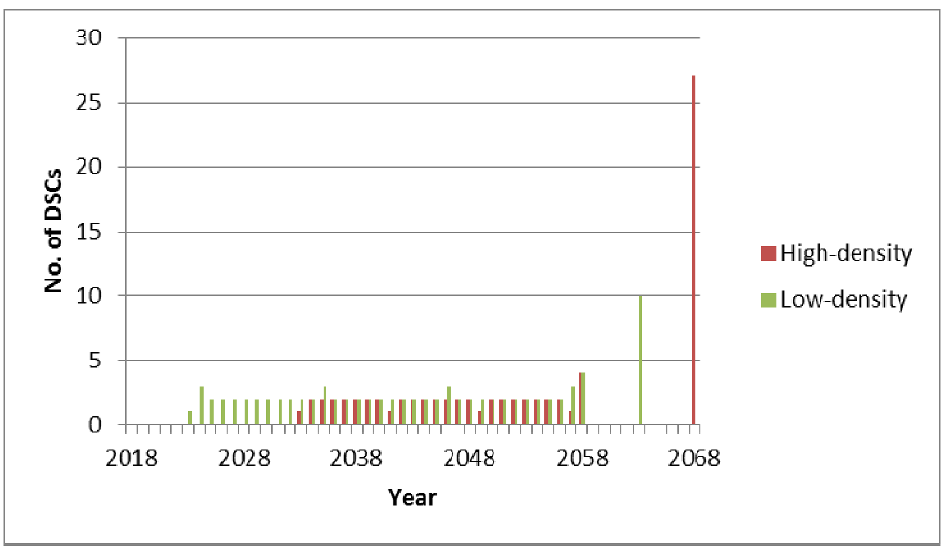

Figure 18 Timing of dry storage cask loading for the representative Group 3 plant

The representative Group 4 SFP which is shared between two PWR units is assumed to have 1,637 fuel assemblies stored in the SFP in a high-density 1x4 loading configuration. Each reactor unit operates on a 24-month refueling cycle and discharges 84 assemblies on a 1-year staggered cycle. The representative shared SFP has already implemented dry storage.

For the regulatory baseline, the Group 4 SFP is expected to load the required number of DSCs with a 37-assembly capacity each refueling outage to retain sufficient space in the SFP to discharge one full core of fuel (full core reserve). For the low-density case, the DSC has a 33-assembly capacity because of the higher heat load of the spent fuel. At the expiration of the operating license in 2038, the full core is offloaded into the SFP. The analysis further assumes that the entire SFP inventory will be placed into dry storage beginning in 2038 and completed by 2048. The estimated timing for DSC loading for the representative Group 4 SFP is shown in Figure 19.

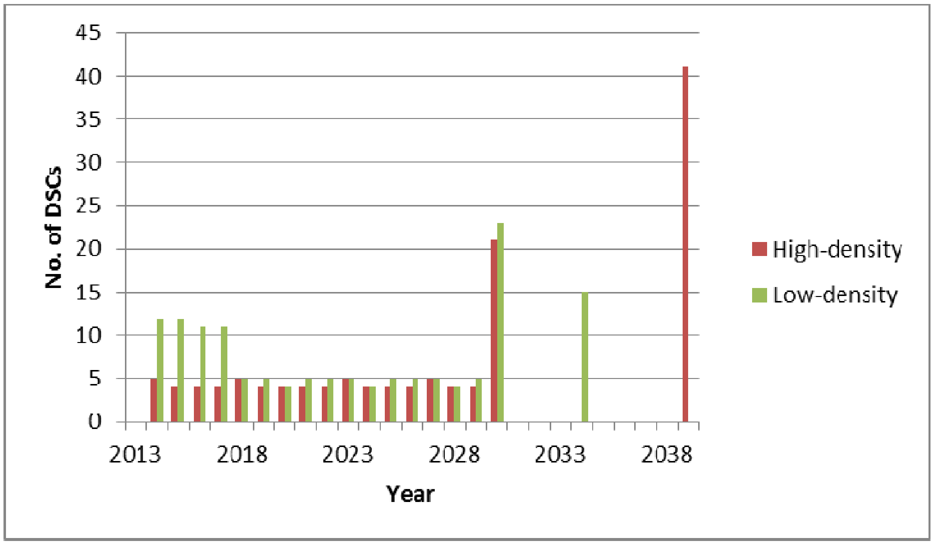

Figure 19 Timing of dry storage cask loading for the representative Group 4 plant

C.4 <u>Cost Assumptions</u>

C.4.1 Generic Costs

Costs presented in this analysis are based on estimates by the author or cited documents. This is a generic cost estimate and should be used accordingly. Site-specific features may result in higher or lower costs than those estimated.

C.4.2 Dry Storage Upfront Costs

Upfront costs include engineering, design, and licensing costs; equipment costs; construction costs; and start up and testing costs. Each of these cost components are further described in EPRI TR-1021048, "Industry Spent Fuel Storage Handbook" (Ref. C.38). As noted in EPRI TR-1025206, "Impacts Associated with Transfer of Spent Nuclear Fuel from Spent Fuel Storage Pools to Dry Storage after Five Years of Cooling, Revision 1" (Ref. C.5), the independent spent fuel storage installation (ISFSI) upfront costs vary widely from site to site and the upfront costs for those in operation vary from several million to tens of millions of dollars (Ref. C.5, p. 2-23). Values for upfront costs were estimated based on two publically available cost estimates that identified the specified number of DSC to be stored. The estimate amortized upfront costs for each site is provided in Table 67.

Table 67 Amortized DSC Upfront Costs

ISFSI Facility	Upfront Cost Estimate (base year)	Upfront Cost Estimate (2012 $)	DSC Storage Capacity	Attributed Upfront Cost per DSC (2012 $)
Monticello	$21.5 million (2005 $)	$25,275,400	30	$842,500
Pilgrim	$22 million (2006 $)	$25,055,800	53	$472,800
Average (Best Estimate)		$25,165,600		$657,700

C.4.3 Incremental Costs Associated with Earlier DSC Purchase and Loading

Incremental costs are the costs associated with the purchase and loading of DSCs on a periodic basis. These costs include the capital costs for the DSC and the loading costs for the storage systems. The unit cost estimates used in this analysis are provided in Table 68. These cost estimates are based on the DSC unit costs that EPRI used for a generic interim storage facility (Ref. C.39) and documented in EPRI TR-1025206 (Ref. C.5). Nuclear power plant licensees may experience incremental DSC purchase and loading costs that are higher or lower than the amount assumed in this cost-benefit analysis.

Table 68 Incremental Unit Cost Estimates

Item	Base Case Unit Cost (Constant $2012)	Adders to load 5-year cooled fuel (Constant $2012)	5-Year cooled fuel Unit Cost (Constant $2012)
Canister	$780,000	$62,400(1)	$842,400
Concrete overpack	$208,000	$41,600(2)	$249,600
Loading of canister-based storage	$312,000	$62,400	$374,400
Total	$1,300,000		$1,466,400

1. The canister cost adder is the product of $780,000 x 40% x 20%.
2. The concrete overpack adder is the sum of the labor adder and the concrete shielding adder (e.g., $208,000 x 40% x 20% + $208,000 x 30% x 40%).

When only five-year cooled, high burnup spent fuel is available for loading into dry storage, there are several potential cost adders to address increased fabrication costs, additional shielding capability in concrete storage overpacks; and higher loading costs because of increased worker dose and work rules that result in longer cask loading durations or the need to utilize additional crews.

Labor costs are approximately 40 percent of the cost of DSCs (Ref. C.5). Assuming that the labor portion of canister and concrete overpack cost increase by 20 percent, this results in a fabrication cost adder of $79,040 per DSC (e.g., 40 percent x $988,000 x 20 percent). This fabrication adder is applied to dry storage incremental costs when five-year cooled inventories are transferred to dry storage.

Concrete shielding costs are approximately 30 percent of the concrete overpack cost (Ref. C.5). Assuming that shielding costs increase by 40 percent, these results in a concrete overpack shielding cost adder of $24,960 per overpack ($208,000 x 30 percent x 40 percent). This shielding adder is applied to dry storage incremental costs when 5-year cooled inventories are transferred to dry storage.

There may be other additional costs associated with amending existing certificates of compliance (CoCs), certifying new designs, or may result from high demand for DSCs in short supply. These costs may be passed on to nuclear plant operators through the price of the DSC systems or may be directly billed to nuclear plant operators if the amended or new designs are specific only to that ISFSI. These additional costs were not estimated given the possibility for a wide range of costs for implementing CoC changes and the possible price swings, which could occur for DSCs if there is limited supply.

Because of the increased costs associated with increased worker dose, longer loading times to comply with work rules, and the need to load more DSCs, and the application of fatigue rules during cask loading operations, the NRC estimates that DSC loading costs increase by 20 percent. This loading cost adder of $62,400 per DSC (e.g., 20 percent times $312,000) is applied when 5-year cooled spent fuel assemblies are loaded into dry storage casks.

C.4.4 Incremental Annual ISFSI Operating Costs

Annual operating costs for an ISFSI during reactor operation include the costs associated with NRC inspections; security; radiation monitoring; ISFSI operational monitoring; technical specification and regulatory compliance, including implementation of new CoC amendments; personnel cost and code maintenance associated with fuel selection for dry storage; personnel costs for spent fuel management and fabrication surveillance activities; electric power usage for lighting and security systems; road maintenance to the ISFSI site; and miscellaneous expenses associated with ISFSI maintenance. NRC license fees for dry storage are included as part of the 10 CFR Part 50 (“Domestic Licensing of Production and Utilization Facilities”) operating license fees and, therefore, are not an incremental cost.

Because most operating nuclear power plants have already implemented dry storage, no incremental annual ISFSI operating costs to implement dry storage at an earlier date is estimated for Group 1, 2, or 4 SFP sites if a policy decision is made to accelerate the transfer of spent fuel stored in SFPs to dry storage.

For the Group 3 SFPs for which the associated reactor is not expected to begin commercial operation until 2018, the NRC estimates that the site would begin transferring fuel to dry storage in 2040. For the expedited transfer alternative, it is expected that the unit would begin transferring fuel to dry storage in 2025 and, therefore, Group 3 sites would incur incremental annual ISFSI operating cost for the earlier ISFSI operating period from 2025 to 2040. EPRI reports a wide variability in published estimates of annual ISFSI operating costs that range from $212,000 to $2 million per year in 2012 dollars and reported their estimate of $1.1 million per year for an ISFSI at an operating nuclear power plant site (Ref. C.5, p. 2–28). This estimate provided in Table 69 is used as the incremental annual Group 3 ISFSI operating cost in this analysis. ISFSIs located at nuclear operating plant sites may experience annual ISFSI operating costs that are higher or lower than this estimated value.

Table 69 Incremental ISFSI Annual Operating Costs

SFP Group	Activity	Incremental ISFSI Annual Operating Cost (2012 dollars)
All	ISFSI operation and maintenance costs	Negligible
3	Early ISFSI operation and maintenance costs	$1,100,00

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C.37 EPRI TR-1018058, "Occupational Risk Consequences of the Department of Energy's Approach to Repository Design, Performance Assessment, and Operation in the Yucca Mountain License Application," dated August 2008.

C.38 EPRI TR-1021048, "Industry Spent Fuel Storage Handbook," dated July 2010.

C.39 EPRI TR-1018722, “Cost Estimate for an Away-From-Reactor Generic Interim Storage Facility (GISF) for Spent Nuclear Fuel,” dated May 2009.

APPENDIX D: SENSITIVITY ANALYSIS INFORMATION

D.1 Present Value Calculations

The choice of a discount rate, over long periods of time, raises questions of science, economics, philosophy, and law. Although the discount rate has a large influence on the current value of future damages, there is no consensus about what rates to use in this context.

The NRC traditionally uses constant discount rates of 7 percent for regulatory decisionmaking and 3 percent as a sensitivity value to reflect reliance on a social rate of time preference discounting concept in accordance with OMB Circular A-4. As Circular A-4 acknowledges, however, the choice of discount rate for intergenerational problems raises distinctive problems and presents considerable challenges. After reviewing those challenges, Circular A-4 states, "If your rule will have important intergenerational benefits or costs you might consider a further sensitivity analysis using a lower but positive discount rate in addition to calculating net benefits using discount rates of 3 and 7 percent."

The 3 percent rate is consistent with estimates provided in the economics literature and approximates the real rate of return on long-term government debt which serves as a proxy for the real rate of return on savings. A low discount rate value of 2.0 percent is included, which represents the lower bound for the certainty-equivalency rate in 100 years using the random walk model approach (Ref. D.1) to address the concern that interest rates are highly uncertain over time.

D.2 Dollar per Person-Rem Conversion Factor

The NRC is currently revising the dollar per person-rem averted conversion factor based on recent information regarding the value of a statistical life (VSL). However, until the NRC completes the update and publishes the appropriate guidance documents, the NRC will perform sensitivity analysis to estimate the impact on the calculated results when more current VSL and cancer risk factor are used. The NRC used the EPA's VSL as an interim value in the sensitivity analysis. The EPA's VSL was developed through a rigorous process, reviewing many published academic papers, and includes review from the Scientific Advisory Board, an independent review board.

The EPA's VSL in 2009 dollars is approximately $7.2 million (Ref. D.2, p. 41). The VSL is derived from "using a mixed effects model (random intercept with fixed effects for study characteristics), the authors regressed the VSL estimates on average income, probability of death, and several study design variables" (Ref. D.2, p. 41). Therefore, using the CPI-U based inflator to adjust from 2009 dollars to 2012 dollars yields a VSL of approximately $7.7 million. The International Commission on Radiation Protection (ICRP) updated the mortality risk factor in ICRP Publication No. 103 (Ref. D.3); the updated risk coefficient is 5×10^{-4}. Using the updated ICRP risk coefficient and escalated EPA-based VSL, the dollar per person-rem conversion, rounded to the nearest thousand, is $4,000 per person-rem.

The staff is aware that the $2,000 per person-rem conversion factor may change as a result of ongoing assessments. However, the value of the dollar per person-rem conversion factor is a matter of Commission policy. Therefore, the NRC used the $2,000 per person-rem conversion value for the recommendation and the $4,000 per person-rem conversion value as a sensitivity study for this analysis.

D.3 Replacement Energy Costs

The NRC is currently updating its estimates for replacement energy costs based on a U.S. competitive electricity market area model. The updated model provides the replacement energy costs by day, week, and year, based on market area, in 2010 dollars. For each U.S. power market area, a lowest cost and highest cost replacement energy cost estimate was calculated, normalizing for reactor megawatt rating differences. The estimated replacement energy cost per reactor per year ranges from a high estimate of $54.4 million to a low estimate of $692,000 across all U.S. power markets. The average estimated cost per reactor per year across all U.S. power markets is $9.6 million and the median estimated cost is $6.4 million in 2010 dollars. Using the CPI-U inflator formula and the 2010 CPI-U inflator value from Table 31, the estimated replacement energy costs range from $57.3 million to $729,000 in 2012 dollars. The average estimated cost per reactor per year across all U.S. power markets is $10.1 million and the median estimated cost is $6.7 million in 2012 dollars.

D.4 Consequences Extending Beyond 50 Miles

NUREG/BR-0184 states that in the case of nuclear power plants, changes in public health and safety from radiation exposure and offsite property impacts should be examined over a 50-mile (80-kilometer) distance from the plant site. However, in this circumstance it is beneficial for the analysis to include supplemental information (e.g., analyses and results) that go beyond the guidance provided in this document. The SFPS uses a plume release model that predicts slow deposition of aerosols containing long-lived (i.e., slowly decaying) isotopes that results in public radiation exposures beyond 50 miles from the postulated accident site. While the accuracy of the model decreases with distance, this cost-benefit analysis evaluates the public health and safety and economic consequences estimated by the plume model beyond the 50-mile distance from the plant site as a sensitivity analysis. Refer to section 4.4.1.4 for results of these sensitivity analyses.

D.5 Sensitivity to a Uniform Fuel Pattern during an Outage

The base case of this analysis assumes that the licensee has prearranged the SFP such that discharged assemblies can be placed directly into a 1x4 arrangement for the discharges of the last two outages. This approach is consistent with the requirements discussed in Section 9.3 of the SFPS. However, those requirements do allow for the fuel to be stored in a less favorable configuration for some time following discharge if other considerations prevent prearrangement. A requirement is associated with the time window by which the 1x4 arrangement must be achieved; however, the specific time requirement is not publicly available information. To capture the effects of nonbeneficial arrangement of discharged fuel, this analysis evaluates the situation in which the discharged spent fuel is uniformly arranged during the outage to evaluate the effect of this aspect on the results. Refer to Appendix C, section C.2.15 for results of this sensitivity analysis.

D.6 References

D.1 Newell, R., and W. Pizer. "Discounting the distant future: how much do uncertain rates increase valuations?" Discussion Paper 00-45, May 14, 2001, Resources for the Future. Retrieved from http://weber.ucsd.edu/~carsonvs/papers/824.pdf, accessed 7/31/2013.

D.2 U.S. Environmental Protection Agency, National Center for Environmental Economics, "Valuing Mortality Risk Reductions for Environmental Policy: A White Paper", dated December 2010, Retrieved from http://yosemite.epa.gov/ee/epa/eerm.nsf/vwAN/EE-0563-1.pdf/$file/EE-0563-1.pdf, accessed July 26, 2013.

D.3 International Commission on Radiological Protection (ICRP), 2008. "The 2007 Recommendations of the International Commission on Radiological Protection," Publication 103. *Ann*. ICRP 37 (2-4), 2008.

APPENDIX E: INDUSTRY IMPLEMENTATION MODEL OF MOVING SPENT FUEL TO DRY CASK STORAGE

E.1 Group 1 Spent Fuel Pool

As previously discussed in Appendix section C.4.3, during each refueling outage the representative Group 1 plant discharges 284 fuel assemblies to the SFP. For the regulatory baseline case, the plant is expected to load the required number of DSCs with a 68-assembly capacity each refueling outage to retain sufficient space in the SFP to discharge one full core of fuel (full core reserve). For the expedited transfer alternative, low-density SFP storage case, the representative Group 1 plant SFP stores 852 assemblies, which is equivalent to the discharge from the last three refueling outages. For the expedited transfer alternative, the plant achieves this low-density storage condition within five years and then maintains this storage condition up through cessation of commercial operation. The cumulative DSC implementation costs for a single Group 1 SFP are shown in Figure 20.

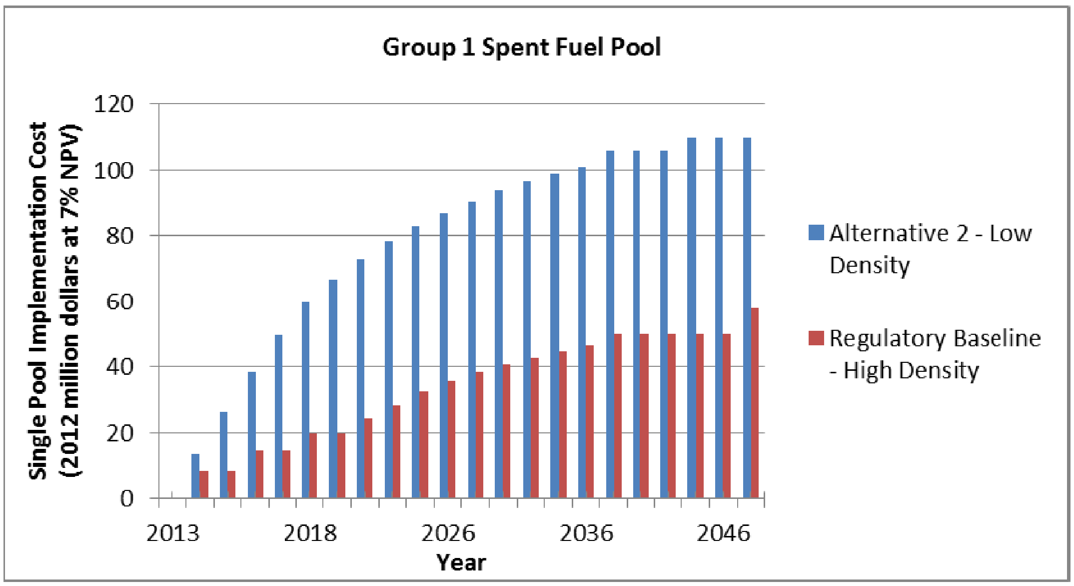

Figure 20 Cumulative dry cask storage implementation costs for a single Group 1 spent fuel pool

E.2 Group 2 Spent Fuel Pool

A similar calculation is performed for the Groups 2 SFPs. As previously discussed in Appendix section C.4.3, every 18-months the representative PWR plant discharges 84 fuel assemblies to the SFP. For the regulatory baseline case, the plant is expected to load the required number of Holtec Hi-Storm FW DSCs with a 37-assembly capacity each refueling outage to retain sufficient space in the SFP to discharge one full core of fuel (full core reserve). For the expedited transfer alternative, low-density SFP storage case, the representative plant SFP stores 312 fuel assemblies, the equivalent to the discharge from the last three refueling outages. The cumulative DSC implementation costs for Group 2 plants are shown in Figure 21.

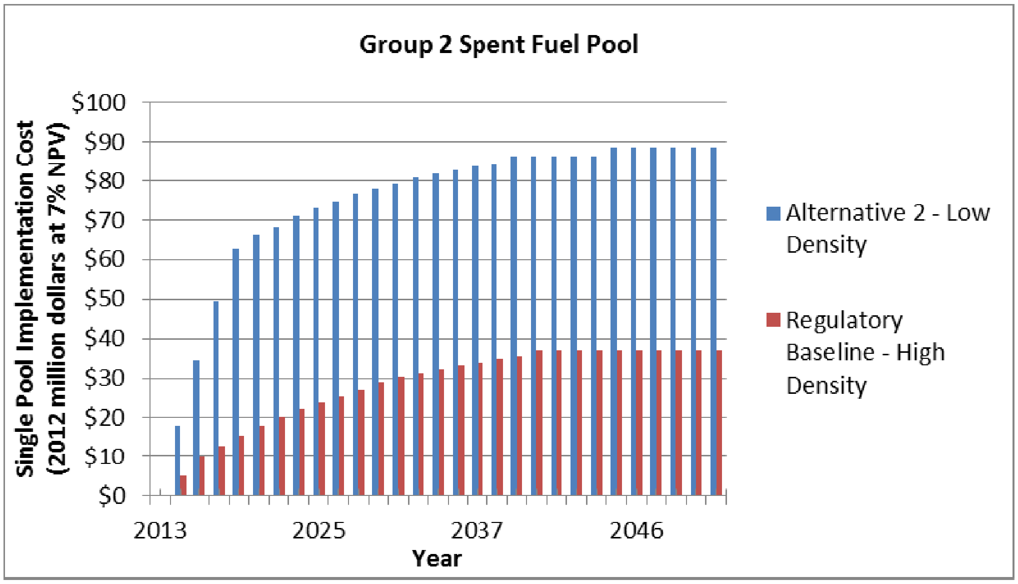

Figure 21 Cumulative dry cask storage implementation costs for a single Group 2 spent fuel pool

E.3 Group 3 Spent Fuel Pool

In 2018, the representative Group 3 plant is assumed to begin commercial operation. At this time, there are no spent fuel assemblies stored in the SFP. The unit is assumed to operate on an 18-month refueling cycle, discharging an estimated 69 assemblies per cycle as discussed in Appendix section C.4.3. For the regulatory baseline, the representative new nuclear plant is expected to begin dry storage in 2038 and will load a sufficient number of Holtec Hi-Storm FW casks to maintain its full core offload capability. The cumulative DSC implementation costs for Group 3 plants are shown in Figure 22.

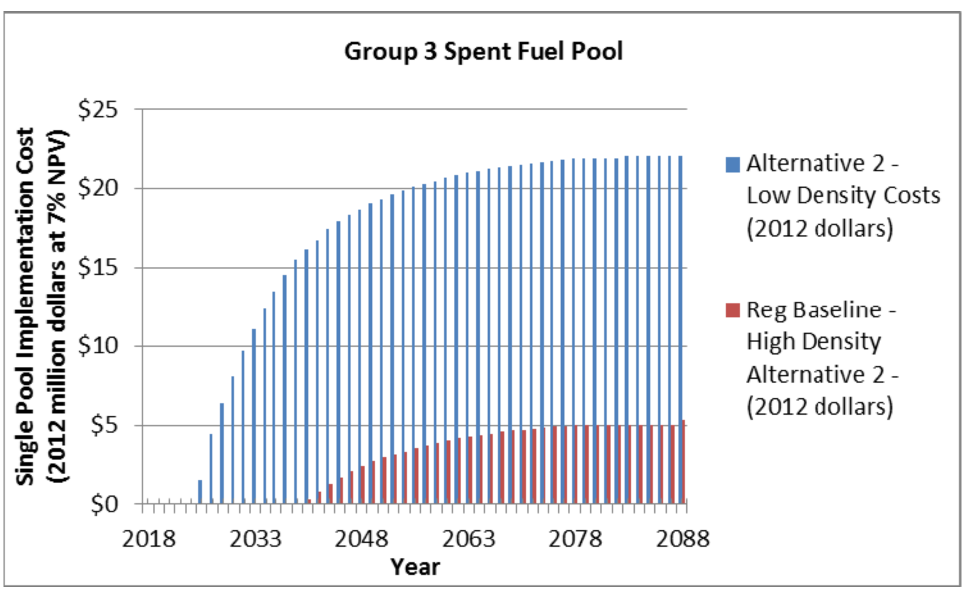

Figure 22 Cumulative dry cask storage implementation costs for a single Group 3 spent fuel pool

E.4 Group 4 Spent Fuel Pool

The representative Group 4 SFP is shared between two PWR units and is assumed to have 1,637 fuel assemblies stored in the SFP in a high-density 1x4 loading configuration. Each reactor unit operates on an 18-month refueling cycle and discharges 84 assemblies during the shoulder months from May through June and September into early November during the same calendar year. For the regulatory baseline, the Group 4 SFP is expected to load the required number of DSCs with a 37-assembly capacity each refueling outage to retain sufficient space in the SFP to discharge one full core of fuel (full core reserve). For the low-density case, the DSC has a 33-assembly capacity because of the higher heat load of the spent fuel. At the cessation of commercial operation, which occurs on average in 2038 for the Group 4 SFP reactors, the full core is offloaded into the SFP. The analysis further assumes that the entire SFP inventory will be placed into dry storage by 2048 for the regulatory baseline and by 2043 for the low-density storage case. The cumulative DSC implementation costs for Group 4 plants are shown in Figure 23.

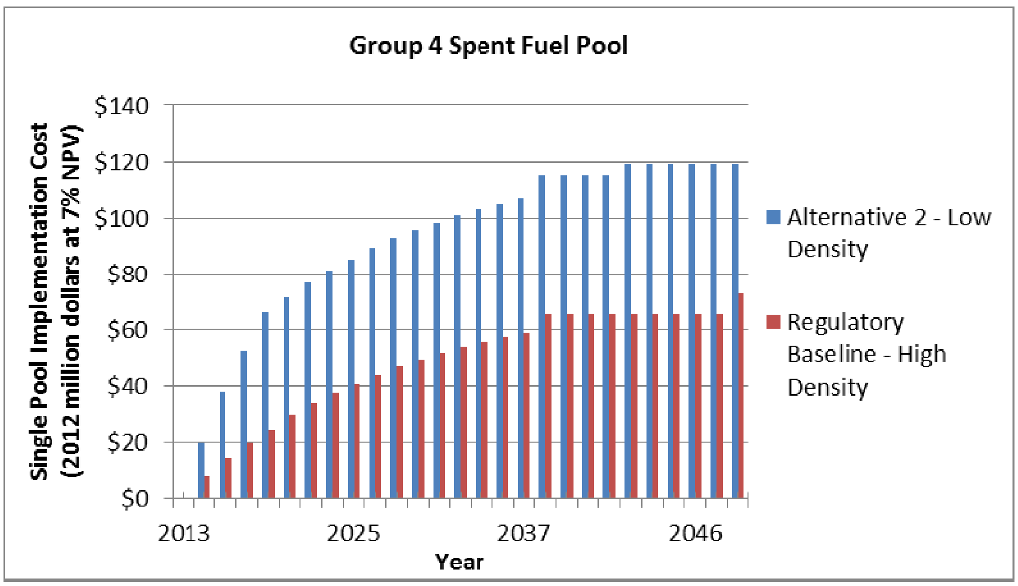

Figure 23 Cumulative dry cask storage implementation costs for a single Group 4 shared spent fuel pool

APPENDIX F: SPENT FUEL DATA AND TABLES

Table 70 Dry Spent Fuel Storage at U.S. Commercial Nuclear Power Plants

Plant Name	Company Name	Fuel Type	Location	License Type	Storage Technology	Year Loaded
Surry 1 & 2	Dominion Generation	PWR	Co-located	Site-specific	CASTOR V/21 MC-10, NAC I-28 CASTOR X/, TN-32	1986
				General	NUHOMS-32PTH	2007
H.B. Robinson	Progress Energy	PWR	Co-located	Site-specific	NUHOMS-07P	1989
				General	NUHOMS-24PTH	2004
Oconee 1, 2, 3	Duke Energy	PWR	Co-located	Site-specific	NUHOMS-24P	1990
				General	NUHOMS-24P NUHOMS-24PHB	2000
Fort St. Vrain (shutdown)	U.S. DOE (Previously owned by Public Service Colorado)	HTGR	–	Site-specific	Foster Wheeler MVDS	1991
Calvert Cliffs 1 & 2	Constellation Energy	PWR	Co-located	Site-specific	NUHOMS-24P NUHOMS-32P	1992
Palisades	Entergy Nuclear Operations	PWR	Co-located	General	VSC-24 NUHOMS-32PT NUHOMS-24PTH	1993
Prairie Island 1 & 2	Xcel Energy	PWR	Co-located	Site specific	TN-40	1993
Point Beach 1 & 2	FPL Energy Point Beach	PWR	Co-located	General	VSC-24 NUHOMS-32PT	1995
Davis Besse	FirstEnergy Nuclear Operating Co.	PWR	Co-located	General	NUHOMS-24P	1995
Arkansas Nuclear One 1 & 2	Entergy Nuclear Operations	PWR	Co-located	General	VSC-24 HI-STORM 24P HI-STORM 32P	1996
North Anna 1 & 2	Dominion Generation	PWR	Co-located	Site-specific	TN-32	1998
				General	NUHOMS-32PTH	2008
Susquehanna 1 & 2	PPL Susquehanna LLC	BWR	Co-located	General	NUHOMS-52B NUHOMS-61BT	1999
Peach Bottom 2 & 3	Exelon Generation	BWR	Co-located	General	TN-68	2000
Dresden 1, 2, 3 (Unit 1 – shutdown)	Exelon Generation	BWR	Co-located	General	HI-STAR 68B HI-STORM 68B	2000
Hatch 1 & 2	Southern Nuclear Operating Co.	BWR	Co-located	General	HI-STAR 68B HI-STORM 68B	2000
Rancho Seco (shutdown)	Sacramento Municipal Utility District	PWR	–	Site-specific	NUHOMS-24P	2001
McGuire 1 & 2	Duke Energy	PWR	Co-located	General	TN-32 NAC UMS	2001
Trojan (shutdown)	Portland General Electric	PWR	–	Site-specific	TranStor Overpack HI-STORM 24P MPC	2002
Oyster Creek	Exelon	BWR	Co-located	General	NUHOMS-61BT	2002

Plant Name	Company Name	Fuel Type	Location	License Type	Storage Technology	Year Loaded
	Generation					
Yankee Rowe (shutdown)	Yankee Atomic Electric Co.	PWR	Stand Alone	General	NAC MPC	2002
Columbia	Energy Northwest	BWR	–	General	HI-STORM 68B	2002
Big Rock Point (shutdown)	Entergy Nuclear Operations	BWR	Stand Alone	General	FuelSolutions W150	2002
FitzPatrick	Entergy Nuclear Operations	BWR	Co-located	General	HI-STORM 68B	2002
Maine Yankee (shutdown)	Maine Yankee Atomic Power	PWR	Stand Alone	General	NAC UMS	2002
Palo Verde 1, 2, 3	Arizona Public Service	PWR	–	General	NAC UMS	2003
San Onofre 1, 2, 3 (Unit 1 – shutdown)	Southern California Edison	PWR	–	General	NUHOMS-24PT	2003
Duane Arnold	FPL Energy.	BWR	Co-located	General	NUHOMS 61BT	2003
Haddam Neck (shutdown)	Connecticut Light & Power	PWR	–	General	NAC MPC	2004
Sequoyah 1 & 2	Tennessee Valley Authority	PWR	Co-located	General	HI-STORM 32P	2004
Millstone 1, 2, 3 (Unit 1 – shutdown)	Dominion Generation	Unit 1 – BWR Unit 2, 3 – PWR	Co-located	General	NUHOMS-32PT	2005
Farley 1 & 2	Southern Nuclear Operating Co.	PWR	Co-located	General	HI-STORM 32P	2005
Browns Ferry 1, 2, 3	Tennessee Valley Authority	BWR	Co-located	General	HI-STORM 68B	2005
Quad Cities 1 & 2	Exelon Generation	BWR	Co-located	General	HI-STORM 68B	2005
River Bend	Entergy Nuclear Operations	BWR	Co-located	General	HI-STORM 68B	2005
Fort Calhoun	Omaha Public Power District	PWR	Co-located	General	NUHOMS-32PT	2006
Hope Creek	PSEG Nuclear	BWR	Co-located	General	HI-STORM 68B	2006
Grand Gulf	Entergy Nuclear Operations	BWR	Co-located	General	HI-STORM 68B	2006
Catawba 1 & 2	Duke Energy	PWR	Co-located	General	NAC UMS	2007
Indian Point 1, 2, 3 (Unit 1 – shutdown)	Entergy Nuclear Operations	PWR	Co-located	General	HI-STORM 32P	2008
Vermont Yankee	Entergy Nuclear Operations	BWR	Co-located	General	HI-STORM 68B	2008
Limerick 1 & 2	Exelon Generation	BWR	Co-located	General	NUHOMS 61BT	2008
St. Lucie 1 & 2	FPL Energy	PWR	Co-located	General	NUHOMS 32PT	2008
Seabrook	FPL Energy	PWR	Co-located	General	NUHOMS 32PT	2008

Plant Name	Company Name	Fuel Type	Location	License Type	Storage Technology	Year Loaded
Monticello	Xcel Energy	BWR	Co-located	General	NUHOMS 61BT	2008
Humboldt Bay (shutdown)	Pacific Gas & Electric	BWR	Co-located	Site-specific	HI-STAR 100	2008
Kewaunee	Dominion Generation	PWR	Co-located	General	NUHOMS-32P	2009
Diablo Canyon 1 & 2	Pacific Gas & Electric	PWR	–	Site-specific	HI-STORM 32P	2009

Source: EPRI TR-1021048, pp. 2-10 to 2-12 (Ref. F.1).

Table 71 Expected Dry Spent Fuel Storage Facility Development at U.S. Commercial Nuclear Power Plants

Plant Name	Company Name	Location	Fuel Type	Approximate Loading Year
Beaver Valley 1	FirstEnergy Nuclear Operating Co.	–	PWR	2013-2014
Brunswick 1 & 2	Progress Energy	Co-located	BWR	2010-2011
Braidwood 1 & 2	Exelon Generation	–	PWR	2011
Byron 1 & 2	Exelon Generation	Co-located	PWR	2010
Clinton	Exelon Generation	–	BWR	2016
Comanche Peak	TXU Generating Company	–	PWR	2014-2016
Cook 1 & 2	Indiana Michigan Power	–	PWR	2011
Cooper	Nebraska Public Power District	Co-located	BWR	2010
Crystal River	Progress Energy	–	PWR	2012
Fermi	Detroit Edison	Co-located	BWR	2010
Ginna	Constellation Energy	Co-located	PWR	2010
LaCrosse (shutdown)	Dairyland Power	–	BWR	2011
LaSalle 1 & 2	Exelon Generation	Co-located	BWR	2010
Nine Mile Point 1 & 2	Constellation Energy	–	BWR	2012
Perry	FirstEnergy	Co-located	BWR	2010
Pilgrim	Entergy Nuclear Operations	–	BWR	2014-2015
Salem 1 & 2	PSEG Nuclear	Co-located	PWR	2010
Summer	South Carolina Electric & Gas	–	PWR	2015-2017
Turkey Point 3 & 4	FPL Energy	–	PWR	2011
Vogtle 1 & 2	Southern Nuclear Operating Co.	–	PWR	2013-2014
Waterford 3	Entergy Nuclear Operations	–	PWR	2011-2012
Watts Bar 1 & 2	Tennessee Valley Authority	–	PWR	2020

Source: EPRI TR-1021048, p. 2-13 (Ref. F.1).

Table 72 Spent Fuel Pool Capacities

Plant Name	Spent Fuel Pool			
	Group[1]	Assoc. Reactor Core Size (no. of assemblies)	Technical Specification Capacity (assemblies/core equivalents)	Estimated Cs-137 Inventory (MCi)
Arkansas Nuclear 1	2	177	968/ 5.5	41.7
Arkansas Nuclear 2	2	177	988/ 5.6	42.8
Beaver Valley 1	2	157	1627/ 10.4	77.6
Beaver Valley 2	2	157	1627/ 10.4	77.6
Braidwood 1	4	193	2984/ 7.7 per unit[2]	142.2
Braidwood 2	4	193		
Browns Ferry 1	1	764	3471/ 4.5[4]	52.3
Browns Ferry 2	1	764	3471/ 4.5[4]	52.3
Browns Ferry 3	1	764	3471/ 4.5	52.3
Brunswick 1	1	560	1803/ 3.2	24.0
Brunswick 2	1	560	1839/ 3.3	24.7
Byron 1	4	193	2984/ 7.7 per unit[2]	142.2
Byron 2	4	193		
Callaway	2	193	2363/ 12.2	114.5
Calvert Cliffs 1	4	217	1830/ 4.2 per unit[4]	79.4
Calvert Cliffs 2	4	217		
Catawba 1	2	193	1421/ 7.3	64.8
Catawba 2	2	193	1421/ 7.3	64.8
Clinton	2	624	3796/ 6.1	61.3
Columbia	1	764	2658/ 3.5	36.6
Comanche Peak 1	2	193	1684/ 8.7[4]	78.7
Comanche Peak 2	2	193	1689/ 8.7[4]	79.0
Cooper	1	548	2651/ 4.8	40.6
Crystal River 3	6	177	1474/ 8.3	68.5
Davis-Besse	2	177	1624/ 9.2	76.4
D.C. Cook 1	4	193	3613/ 9.3 per unit[2]	175.4
D.C. Cook 2	4	193		
Diablo Canyon 1	2	193	1324/ 6.9	59.7
Diablo Canyon 2	2	193	1324/ 6.9	59.7
Dresden 2	1	724	3537/ 4.9	54.3
Dresden 3	1	724	3537/ 4.9	54.3
Duane Arnold	1	368	2829/ 7.7	47.5
Farley 1	2	157	1407/ 9.0	66.0
Farley 2	2	157	1407/ 9.0	66.0
Fermi 2	1	764	4608/ 6.0	74.2
FitzPatrick	1	560	3239/ 5.8	51.7
Fort Calhoun	2	133	1083/ 8.14	50.1
Ginna	2	121	1321/ 10.9	63.3
Grand Gulf 1	2	800	4348/ 5.4	68.5
Hatch 1	1	560	3349/ 6.0[4]	53.9
Hatch 2	1	560	2933/ 5.2[4]	45.8
Hope Creek 1	1	764	4006/ 5.2	62.6
Indian Point 2	2	193	1374/ 7.1	62.3
Indian Point 3	2	193	1345/ 7.0	60.8
Kewaunee	6	121	1205/ 10.0	57.2

Plant Name	Spent Fuel Pool			
	Group[1]	Assoc. Reactor Core Size (no. of assemblies)	Technical Specification Capacity (assemblies/core equivalents)	Estimated Cs-137 Inventory (MCi)
La Salle County 1	1	764	3986/ 5.2[4]	62.2
La Salle County 2	1	764	4078/ 5.3[4]	64.0
Limerick 1	1	764	4117/ 5.4	64.8
Limerick 2	1	764	4117/ 5.4	64.8
McGuire 1	2	193	1463/ 7.6	67.0
McGuire 2	2	193	1463/ 7.6	67.0
Millstone 1	6	–	–	–
Millstone 2	2	217	1346/ 6.2	59.6
Millstone 3	2	193	1860/ 9.6	88.0
Monticello	1	484	2301/ 4.75	35.1
Nine Mile Point 1	1	532	4086/ 7.7	68.6
Nine Mile Point 2	1	764	4049/ 5.3	63.4
North Anna 1	4	157	1737/ 5.5 per unit[2]	79.2
North Anna 2	4	157		
Oconee 1	4	177	1312/ 3.7 per unit[2]	55.2
Oconee 2	4	177		
Oconee 3	2	177	825/ 4.7	34.2
Oyster Creek	1	560	3035/ 5.4	47.8
Palisades	2	204	892/ 4.4	36.3
Palo Verde 1	2	241	1329/ 5.5	57.4
Palo Verde 2	2	241	1329/ 5.5	57.4
Palo Verde 3	2	241	1329/ 5.5	57.4
Peach Bottom 2	1	764	3819/ 5.0	59.0
Peach Bottom 3	1	764	3819/ 5.0	59.0
Perry 1	2	748	4020/ 5.4	63.2
Pilgrim 1	1	580	3859/ 6.7	63.3
Point Beach 1	4	121	1502/ 6.2 per unit[2]	69.7
Point Beach 2	4	121		
Prairie Island 1	4	121	1386/ 5.7 per unit[2]	63.6
Prairie Island 2	4	121		
Quad Cities 1	1	724	3657/ 5.1[4]	56.6
Quad Cities 2	1	724	3897/ 5.4[4]	61.3
River Bend 1	2	624	3104/ 5.0	47.9
Robinson 2	2	157	544/ 3.5	20.4
St. Lucie 1	2	217	1706/ 7.9	78.6
St. Lucie 2	2	217	1491/ 6.9	67.2
Salem 1	2	193	1632/ 8.5	75.9
Salem 2	2	193	1632/ 8.5	75.9
San Onofre 2	6	217	1542/ 7.1	69.9
San Onofre 3	6	217	1542/ 7.1	69.9
Seabrook 1	2	193	1236/ 6.4	55.0
Sequoyah 1	4	193	2091/ 5.4 per unit[2]	95.1
Sequoyah 2	4	193		
Shearon Harris 1	2	157 (PWR)	PWR fuel: 3404 / 21.7 or	167.2
		560 (BWR)	BWR fuel: 4628 / 8.3	73.2
South Texas Project 1	2	193	1969/ 10.2	95.6

Plant Name	Spent Fuel Pool			
	Group[1]	Assoc. Reactor Core Size (no. of assemblies)	Technical Specification Capacity (assemblies/core equivalents)	Estimated Cs-137 Inventory (MCi)
South Texas Project 2	2	193	1969/ 10.2	95.6
Summer 1	2	157	1276/ 8.1	59.1
Summer 2	3	–	–	–
Summer 3	3	–	–	–
Surry 1	4	157	1044/ 3.3 per unit[2]	42.7
Surry 2	4	157		
Susquehanna 1	1	764	2840/3.7[4]	40.1
Susquehanna 2	1	764	2840/ 3.7[4]	40.1
Three Mile Island 1	2	177	1338/ 7.6	61.3
Turkey Point 3	2	157	1395/ 8.9	65.3
Turkey Point 4	2	157	1389/ 8.9	65.0
Vermont Yankee	1	368	3355/ 9.1	57.7
Vogtle 1	2	193	1476/ 7.6[4]	67.7
Vogtle 2	2	193	2098/ 10.9[4]	100.5
Vogtle 3	3	–	–	–
Vogtle 4	3	–	–	–
Waterford 3	2	217	2398/ 11.0	115.1
Watts Bar 1	2	193	1610/ 8.3	74.8
Wolf Creek 1	2	193	2363/ 12.2	114.5
Zion 1	6	–	–	–
Zion 2				

Notes:
1. The Group column corresponds to the SFP groupings discussed in Section 4.1.1.
2. Common pool shared by two reactors. Shared SFPs are required to maintain one full core reserve. However, with the practice that both reactors refuel during the shoulder months of the same year it was judged that shared pools attempt to maintain at least a 1.5 full core reserve in practice.
3. Shearon Harris SFP holds fuel from Robinson and Brunswick.
4. SFPs connected by transfer canal.

Table 73 Cost-Benefit Analysis Inputs Summary

Parameter	Spent Fuel Pool Group 1			Spent Fuel Pool Group 2, 3, & 4		
	Low Est.	Base Case	High Est.	Low Est.	Base Case	High Est.
Seismic hazard initiating event frequency (USGS 2008 model) (per year)						
- Seismic bin 3	1.65E-05	1.65E-05	2.24E-05	1.65E-05	1.65E-05	see Table 43
- Seismic bin 4	4.90E-06	4.90E-06	7.09E-06	4.90E-06	4.90E-06	see Table 43
ac power fragility	100% (bounding value)					
Liner fragility						
- Seismic bin 3	10%	10%	100%	2%	5%	25%
- Seismic bin 4	50%	100%	100%	16%	50%	100%
- Cask drop	100% (bounding value)					
Percent of operating cycle natural circulation cooling is insufficient						
- Seismic bin 3	8%	8%	100%	8%	100% (bounding value)	
- Seismic bin 4	30%	100% (bounding value)		30%	100% (bounding value)	
- Cask drop	8%	100% (bounding value)		8%	100% (bounding value)	
- All other initiators	100% (bounding value)					
Cs-137 release fraction						
- Alternative 1	3%	40%	90%	10%	75%	90%
- Alternative 2	0.5%	3%	5%	0.5%	3%	5%
High-density loading spent fuel pool Cs-137 inventory (MCi)						
- SFP Group 1	40.6	52.7	63.3	-	-	-
- SFP Group 2	-	-	-	57.4	67.9	78.2
- SFP Group 3	-	-	-	33.7	44.4	54.2
- SFP Group 4	-	-	-	63.6	101.1	142.2
Low-density loading spent fuel pool Cs-137 inventory (MCi)						
- SFP Group 1	19.8	22	26.4	-	-	-
- SFP Group 2	-	-	-	15.7	17.4	20.9
- SFP Group 3	-	-	-	15.7	17.4	20.9
- SFP Group 4	-	-	-	31.4	34.8	41.8
Population density within 50 miles of site (people/square mile)	169	317	722	169	317	722
Long-term habitability criteria	500 mrem annually	2 rem first year and 500 mrem each year thereafter	2 rem annually	500 mrem annually	2 rem first year and 500 mrem each year thereafter	2 rem annually
Onsite Property: decontamination, repair, & refurbishment	$303 million	$606 million	$1.82 billion	$303 million	$606 million	$1.82 billion
Short-term occupational exposure (accident) (person-rem)	18,070	28,380	48,880	18,070	28,380	48,880
Long-term occupational exposure (accident (person-rem)	4,580	14,000	46,000	4,580	14,000	46,000
Economic data near site	Palisades	Surry	Peach Bottom	Palisades	Surry	Peach Bottom

APPENDIX F REFERENCES

F.1 EPRI TR-1021048, “Industry Spent Fuel Storage Handbook,” dated July 2010.

APPENDIX G: QUESTIONS RAISED BY THE PUBLIC

The NRC staff conducted two public meeting pertaining to this body of work to gain stakeholder input and feedback on the work conducted and the staff's preliminary conclusions pertaining to the issue. This section addresses some of the questions received during those public meetings.

1) Question

The analysis applies to all sites across the U.S. fleet, so how were the variations in seismicity at the sites considered?

Response

The staff used conservative values for several parameters in the cost-benefit analysis to ensure that design, operational and other site variations among the new and operating reactor fleet were encompassed. For example, the probabilities of exceeding a specified peak ground acceleration at the reference plant fall close to the upper end of each SFP group. However, the amount of conservatism used in the other base case parameters overwhelms the slight non-conservatism in the outlying site seismicity parameter. Therefore, the overall results of the base case are conservative for all plants.

To quantify the effect of exceeding the ground motion estimates used in the base case, a high estimate case was created that conservatively selected the site within each SFP group with the highest plant hazard exceedance frequency for peak ground accelerations greater than 0.6g. The sites selected and the seismic initiating event frequencies values used are listed in Table 37.

2) Question

How were the differences in likelihood of successful mitigation across the sites treated?

Response

Operator diagnosis and recovery are important factors considered in the development of the event frequencies for the successful mitigation of accident events. Success is premised on licensees having taken appropriate actions to understand the potential consequences of spent fuel pool accident events and develop appropriate procedures and mitigating strategies to respond and mitigate the consequences. Specific spent fuel pool loss of water inventory mitigation measures are required under Title 10 of the Code of Federal Regulations (10 CFR) 50.54(hh)(2), which were implemented following the September 11, 2001 attacks. Additionally, the post-Fukushima mitigation required by the NRC in Orders EA-12-051 and EA-12-049 and currently being implemented by all U.S. nuclear power plants should serve to further reduce spent fuel pool accident risk by increasing the capability of nuclear power plants to mitigate beyond-design-basis external events further reducing the frequency of a spent fuel pool accident release. This cost-benefit analysis used a conservative approach to mitigation by crediting successful mitigation to the low-density spent fuel pool storage alternative and assuming no successful mitigation for the high-density spent fuel pool storage regulatory baseline. In this manner the staff biased the results to favor the regulatory action of expediting fuel transfer to dry casks and provided margin to address uncertainties associated with other assumptions.

3) Question

Since the event considered is a large seismic event, how would the accident involving the reactor core affect the study results?

Response

Detailed accident progression analysis were performed in SFPS for high density loading cases including sensitivities to hydrogen combustion (Section 9.1 of the SFPS) and concurrent reactor events (Section 9.4 of the SFPS) leading to the failure of the reactor building. These calculations considered uncertainties associated with hydrogen ignition, and formation of debris leading to blockages at the exit of the assemblies and reduced flow area, and led to a range of release fractions. The base case in the regulatory analysis for high-density loading was based on the average release fractions for small leak scenarios (which result in larger releases than medium leaks) including the uncertainties that resulted in high releases because of hydrogen combustion and significant air oxidation.

A concurrent reactor accident would affect the likelihood of successful implementation of mitigating strategies for the pool. The cost-benefit analysis bounds this effect by assuming that operators were unsuccessful in mitigating the ongoing SFP accident for 72 hours for the high-density case (regulatory baseline) and assumed 100% success rate for mitigation strategies for the low-density case. This biases the results in favor of low-density loading.

Although not considered explicitly in the analysis, the NRC recently issued Orders EA-12-051 and EA-12-049 to all U.S. nuclear power plants which should serve to further reduce core damage risk and SFP accident risk by increasing the capability of nuclear power plants to mitigate beyond-design-basis external events. The staff is currently performing a comprehensive site Level 3 PRA for a U.S. PWR as discussed in SECY-11-0089 and the staff will revisit this issue upon its completion.

4) Question

How was debris generated by hydrogen explosions in the SFP considered?

Response

Based on the structural analyses performed in the SFPS for the reference plant reactor building and overhead crane, no significant debris generated by the seismic event is expected to enter the SFP. Although as stated in Table 3 of the SFPS, some debris could be generated and could fall into the pool as a result of hydrogen combustion. However, the occurrence of a hydrogen combustion event in the SFPS denotes that the fuel in the SFP has already become uncovered and is undergoing a fission product release. The reduction in flow area and losses associated with debris generated from a hydrogen combustion resulting from a reactor accident are explicitly considered in Section 9.4 of the SFPS and described in the response to Question 4 above.

5) Question

The probability of a loss of SFP inventory calculated for the reference was used in the cost-benefit analysis. How were differences in SFP liner failure rates, perhaps due to aging, considered?

Response

The detailed structural analyses performed for the reference plant in the Spent Fuel Pool Study predicted that under the seismic load studied, the liner would fail approximately 10% of the time with either a small or medium sized rupture. To account for any variations in liner material properties, the Tier 3 analysis assumed the liner failure values listed in Table 39. These liner fragility values in combination with other assumed failures provide a conservative estimate for the cost-benefit analysis of expedited transfer of spent fuel.

6) Question

Would the results change if open-frame racks were considered as an option?

Response

For the reference plant studied, the BWR fuel assemblies channel boxes would impede crossflow even with open-frame racks. Furthermore, even for the high-density racking, the study showed that without mitigative actions, fuel is estimated to be air-coolable for at least 72 hours for all but roughly 10% of the operating cycle. Based on the insights from the accident progression analyses in the SFPS, within the first few months after the fuel comes out of the reactor, the decay heat in the freshly unloaded spent fuel is high enough to cause a zirconium fire even in the presence of any additional convective cooling once natural circulation is established (see Figures 90 and 93 in the SFPS for the high-density and low-density pool loadings and a moderate leak). Therefore, open frame racks even with channel boxes removed to allow potential crossflow, would not necessarily prevent a radiological release during this time. In the cost-benefit analysis, values from the SFPS were used to model the BWR Mark I and II SFPs. For the other SFP groups, a simplifying assumption was made to account for the concern. In the base case and high estimate case as shown in Table 41, the analysis assumed that the fuel was never coolable under natural convection of air. This approach bounds any effect of considering open-frame racks.

7) Question

High burnup fuel, which has reduced ductility compared to fresh fuel, is being used across the industry. How would the results change if this reduction in fuel clad ductility were considered?

Response

Seismic loads would be relatively small and loading is slow due to spent fuel rack and fuel assemblies design and widespread damage is not expected even considering the mechanical property changes of high burnup spent fuel cladding. Furthermore, high-density spent fuel storage uses freestanding sliding racks that tend to limit the stresses on the racks and spent fuel assemblies immersed in water, even when the seismic loads increase beyond those calculated for design basis seismic events. Adequate clearances are provided between the

racks and pool walls to avoid, with a margin, impacting of the racks during design basis seismic events. Collisions between racks that might result from seismic loads several times greater than the design basis loads would involve low impact speeds that are expected to be several times smaller than impact speeds in design basis transportation and storage accidents (e.g., 30 feet drop). In addition, some of the impact energy of seismic loads on the fuel would be dissipated by small permanent deformation of the rack structures, which would reduce the shock forces transmitted to the spent fuel relative to those in transportation and storage accidents.

The NRC continues to research the mechanical properties of high burnup cladding relevant for normal conditions of transport, including transportation vibration and fatigue failure. NRC is conducting tests to measure the loads required to fail high burnup spent fuel rods under static loading and a wide range of cycling loading conditions. The loading levels and test speeds are more comparable to seismic loading conditions than those in drop tests for storage and transportation accident conditions. Therefore, the conditions of these tests may be more applicable to assessments of safety margins for high burnup spent fuel assemblies stored in SFPs.

8) Question

Were inadvertent criticality scenarios for spent fuel in the pool considered in the analysis?

Response

Yes, inadvertent criticality scenarios were considered but are not expected to significantly affect this analysis based on the following reasons. Design requirements and related safety analyses ensure fuel stored in the SFP will remain safely subcritical under conditions considered as part of the design basis, but rare conditions beyond the design basis may challenge some measures used to control reactivity. To maintain adequate margin to criticality in U.S. SFPs, the safety analyses credit the geometric configuration of the fuel and a combination of other measures that may include fixed neutron poison material (e.g., Boraflex) and limits on fuel reactivity. In addition, the presence of soluble boron in the coolant of pressurized water reactor SFPs may be credited, but the stored fuel must remain subcritical assuming unborated water is present (10 CFR 50.68). Since these measures may be challenged by a beyond design-basis event, the NRC staff cannot rule out the potential for an inadvertent criticality event. However, the NRC staff judges that the potential consequences of a zirconium fire in the SFP and an associated hydrogen deflagration considered in this analysis would not be significantly affected by an inadvertent criticality event. The NRC staff bases this judgment on the following considerations.

While the earthquakes considered in this analysis are beyond what the fuel was designed to withstand, it is not likely that the fuel would experience sufficient damage to cause significant changes in the geometric configuration of the fuel needed to cause inadvertent criticality.

The necessary moderator would tend to shield and contain the effects of a criticality such that it would primarily pose an on-site rather than off-site hazard.

Criticality requires the presence of a moderator and therefore power would not be sustained as the pool lost inventory due to boiling or draindown. Since the power generated by any inadvertent criticality would be far lower than in the reactor, the inadvertent criticality would have

negligible impact on the long-lived fission product isotope inventory. The additional short-lived isotope inventory would not result in any early fatality risk because of the emergency response as modeled precludes such exposure. This is due in part because of the length of time needed before any fission products are released off-site.

Therefore, any off-site release associated with a criticality would be small relative to potential releases from a zirconium fire.

9) Question

How was the more limiting case of partial draindown considered?

Response

The cost-benefit analysis considered partial draindown events for the plants where this damage state was judged to be the more probable damage state such as SFPs located at grade. The effect was bounded by assuming that the fuel was not coolable by natural convection of air for the base case and high estimate case for these SFP groups as shown in Table 41.

10) Question

How does the study treat variations in population density across the sites?

Response

Since population density varies across the sites, the analysis includes a sensitivity study where the value is varied from low to high population density levels as represented by U.S. operating plant locations. Representative operating reactor site demographics were selected to represent the 90th percentile, the mean, the median, and the 20th percentiles.

11) Question

Since the SFP accident primarily occurs in an air oxidizing environment, how was the release of ruthenium accounted for?

Response

The study uses best estimate ruthenium release rates calculated by the MELCOR code. A model is provided to account for the high volatility of the ruthenium oxides when air ingress is assumed to lead to the formation of a moderately hyperstoichiometric fuel. Details of the modeling approach used for ruthenium release is provided in Section 6.1.5 of the SFPS.

12) Question

Plants may eventually use MOX fuel. Will the results apply to such cases?

Response

Mixed oxide fuel (commonly known as MOX) is not commonly used in the U.S. and very few assemblies of MOX fuel are currently stored. Fission product inventories in MOX fuel do not

differ significantly from those in uranium fuel. In general, since plutonium oxides accumulate in low-enrichment urania fuel as the burnup progresses, large differences in the degradation of MOX fuel and high burnup UO2 fuel would not be expected. Experiments with single MOX fuel pellets indicate higher release of volatile fission products from MOX than from uranium oxide fuel at low temperatures, with release rates converging as the temperature is increased. For SFPs, significant offsite radiological consequences only result from high temperature zirconium fire scenarios. In this study, large releases were associated with small leak scenarios that resulted in very high temperatures and collapse of the fuel rods, which included a range of release fractions depending on the size of the leakage and other factors to reasonably bound the differences between the MOX and uranium fuel types.

Non-Concurrence Process Record for NCP-2013-013

The U.S. Nuclear Regulatory Commission (NRC) strives to establish and maintain an environment that encourages all employees to promptly raise concerns and differing views without fear of reprisal and to promote methods for raising concerns that will enhance a strong safety culture and support the agency's mission.

Individuals are expected to discuss their views and concerns with their immediate supervisors on a regular, ongoing basis. If informal discussions do not resolve concerns, individuals have various mechanisms for expressing and having their concerns and differing views heard and considered by management.

Management Directive MD 10.158, "NRC Non-Concurrence Process," describes the Non-Concurrence Process (NCP). http://pbadupws.nrc.gov/docs/ML0706/ML070660506.pdf

The NCP allows employees to document their differing views and concerns early in the decision-making process, have them responded to, and attach them to proposed documents moving through the management approval chain.

NRC Form 757, Non-Concurrence Process is used to document the process.

Section A of the form includes the personal opinions, views, and concerns of an NRC employee.

Section B of the form includes the personal opinions and views of the NRC employee's immediate supervisor.

Section C of the form includes the agency's evaluation of the concerns and the agency's final position and outcome.

NOTE: Content in Sections A and B reflects personal opinions and views and does not represent official factual representation of the issues, nor official rationale for the agency decision. Section C includes the agency's official position on the facts, issues, and rationale for the final decision.

The agency's official position (i.e., the document that was the subject of the non-concurrence) is included in ADAMS Accession Number ML13273A572

This record is public.

NRC FORM 757
NRC MD 10.158
(7-2011)

U.S. NUCLEAR REGULATORY COMMISSION

NON-CONCURRENCE PROCESS

NCP TRACKING NUMBER
NCP-2013-013

SECTION A - TO BE COMPLETED BY NON-CONCURRING INDIVIDUAL

TITLE OF SUBJECT DOCUMENT Staff Evaluation and Recommendation for Japan Lessons-Learned Tier 3 Issue on Expedited Transfer of	ADAMS ACCESSION NO. ML13256A348
DOCUMENT SIGNER Michael R. Johnson (Revised to Mark Satorious, EDO)	SIGNER PHONE NO. (301) 415-1713
TITLE Deputy Executive Director for Reactors and Preparedness Progra	ORGANIZATION EDO
NAME OF NON-CONCURRING INDIVIDUAL(S) Brian Wagner	PHONE NO. (301) 251-7595
TITLE Reliability and Risk Engineer	ORGANIZATION RES/DRA/PRB

☐ DOCUMENT AUTHOR ☐ DOCUMENT CONTRIBUTOR ☑ DOCUMENT REVIEWER ☐ ON CONCURRENCE

REASONS FOR NON-CONCURRENCE AND PROPOSED ALTERNATIVES

The regulatory analysis contains a lot of good work and important insights, especially considering it was performed on a very abbreviated timetable. Further, it contains a respectable range of sensitivity analyses of the major parameters. However, the analysis and the transmittal memo omit some key information and analyses that would be beneficial to inform a decision on whether to continue regulatory activity in this area. In addition, the memo's discussion of the regulatory analysis' results is misleading in some areas.

1. Contrary to NUREG/BR-0058, "Regulatory Analysis Guidelines of the USNRC" guidance which recommends that "the range of all potentially reasonable and practical approaches to the problem are considered," only a single alternative is considered. Other alternatives may be more cost beneficial. For example, transferring less fuel or discharging into a 1x8 pattern may yield the same benefits while costing significantly less than the analyzed alternative. Both the draft Spent Fuel Pool Study (ML13133A132) and the ACRS letter (ML13224A060) recommended further analysis of the 1x8 fuel pattern. The draft COMSECY transmitting the regulatory analysis claims this would not provide a substantial safety enhancement despite it not being analyzed (or even mentioned) in the attached regulatory analysis.

2. The regulatory analysis uses $2000/person-rem as the baseline. It is known that a change in guidance is imminent that would change this value to the $4000-$5000/person-rem range to be more consistent with the practices of other agencies.

3. The regulatory analysis uses a 50-mile truncation as a baseline. Guidance in NUREG/BR-0058 indicates that a 50 mile truncation should be used for nuclear power plants but that the appropriate distance for other facilities should be decided on a case-by-case basis. For SFP accidents in high density pools, which are expected to release much more material than reactor accidents, this truncation can decrease the calculated consequences by nearly a factor of 10. This truncation is arbitrary and technically indefensible.

4. The SECY paper and regulatory analysis argues that no further action is necessary since the alternative does not represent a substantial safety enhancement. It is not clear how this position reconciles with the SRM to SECY-93-086, which states that the substantial standard "is not intended to be interpreted in a manner that would result in disapprovals of worthwhile safety or security improvements having costs that are justified in view of the increased protection that would be provided." The substantial safety enhancement screen should not be used to dismiss cost-beneficial results or as a reason to not compute cost-benefit information for other reasonable alternatives.

☑ CONTINUED IN SECTION D

SIGNATURE Brian Wagner DATE 9/24/13

SEE SECTION E FOR IMPLEMNATION GUIDANCE

U.S. NUCLEAR REGULATORY COMMISSION

NON-CONCURRENCE PROCESS

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ML13256A348

SECTION D: CONTINUATION PAGE

CONTINUATION OF SECTION ☑ A ☐ B ☐ C

5. The regulatory analysis answers the substantial safety enhancement question by comparing to the Quantitative Health Objectives (QHOs) found in the Safety Goal Policy Statement. Though this is standard practice, the QHOs were developed for reactor accidents and are not well suited for making this determination for SFP accidents. SFP accidents in high density pools can lightly contaminate very large areas, displacing millions of people and requiring extensive protective actions. Conversely, the individual LCF risk from 0-10 miles is relatively low, even for the largest releases. SFP releases would have to occur with a frequency greater than 10^-3 per year to approach the safety goals (100x higher than the Large Early Release Frequency subsidiary objective used for reactors.) While an alternative measure of a substantial safety enhancement is not readily available, one informative metric is that, for some "high estimate" cases, the proposed alternative results in nearly a billion dollars in frequency-weighted safety benefits. The SECY paper should acknowledge the significant limitations of applying the QHOs to non-reactors to provide The Commission with relevant information to inform their decision.

6. The regulatory analysis concludes the alternative is not cost-beneficial. This is in spite of the fact that the fleet is only bounded by the high estimates (which are shown to be cost-beneficial) and not the base case estimates.

7. Though the Regulatory Analysis contains an appropriate range of estimates and sensitivity results, both the "Decision Rationale" section of the regulatory analysis and the discussion of the results in the COMSECY transmitting the regulatory analysis fail to provide a balanced view of the range of results. There are several examples of this:

o The COMSECY states that conservative assumptions are used in the regulatory analysis without making it clear that conservatives are primarily to account for variations within the group considered in the high estimates. The base case estimates represent a point estimate and contain a few minor conservatisms. The base case estimates do not bound the group of SFPs.

o The COMSECY states "it is unlikely that individual plants would meet or exceed the most conservative assumptions made in these sensitivity cases within the regulatory analysis." This is highly misleading. The cases referenced are extremely cost-beneficial so a pool even approaching these assumptions would be very cost beneficial.

o The "Decision Rationale" section of the regulatory analysis states there are other considerations discussed in Section 4.5.10 that would further decrease the benefits and make the proposed alternative less cost-justified. Though some of the items discussed would clearly decrease the benefits (e.g. credit for mitigation) others could increase or decrease the benefits. The list omits considerations which would increase the benefits such as relaxing the potentially optimistic assumptions that extensive protective actions are effective following a severe seismic event.

o The analysis concludes that the alternative is not cost-beneficial by apparently focusing on the base case estimate truncated at 50 miles and using $2000/person-rem. Results that are cost-beneficial are downplayed as resulting from combinations of high estimates "sensitivity studies and some combinations of high estimates … such that, in a few cases, the benefits…appear to be cost beneficial." This is inconsistent with the results of the regulatory analysis which are: all high estimates are cost beneficial regardless of what other assumptions are used; and, when considering all consequences and an updated value of $4000/person-rem, all base cases are essentially cost neutral.

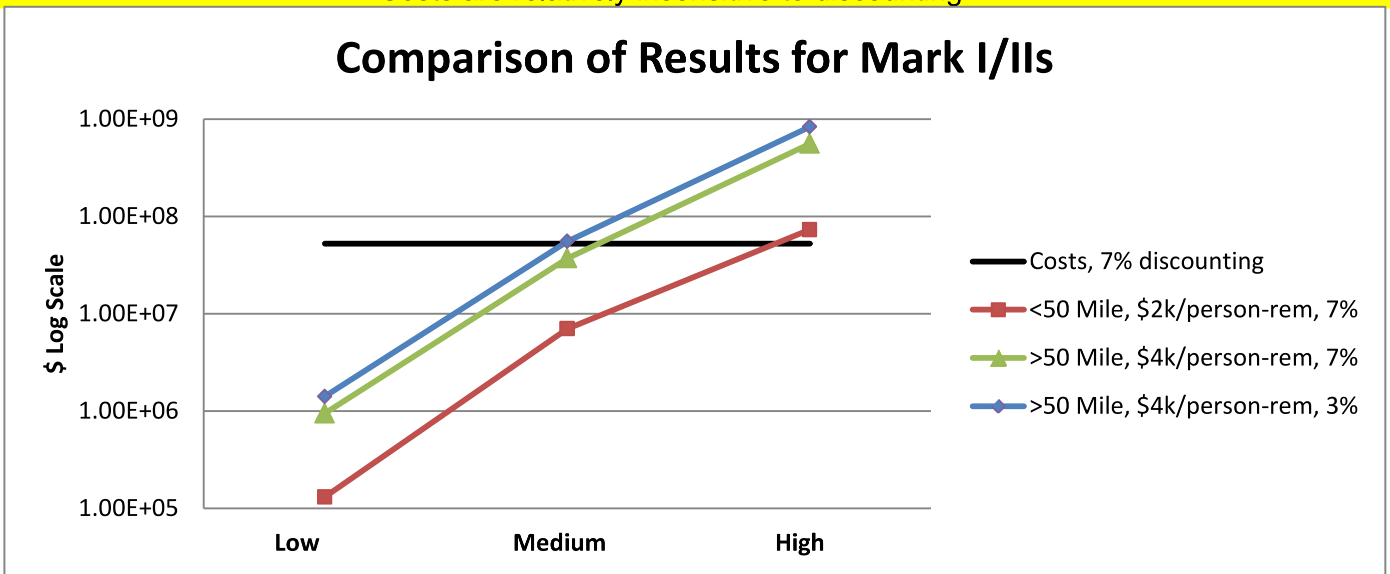
I have produced several figures and tables below to illustrate the results of the regulatory analysis. They paint a much muddier picture as to whether or not the alternative is cost-beneficial when compared to the COMSECY.

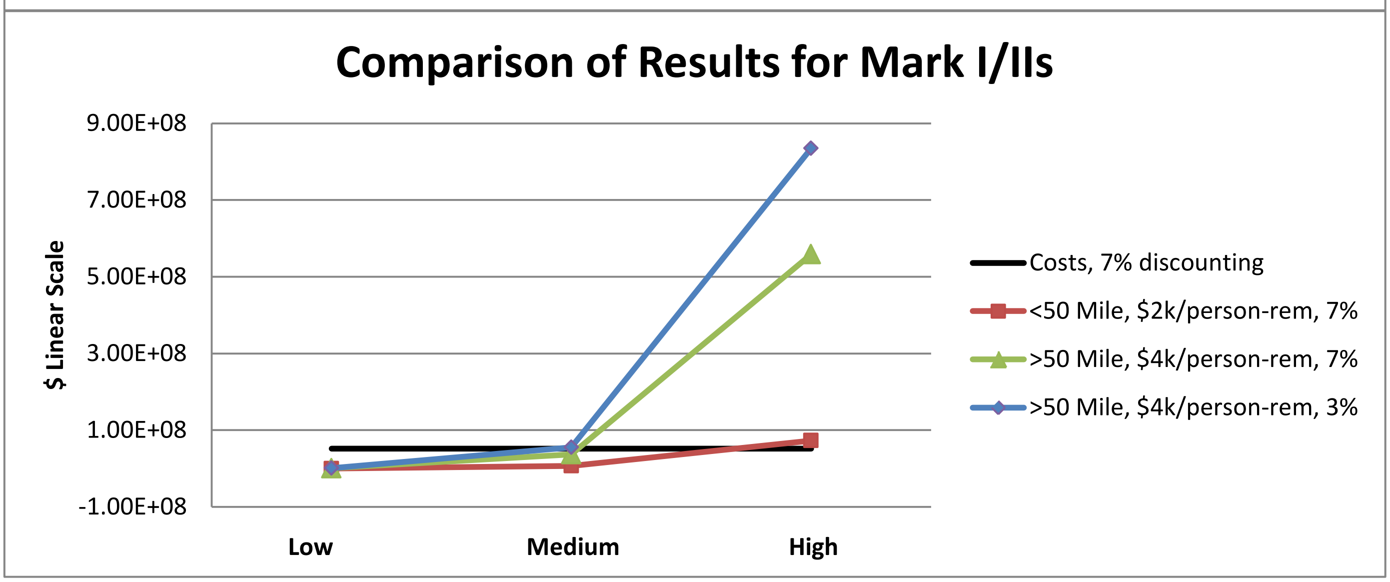
SEE SECTION E FOR IMPLEMENTATION GUIDANCE

9/18/13 version of RA

Group 1		Costs, 7% discounting	<50 Mile, $2k/person-rem, 7%	>50 Mile, $4k/person-rem, 7%	>50 Mile, $4k/person-rem, 3%
Low Estimate	1	5.23E+07	1.31E+05	9.47E+05	1.42E+06
Medium Estimate	2	5.23E+07	7.01E+06	3.71E+07	5.54E+07
High Estimate	3	5.23E+07	7.31E+07	5.59E+08	8.35E+08

Costs are relatively insensitive to discounting.

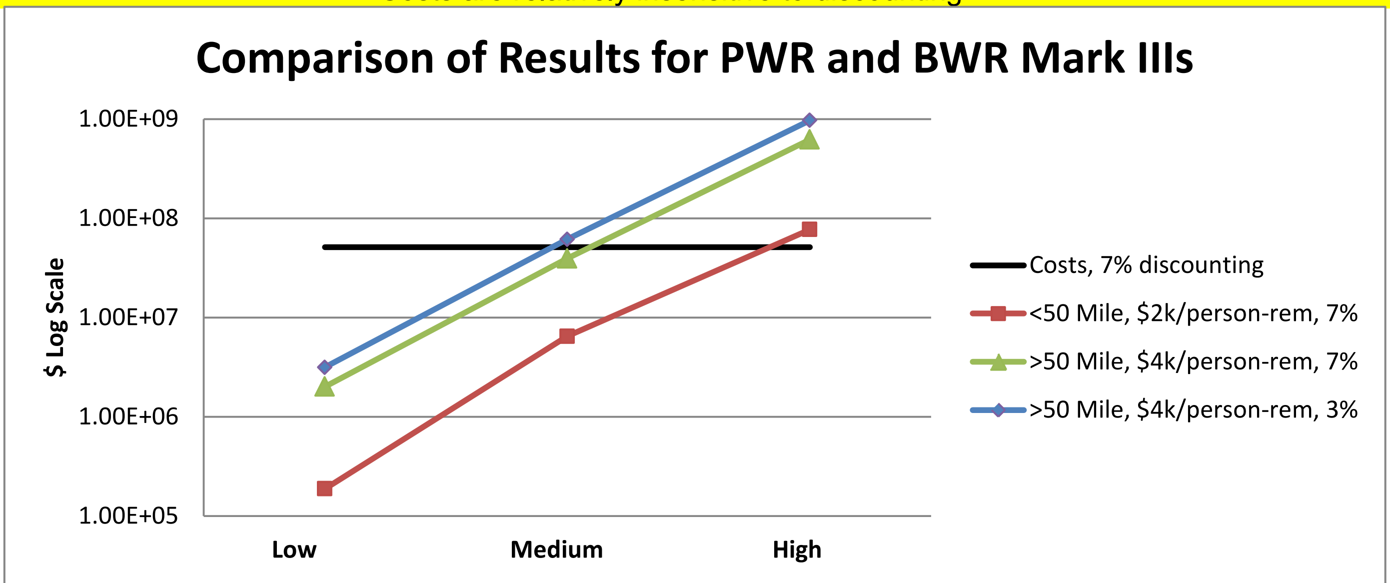


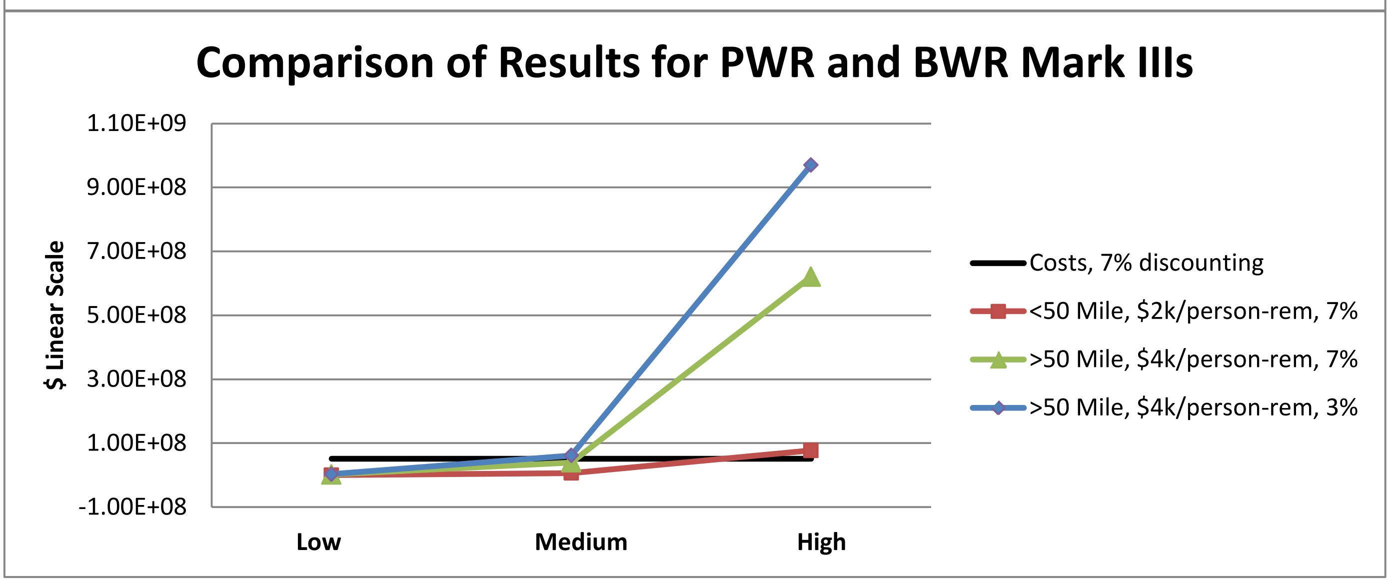


9/18/13 version of RA

Group 2		Costs, 7% discounting	<50 Mile, $2k/person-rem, 7%	>50 Mile, $4k/person-rem, 7%	>50 Mile, $4k/person-rem, 3%
Low Estimate	1	5.14E+07	1.88E+05	2.02E+06	3.15E+06
Medium Estimate	2	5.14E+07	6.48E+06	3.94E+07	6.16E+07
High Estimate	3	5.14E+07	7.70E+07	6.21E+08	9.70E+08

Costs are relatively insensitive to discounting.

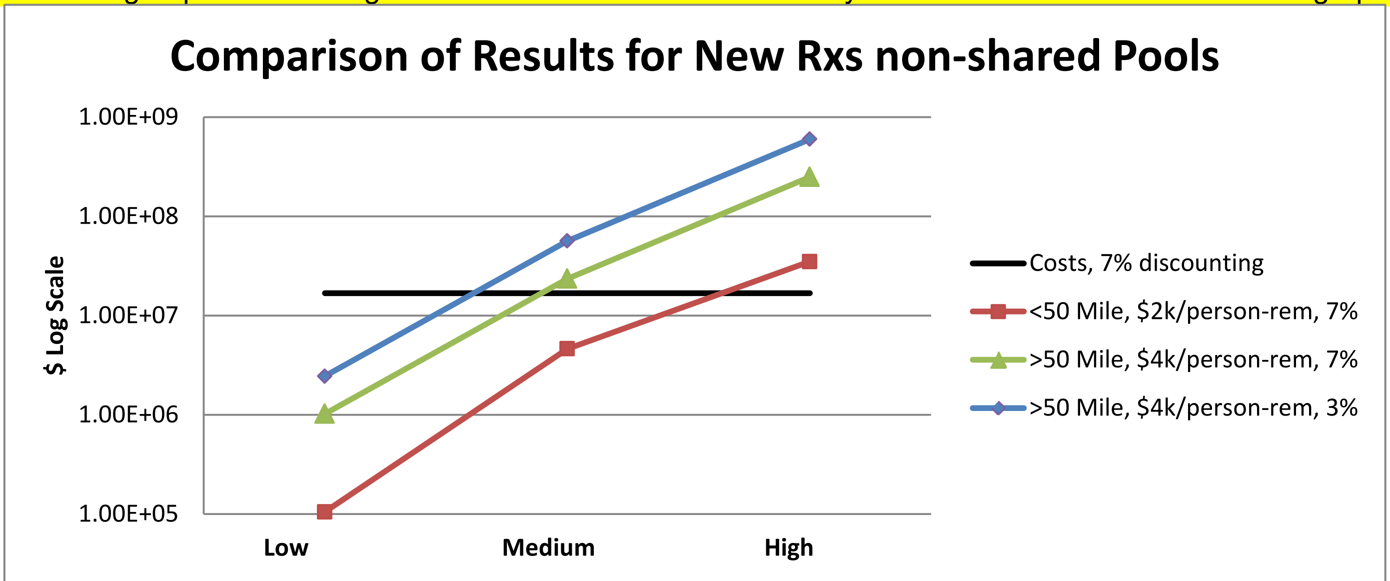


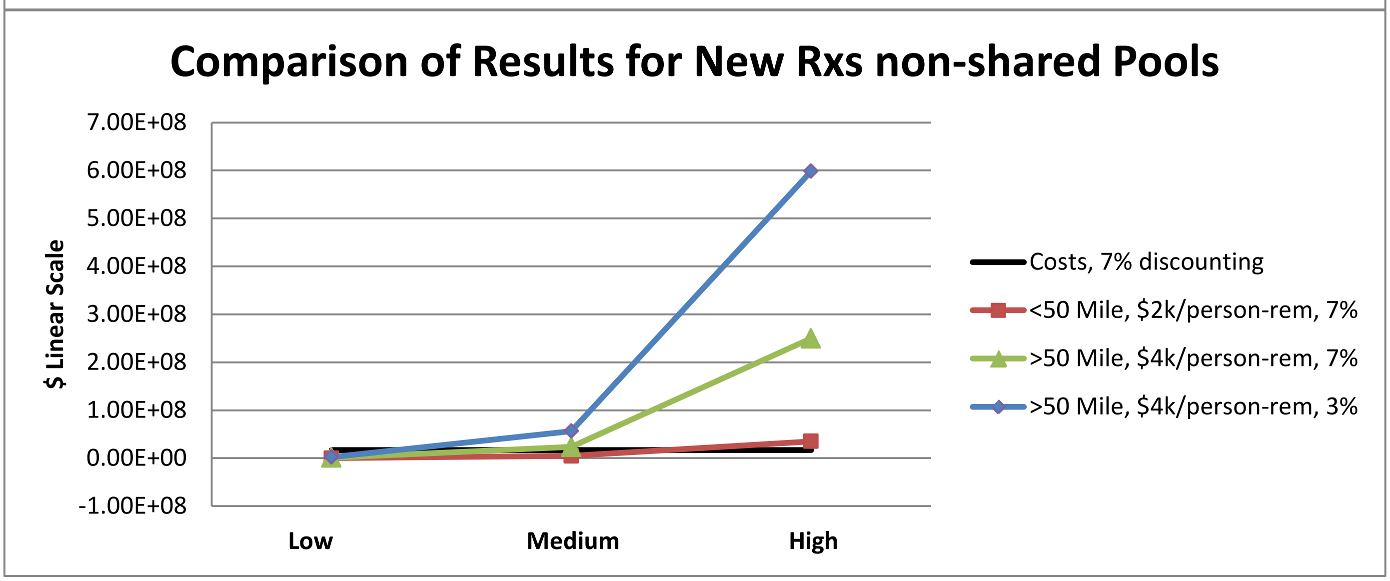


9/18/13 version of RA

Group 3		Costs, 7% discounting	<50 Mile, $2k/person-rem, 7%	>50 Mile, $4k/person-rem, 7%	>50 Mile, $4k/person-rem, 3%
Low Estimate	1	1.68E+07	1.05E+05	1.02E+06	2.45E+06
Medium Estimate	2	1.68E+07	4.63E+06	2.35E+07	5.64E+07
High Estimate	3	1.68E+07	3.49E+07	2.50E+08	5.98E+08

Note that for group 3 discounting costs at 3% would increase them by a factor of ~2. Not reflected in the graph.

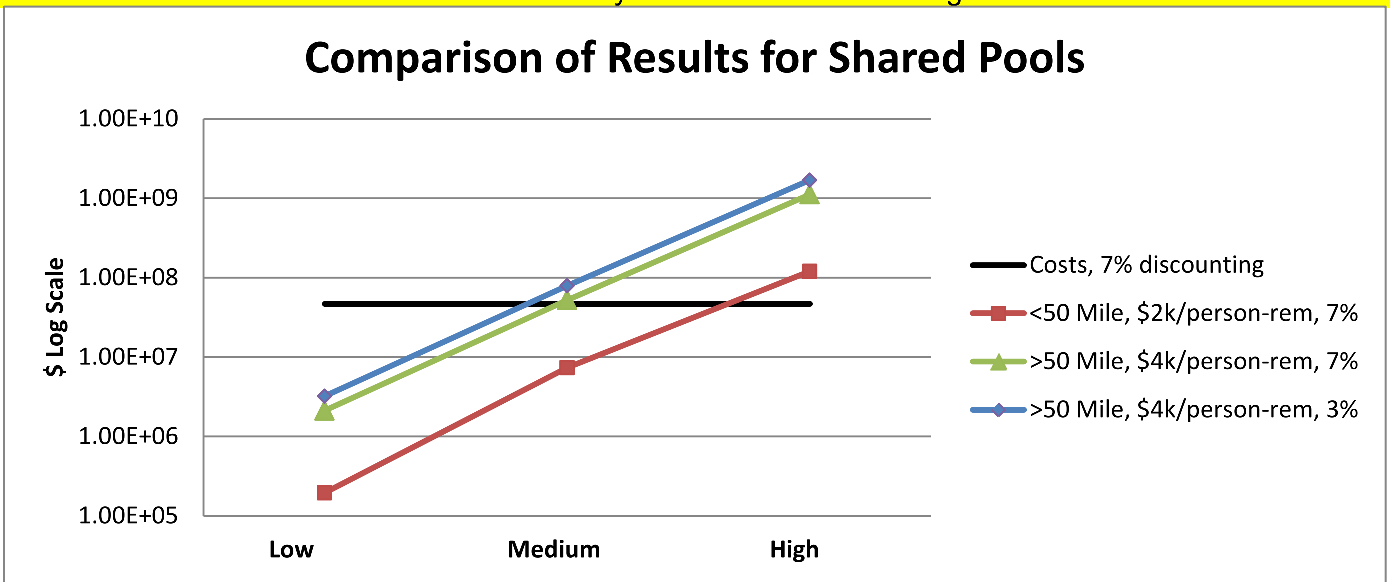


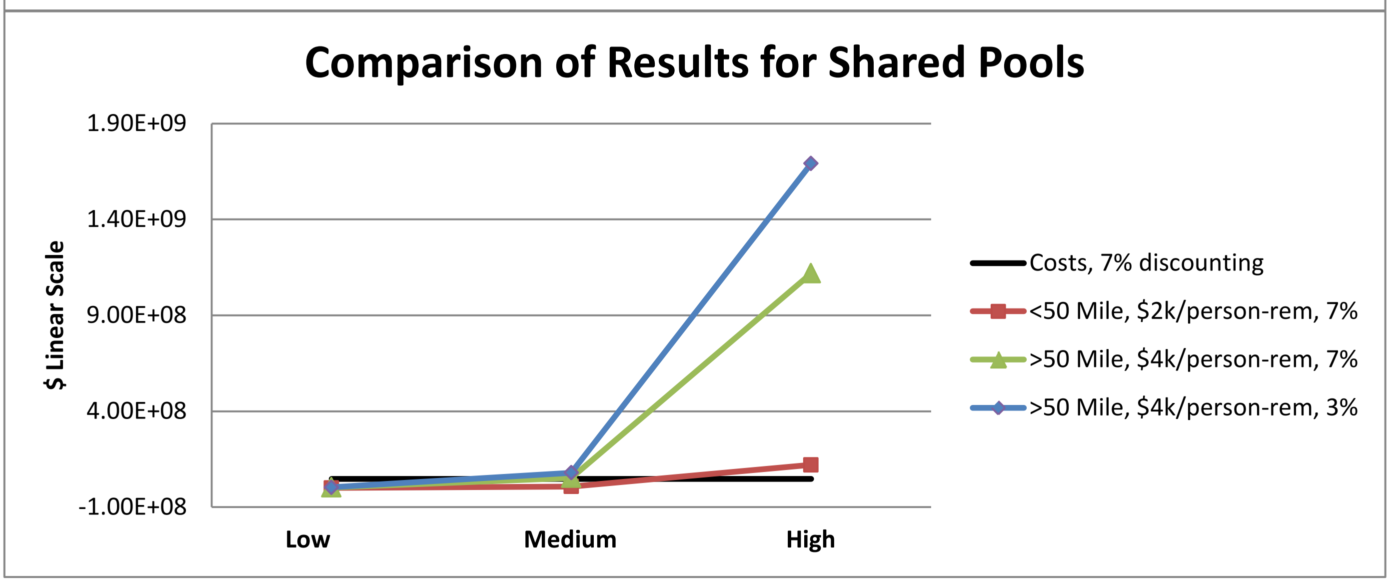


9/18/13 version of RA

Group 4		Costs, 7% discounting	<50 Mile, $2k/person-rem, 7%	>50 Mile, $4k/person-rem, 7%	>50 Mile, $4k/person-rem, 3%
Low Estimate	1	4.64E+07	1.95E+05	2.12E+06	3.21E+06
Medium Estimate	2	4.64E+07	7.33E+06	5.22E+07	7.89E+07
High Estimate	3	4.64E+07	1.20E+08	1.12E+09	1.69E+09

Costs are relatively insensitive to discounting.





Ratio of Benefits to Costs, 7% discounting

		Group 1	Group 2	Group 3	Group 4
$2000/person-rem <50 miles	Low	0.003	0.004	0.006	0.004
	Medium	0.134	0.126	0.276	0.158
	High	1.398	1.499	2.085	2.582

$4000/person-rem >50 miles	Low	0.018	0.039	0.061	0.046
	Medium	0.709	0.767	1.405	1.124
	High	10.680	12.089	14.901	24.085

Benefit/Cost < 0.1	Very not beneficial
0.1 < Benefit/Cost < 0.5	Not beneficial
0.5 < Benefit/Cost < 2	Borderline
2 < Benefit/Cost <10	Cost beneficial
10 < Benefit/Cost	Very beneficial

NRC FORM 757
NRC MD 10.158
(7-2011)

U.S. NUCLEAR REGULATORY COMMISSION

NON-CONCURRENCE PROCESS

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TITLE OF SUBJECT DOCUMENT
Staff Evaluation and Recommendation for Japan Lessons-Learned Tier 3 Issue on Expedited Transfer of

ADAMS ACCESSION NO.
ML13256A348

SECTION B - TO BE COMPLETED BY NON-CONCURRING INDIVIDUAL'S SUPERVISOR

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ORGANIZATION
Division of Risk Analysis, Office of Nuclear Regulatory Research

COMMENTS FOR THE NCP REVIEWER TO CONSIDER

Comments of Supervisor, D. Coe, Deputy Director, Division of Risk Analysis, RES
The regular supervisor is not available for the foreseeable future, therefore as the next higher supervisor who has been engaged in dialog with several staff and managers over these various issues, I am providing Section B comments in lieu of the regular supervisor. The following comments are not intended to either approve or disapprove of the points made in the non-concurrence.

This non-concurrence aims to improve clarity and understandability through use of plain language and visual/graphical forms to communicate the Regulatory Analysis (RA) and its assumptions. My recent discussions with the document sponsors and other concurring officials have consistently supported improving the clarity and transparency of the RA results within the time constraints available to do so. Through this dialog and the points made in this non-concurrence, it appears to me that important assumptions in the RA that were used to support the staff conclusions, such as the use of $2000 per person-rem-avoided and limiting the consequence analysis to a 50 mile radius from a site, and the cost-benefit implications of using higher values for both, have been and will be made more clear and that this should aid in understanding the basis for the RA results regarding cost-benefit. A more important issue raised in this non-concurrence, I believe, is the use of the QHOs.

The staff has compared the calculated health risks from spent fuel pools to the Quantitative Health Objectives (QHOs) and concluded that a substantial safety enhancement is not achieved by expediting spent fuel transfer to dry storage. Even if a RA were to demonstrate such a proposed backfit were potentially cost-beneficial, a backfit may not be justified due to the former conclusion. Simply stated, a slow accident progression if one should occur suggests a high confidence of evacuating the public. Coupled with a low probability of an accident, this reduces the estimated public health risk to substantially less than the QHOs even if reducing that risk further can be shown to be potentially cost-beneficial. The separateness of the evaluations for public health risk and cost-benefit appears consistent with the Commission's recent affirmation (SRM-SECY-12-0110) that it intends to continue to keep separate the "regulatory character" of adequate protection of public health and safety from the regulatory character of economic consequences. However, using the QHOs for this purpose requires consideration of several factors.

The use of the QHOs for this type of determination may not be the only possible quantitative risk approach, so any Commission-endorsed use here may set a significant precedent and should therefore be carefully considered. Toward that end, recent dialog within the staff has sought to understand the basis for the QHOs, including the reactor-centric nature of the QHOs and justification for extending their use to non-reactor accidents, their emphasis on risk from direct exposure to releases during the accident at specific distances from the accident site (1 mile and 10 miles), the apparent intent that the QHOs represent the maximum allowed total risk from a site (versus their use as a threshold criteria for individual specific accidents), and any past staff or Commission use of QHOs in a manner that may bear on the present case. If such factors have been duly considered, then it may still be possible to use the QHOs for determining whether a proposed backfit is a substantial safety enhancement. As noted, other quantitative risk approaches may be possible (for example, since we have been able to define an acceptable maximum 'small' increase in risk, i.e. RegGuide 1.174, it seems that we might similarly be able to define an acceptable minimum 'substantial' decrease in risk). Given the importance of such precedents for backfit decisions, I believe the ongoing dialog is a healthy contribution to this decision process.

☐ CONTINUED IN SECTION D

SIGNATURE [signature]

DATE 9/30/13

SEE SECTION E FOR IMPLEMENTATION GUIDANCE

NRC FORM 757
NRC MD 10.158
(7-2011)

U.S. NUCLEAR REGULATORY COMMISSION

NON-CONCURRENCE PROCESS

NCP TRACKING NUMBER: 2013-013

TITLE OF SUBJECT DOCUMENT: Staff Evaluation and Recommendation for Tier 3 Issue on Expedited Transfer of Spent Fuel

ADAMS ACCESSION NO.: ML13273A572

SECTION C - TO BE COMPLETED BY DOCUMENT SPONSOR

NAME: David Skeen

TITLE: Director

PHONE NO.: (301) 415-3091

ORGANIZATION: Office of Nuclear Reactor Regulation - Japan Lessons Learned Project Directorate (NRR/JLD)

SUMMARY OF ISSUES

SEE ATTACHED

ACTIONS TAKEN TO ADDRESS NON-CONCURRENCE

SEE ATTACHED

SIGNATURE--DOCUMENT SPONSOR | TITLE: Deputy Director

ORGANIZATION: NRR/JLD | DATE

SIGNATURE--NCP REVIEWER | TITLE: Deputy Director

ORGANIZATION: Office of Nuclear Reactor Regulation | DATE

NCP OUTCOME

Non--Concurring Individual: ☐ CONCURS ☑ NON-CONCURS ☐ WITHDRAWS NON-CONCURRENCE (i.e., discontinues process)

AVAILABILTY OF NCP FORM

Non--Concurring Individual: ☑ WANTS NCP FORM PUBLIC ☐ WANTS NCP FORM NON-PUBLIC

☐ CONTINUED IN SECTION D

SEE SECTION E FOR IMPLEMENTATION GUIDANCE

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SECTION D: CONTINUATION PAGE

CONTINUATION OF SECTION ☐ A ☐ B ☑ C

As the document sponsor, I first want to commend the non-concurer for raising his views on the paper. The fact that the NRC has an established process for individuals to raise differing views, and that individuals feel comfortable and confident in exercising that process, is a reflection of the NRC's strong safety culture. In addition, the fact that the agency has historically demonstrated its ability to work collaboratively to address important issues such as these while recognizing that is not always possible nor appropriate to achieve unanimous consensus on such issues, reflects that we can effectively and reasonably balance our need to consider differing views against our needs to be a timely, effective, and consistent regulator.

As with any issue that involves both complex technical and regulatory aspects, it is not surprising that different views and perspectives will arise. The NRC places strong emphasis on considering those views as part of its decision making process and documenting its response to those views. The views expressed in this non-concurrence raise good questions which were considered in the development of the regulatory analysis and the Commission paper. Most of the views expressed are focused on prior Commission policy decisions and standard practices for how the NRC performs regulatory analyses. While good questions, they do not provide a compelling reason for the staff to deviate from Commission policy or past practice. Instead, as discussed in greater detail in the enclosure, the staff is confident that each of the concerns is addressed in the regulatory analysis through the use of bounding or conservative assumptions and/or sensitivity studies. In addressing the concerns, the staff has included additional information in both the body of the paper and its enclosure to provide the Commission with the information necessary to make an informed decision as well as providing the Commission with a copy of the non-concurrence and staff response.

A detailed response to each of the non-concurrence items is provided in the attachment.

SEE SECTION E FOR IMPLEMENTATION GUIDANCE

Non-Concurrence Process Documentation

NCP-2013-013; Section C (Document Sponsor)

Summary of Issues

In SECY-11-0137, “Prioritization of Recommended Actions to be Taken in Response to Fukushima Lessons Learned” (ADAMS Accession No. ML11272A111), the staff identified the expedited transfer of spent fuel to dry cask storage as an issue warranting further consideration as part of the activities following the accident at the Fukushima Daiichi nuclear facility in Japan. The staff prioritized this issue in the Tier 3 category and said it requires further study to determine if regulatory action is warranted. In SECY-12-0095, “Tier 3 Program Plans and 6-Month Status Update in Response to Lessons Learned from Japan’s March 11, 2011, Great Tohoku Earthquake and Subsequent Tsunami” (ADAMS Accession No. ML12165A092), the staff described a plan involving various stages of assessments to help determine if regulatory action is warranted for the expedited transfer of spent fuel from spent fuel pools (SFPs) into dry cask storage. In a memorandum to the Commission entitled, “Updated Schedule and Plans for Japan Lessons-Learned Tier 3 Issue on Expedited Transfer of Spent Fuel,” dated May 7, 2013 (ADAMS Accession No. ML13105A122), the staff outlined updated plans involving three possible phases of evaluations to determine if regulatory action is warranted to require licensees to expedite transfer of spent fuel from SFPs to dry cask storage. The staff aligned the ongoing research activities related to the report entitled, “Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor,” dated October 2013 (SFP study; ADAMS Accession No. ML13256A342) with the previously established Tier 3 program plan while considering the schedule to support the agency’s ongoing waste confidence efforts. The staff’s objective with this integration was to facilitate the public’s involvement in these activities and related policy issues.

The result of the Tier 3 effort was a COMSECY and related analyses providing the first (Phase 1) evaluation to determine if additional studies were warranted. That COMSECY and enclosed analysis (referred to as a “regulatory analysis” in previous draft) is the subject of the NCP-2013-013. The stated purpose of the COMSECY is:

> The purpose of this memorandum is to provide the Commission with information and a recommendation on whether additional study is warranted to assess possible regulatory action to require expedited transfer of spent fuel from nuclear power plants’ spent fuel pools to dry cask storage. The staff’s assessment, as documented in this memorandum, concludes that the expedited transfer of spent fuel to dry cask storage would neither provide a substantial increase in the overall protection of public health and safety nor sufficient safety benefit to warrant the expected implementation costs. Therefore, the staff recommends that no further generic assessments be pursued related to possible regulatory actions to require the expedited transfer of spent fuel to dry cask storage and that this Tier 3 Japan lessons learned activity be closed.

The issues raised in the nonconcurrence relate to the presentation of the information being provided to the Commission and also to the recommendation that no further generic assessments be pursued. In preparing the paper and recommendation, the staff followed the normal processes of routinely meeting with the Japan Lessons Learned Steering Committee and ensuring alignment between the Executive Director for Operations and management of the various NRC offices contributing to or affected by the assessment and recommendations. Additional discussion and the action taken for each of the points raised in NCP-2013-013 are provided below.

Specific Issue Summaries and Actions Taken

1. *Contrary to NUREG/BR-0058, "Regulatory Analysis Guidelines of the USNRC" guidance which recommends that "the range of all potentially reasonable and practical approaches to the problem are considered," only a single alternative is considered. Other alternatives may be more cost beneficial. For example, transferring less fuel or discharging into a 1x8 pattern may yield the same benefits while costing significantly less than the analyzed alternative. Both the draft Spent Fuel Pool Study (ML13133A132) and the ACRS letter (ML13224A060) recommended further analysis of the 1x8 fuel pattern. The draft COMSECY transmitting the regulatory analysis claims this would not provide a substantial safety enhancement despite it not being analyzed (or even mentioned) in the attached regulatory analysis.*

Summary/Discussion

The Tier 3 Program Plan provided in SECY-12-0095 and the memorandum to the Commission dated May 7, 2013 describe the issue being evaluated as whether or not to pursue additional studies to help determine if regulatory actions should be taken to require the expedited transfer of spent fuel from SFPs to dry cask storage. This question arises from the premise that reducing the amount of spent fuel in SFPs will reduce the heat load in the pools if there is a loss of heat removal or inventory and reduce the amount of radioactive material that might be released if a lack of cooling results in a zirconium fire within the SFP. The purpose of the Tier 3 activity, supported by the SFP study and previous studies, was to assess if there was a reasonable likelihood that additional studies and more refined assessments would support future regulatory actions to require expedited transfer of spent fuel or if the current regulatory requirements were likely to be deemed sufficient for protection of public health and safety and protection of the environment. The focus of the assessment and the COMSECY was therefore on the safety benefits of expedited transfer of spent fuel, resulting in plants moving from high density loading patterns to low density loading patterns in SFPs.

The SFP study and previous assessments did identify the possible benefits of changing the arrangement of spent fuel while keeping a high density loading within the SFP (e.g., going from the current 1x4 pattern to a 1x8 pattern). The SFP study also identified possible enhancements to mitigating capabilities (e.g., makeup or sprays to the SFP) to address specific periods of time when the heat load was higher due to recently discharged spent fuel assemblies. While these types of actions were not the primary focus of the COMSECY and related analysis, the staff provides the following discussion within the memorandum:

> In addition to assessing whether further studies of expedited transfer of spent fuel to dry cask storage are warranted, the SFP study and staff's interactions with stakeholders identified other possible improvements to the storage of spent fuel. Examples include the possible investigation of alternate loading patterns (e.g., the 1 x 8 high-density loading pattern assessed in the SFP study, in addition to the standard 1 x 4 high-density loading pattern), capability of licensees to directly offload fuel into more coolable patterns, and the possible enhancement of mitigation strategies during identified periods when the heat load from recently discharged fuel assemblies is especially high. The staff has taken note of these possible improvements, but determined that they do not provide a substantial safety enhancement warranting generic regulatory action. This finding reflects the low probability of the initiating events that would challenge the integrity of the spent fuel pools and the fact that these alternative actions would have similar or

> lesser safety benefit as that estimated for the expedited transfer of spent fuel. So even though these alternatives would likely involve lower costs than the expedited transfer of spent fuel to dry cask storage, the staff finds that they do not satisfy the criterion for a substantial safety enhancement. However, licensees will be informed of and encouraged to assess and implement, as appropriate, such improvements on their own initiative to help manage the risks associated with plant specific SFP designs, operating practices, and mitigation capabilities.

The staff did not present a detailed assessment supporting the above conclusion or proceed, as it did for the primary topic of expedited transfer of spent fuel (comparing high-density to low-density loadings), to more detailed backfit and regulatory analyses. The above discussion was expanded slightly to address the comment but more generally relies on the observation that the estimated low frequencies for the potential conditions in which these actions would be beneficial would result in a similar finding as for expedited transfer – that additional studies would be unlikely to support a conclusion that possible regulatory actions would provide a substantial safety improvement.

Action

The staff had previously noted the possible alternatives in the COMSECY and revised the paragraph to address the issues raised in NCP-2013-013. The discussion was expanded slightly to refer to the estimated low frequencies of initiating events as the major contributor to the likely conclusion that these alternatives would not be a substantial safety increase. The discussion was also revised to mention the lower costs of these alternatives compared to expediting the transfer of spent fuel to dry casks.

Conclusion

The schedule for preparation of the COMSECY was coordinated and aligned with the comment period for the environmental impact statement related to the Waste Confidence Decision to ensure its availability to members of the public providing comments on the environmental impact statement. The staff could have performed additional assessments of these alternatives in terms of the costs and benefits to include in the paper. The lower costs of these alternatives may have resulted in the calculated benefits exceeding the estimated costs. However, the additional detailed evaluations would not have changed the conclusion that these actions would not constitute a substantial safety improvement in accordance with the NRC's backfit requirements. Beyond the actions taken to clarify the limited assessment performed, no additional changes or delays in providing the paper to the Commission are needed.

2. *The regulatory analysis uses $2000/person-rem as the baseline. It is known that a change in guidance is imminent that would change this value to the $4000-$5000/person-rem range to be more consistent with the practices of other agencies.*

Summary/Discussion

In the Staff Requirements Memorandum (SRM) for SECY-12-0110, “Consideration of Economic Consequences Within the U.S. Nuclear Regulatory Commission’s Regulatory Framework,” the Commission directed:

> The Commission has approved the staff’s recommended Option 2, to enhance the currency and consistency of the existing framework through updates to guidance documents integral to performing cost-benefit analyses in support of regulatory, backfit, and environmental analysis, subject to the following comments and additional direction. The staff should identify the potential changes to current methodologies and tools that would enhance regulatory analysis guidance under current Option 2 in a comprehensive paper on Option 2 implementation so it is clear how Option 2 “would help harmonize regulatory guidance across the agency” in both the reactor and materials programs arenas. The development of implementation approaches for Option 2 will likely expose policy issues (e.g., use of a particular decontamination level) during the staff’s efforts to improve guidance for estimating offsite economic costs or to identify potential areas to develop new guidance, as needed, for other regulatory applications, and these issues should be brought to the Commission for review and approval. Given this, the Option 2 paper should be a notation vote paper. However, the staff may continue with ongoing staff activities described in SECY-12-0110 to update guidance documents (i.e., an update to NRC’s dollar per person-rem conversion factor policy and an update to replacement energy costs).

In addition, the Commission stated:

> The staff should provide to the Commission any cost benefit model developed for use in guidance documents to address offsite property damage costs. This would include any proposed methodology for changing the calculated value of averted dose referenced in NUREG-1530 [Reassessment of NRC’s Dollar Per Person-Rem Conversion Factor Policy (December 1995)].

The staff is aware that the traditional $2000 per person-rem conversion factor may change as a result of ongoing assessments. However, the value of the dollar per person-rem conversion factor is a matter of Commission policy. Therefore, the staff followed the established processes and guidance and addressed the possible change in the conversion factor via sensitivity studies included in the enclosure to the COMSECY (see Section 4.4.1.4.1, Tables 19-22 and Appendix D). Sensitivity studies allow the decisionmakers to take into account the contributions of offsite doses to the cost/benefit assessments and to see the differences that result from an increase in the conversion factor from $2000 to $4000 per person-rem.

Actions

To clarify the discussion and highlight the importance of the dollars per person-rem conversion factor, the staff added the following sentences to the COMSECY:

> … Sensitivity studies were also conducted on key factors such as the dollars per person-rem conversion factor and consideration of consequences beyond 50 miles to measure each attribute's effect upon the overall result. The sensitivity of the dollars per person-rem conversion factor is important to consider because related guidance is currently being updated. The sensitivity of consequences beyond 50 miles is important to consider for accidents involving SFP fires as the spread of radioactive materials could extend over long distances. …

Conclusion

While it is a valid observation that the current $2000 per person-rem conversion factor is currently being reevaluated and might be revised should the Commission decide to do so in the future, it is appropriate to follow the current processes and guidance as directed by the Commission and address the issue through inclusion of sensitivity studies. The additional discussion of this issue in the COMSECY is an improvement and provides the needed information to the Commission for their deliberations.

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3. *The regulatory analysis uses a 50-mile truncation as a baseline. Guidance in NUREG/BR-0058 indicates that a 50 mile truncation should be used for nuclear power plants but that the appropriate distance for other facilities should be decided on a case-by-case basis. For SFP accidents in high density pools, which are expected to release much more material than reactor accidents, this truncation can decrease the calculated consequences by nearly a factor of 10. This truncation is arbitrary and technically indefensible.*

Summary/Discussion

The COMSECY and its enclosure point out that accidents involving spent fuel pool fires have the potential to involve releases that would contaminate much larger areas than are calculated for most reactor accidents. This potential is also a major point made by the SFP study and other references in the regulatory analysis (e.g., NUREG-1738, "Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants"). The Tier 3 assessment addresses this issue by including sensitivity studies in the enclosure provided with the COMSECY (see Section 4.4.1.4.2, Tables 23-26, and Appendix D). Sensitivity studies allow the decisionmakers to take into account the possible contamination of larger land areas within the cost/benefit assessment.

In terms of the assessment and primary basis for the staff's recommendation, the sensitivity studies reflecting both the increase in dollars per person-rem and impacts beyond 50 miles are addressed in Section 4.4.1.4.3 and are shown in Tables 27 through 30 in the enclosure provided with the COMSECY. A summary is provided below for the base case assessments for each grouping of spent fuel pools at the traditional discount factors (3% and 7%) considered for regulatory analyses:

Group	Net Benefit 3% Discount Factor	Net Benefit 7% Discount Factor
1 (BWRs - Mark I/II)	+173,000	-15,265,000
2 (PWRs, BWRs - Mark III)	+7,693,000	-12,007,000
3 (New Reactors)	+20,637,000	+6,785,000
4 (PWR Shared Pool)	+28,457,000	+5,762,000

To help place the discussions in context, the majority of plants are in either Group 1 or 2. In addition, the staff has typically put more weight on the cases performed at the 7% discount rate when assessing costs and benefits, consistent with OMB Circular A-94, and uses other discount rates to indicate the sensitivity of the results to the choice of discount rate. An additional point (to be discussed further below) is that the assessments included in the COMSECY and related analyses were performed to help determine if additional studies should be undertaken and not, per se, to justify proposing a regulatory requirement. With this in mind, the staff made some simplifying and generally conservative assumptions that tend to increase the calculated benefits of moving to low density SFP loadings. These assumptions and their influence on the calculations need to be considered when looking at the results and using them in making a recommendation on the likely outcome of additional studies on this topic. In the more realistic studies envisioned if the Commission were to direct the staff to proceed to Phase 2 of Tier 3 plan, the staff would revisit assumptions such as crediting mitigation for low density pools and not for high density pools and other factors that inflated the benefits of expediting the transfer of spent fuel. In addition, Phase 2 assessments would include the additional risks introduced by additional cask loadings and the risks associated with dry cask storage. The recommendation that additional studies of this issue are unlikely to result in future regulatory actions is based on the calculated results included in the sensitivity studies but with full consideration of the assumptions and general approach taken for these assessments.

Actions

To clarify the discussion and highlight the importance of the possible contamination of larger land area, the staff added the following sentences to the COMSECY:

> … Sensitivity studies were also conducted on key factors such as the dollars per person-rem conversion factor and consideration of consequences beyond 50 miles to measure each attribute's effect upon the overall result. The sensitivity of the dollars per person-rem conversion factor is important to consider because related guidance is currently being updated. The sensitivity of consequences beyond 50 miles is important to consider for accidents involving SFP fires as the spread of radioactive materials could extend over long distances. …

Conclusion

The presentation of the information in the COMSECY and regulatory analysis is a reasonable way to ensure the Commission has the needed information and is made aware of the issues that influence the assessments and recommendation. The additional discussion of this issue in the COMSECY is an improvement and provides the needed information to the Commission for their deliberations.

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4. *The SECY paper and regulatory analysis argues that no further action is necessary since the alternative does not represent a substantial safety enhancement. It is not clear how this position reconciles with the SRM to SECY-93-086, which states that the substantial standard "is not intended to be interpreted in a manner that would result in disapprovals of worthwhile safety or security improvements having costs that are justified in view of the increased protection that would be provided." The substantial safety enhancement screen should not be used to dismiss cost-beneficial results or as a reason to not compute cost-benefit information for other reasonable alternatives.*

Summary/Discussion

The safety goal screening evaluation, as outlined in the regulatory analysis guidelines (NUREG/BR-0058), is designed to answer when a regulatory requirement should not be imposed generically on nuclear power plants because the residual risk is already acceptably low. This evaluation is intended to eliminate some proposed requirements from further consideration independently of whether they could be justified by a regulatory analysis on their net value basis. The safety goal evaluation can also be used for determining whether the substantial added protection standard of 10 CFR 50.109(a)(3) is met. However, the guidance is not intended to remove all flexibility and judgment from the backfit process and therefore points out that the safety goal screening evaluation is not intended to block worthwhile safety or security improvements that would otherwise be found to be cost-beneficial. Use of this guidance therefore requires a judgment by the NRC staff and Commission as to whether the safety goal screening evaluation provides an unreasonable finding on whether a proposed action provides a marginal or substantial safety improvement. In this case, the staff finds and includes in the COMSECY that the safety goal evaluation identifies safety improvements as marginal and that this finding is consistent with previous studies and the prevailing view of the staff. However, even though the staff finds that a possible requirement to expedite the transfer of spent fuel does not meet the safety goal screening evaluation as a substantial safety improvement, the staff prepared and provides to the Commission for their consideration an analysis of the cost/benefits in terms of the NRC's backfit process and within a broader regulatory analysis.

As previously mentioned, the staff did not prepare or provide to the Commission cost/benefit assessments of other possible regulatory actions, such as requiring alternate loading patterns within a high density pool or requiring enhancements to accident mitigation capabilities. In these cases, the staff finds that if the major action is a marginal safety improvement, similar or lesser actions would likewise not provide a substantial safety improvement.

Actions

A discussion was added to Section 3 of the enclosure, which deals with the QHO screening evaluation as a test of substantial safety enhancements. Section 3 of the enclosure is specifically mentioned in the COMSECY during the discussion on QHOs.

Conclusion

The staff followed established processes and guidance and provided their findings to the Commission for consideration. Despite finding that the expedited transfer of spent fuel does not constitute a substantial safety improvement, the staff prepared and provided cost/benefit assessments to support Commission deliberations on this issue. The additional discussion of this issue in the COMSECY is an improvement and provides the needed information to the Commission for their deliberations.

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5. *The regulatory analysis answers the substantial safety enhancement question by comparing to the Quantitative Health Objectives (QHOs) found in the Safety Goal Policy Statement. Though this is standard practice, the QHOs were developed for reactor accidents and are not well suited for making this determination for SFP accidents. SFP accidents in high density pools can lightly contaminate very large areas, displacing millions of people and requiring extensive protective actions. Conversely, the individual LCF risk from 0-10 miles is relatively low, even for the largest releases. SFP releases would have to occur with a frequency greater than 10^{-3} to approach the safety goals (100x higher than the Large Early Release Frequency subsidiary objective used for reactors.) While an alternative measure of a substantial safety enhancement is not readily available, one informative metric is that, for some "high estimate" cases, the proposed alternative results in nearly a billion dollars in frequency-weighted safety benefits. The SECY paper should acknowledge the significant limitations of applying the QHOs to non-reactors to provide The Commission with relevant information to inform their decision.*

Summary/Discussion

The staff presented the safety goal evaluation and cost/benefit assessments as part of the backfit and broader regulatory analyses because SFPs are part of nuclear power facilities and therefore contribute to the overall risk for which the safety goals and quantitative health objectives were formulated. It is true that the focus of the safety goal policy statement and the subsidiary criteria (e.g., core damage frequency and large early release frequency) are primarily related to reactor accidents. However, SFP accidents are similar enough to reactor cores (as compared to other types of NRC regulated materials and facilities) to be covered by much of the guidance developed for reactor accidents. The safety goal evaluation was also used in previous NRC assessments of SFP issues (e.g., NUREG-1353 and NUREG-1738). Particular issues and differences related to using the safety goal screening evaluation for SFP accidents, such as the possible contamination of larger land areas, were addressed in the paper by providing sensitivity studies.

In addition to the staff's general belief that the safety goal evaluation is an appropriate tool for this assessment, the Commission has provided direction to the staff on developing new or different criteria for evaluating costs/benefits and the treatment of economic consequences in regulatory decisions. In the Staff Requirements Memorandum (SRM) for SECY-12-0110, the Commission directed:

> The identification of new areas to develop guidance for other regulatory applications under Option 2 should be limited and should be resourced as a lower priority than activities under Option 2 associated with applying SOARCA insights and improving guidance and analysis tools (such as the MACCS2 computer code) based on up-to-date data and advancements in accident consequence assessment knowledge.
>
> The staff should provide the Commission with a regulatory gap analysis prior to developing new guidance for application across business lines (e.g., materials, fuel cycle facilities, or emergency preparedness).

Actions

To clarify the discussion and highlight the possible issues related to the use of the safety goal screening evaluation as part of the staff's finding that possible regulatory actions for SFPs are unlikely to result in substantial safety improvements; the following discussion was added to Section 3, "Substantial Safety Enhancement Evaluation," in the analysis that is enclosed to the COMSECY:

> Comparing this analysis results to the NRC Safety Goal Policy Statement involves important limitations. First, the safety goal is intended to encompass all accident scenarios on a nuclear power plant site, including those involving reactors and spent fuel. This analysis does not examine reactor scenarios that would need to be considered, although the analysis does consider the most important contributors to SFP risk. As a result, comparison of the calculated individual latent cancer fatality (LCF) risk to the NRC Safety Goal Policy Statement is incomplete. However, it is intended to show that SFP risk is less than one percent of the individual LCF risk that corresponds to the safety goal for latent cancer fatalities. It is unlikely that the additional accident scenarios provided above would contribute significantly to a risk that would challenge the Commission's Safety Goal Policy Statement.
>
> The QHOs effectively establish expectations related to the frequency of severe accidents associated with nuclear reactors and the potential for release of radioactive materials from an operating reactor core. Previous NRC evaluations of SFPs, including NUREG-1353 and NUREG-1738, compared the estimated risks from SFP accidents to the QHOs as part of the rationale for determining appropriate regulatory actions. Some considerations in comparing SFP risks to the QHOs are that the potential consequences of a SFP accident can exceed those of reactor accidents in terms of the amount of long-lived radioactive material released, the land area affected, and the economic consequences. The safety goal relates to the risks to an individual from nuclear power in comparison to other risks that an individual faces. The staff uses the safety goal in regulatory decisionmaking processes as a measure of health consequences to determine if a potential action provides a substantial safety improvement. Although a SFP accident might affect larger areas and more people than a reactor accident, the risks to individuals remains bounded by the assessment of the population close to the

facility. For this reason, the staff uses the existing QHOs for determining whether the substantial safety enhancement threshold is met.

The significant difference between the calculated consequences of a SFP accident and a reactor accident has led some stakeholders to propose alternate performance measures to help in the decisionmaking process. Such measures could include a revised consideration of economic consequences, collective dose to populations, or other estimates that reflect the large consequences and reduce the influence of the low event frequencies and implementation of protective actions in assessing the overall societal risks associated with SFP accidents. However, the Commission has previously directed that these performance measures should be consistent with the overall safety goals the Commission policy established and should not be so conservative that it creates a de facto new policy.[1]

The development of surrogate measures for SFPs could be useful if the conditional probability of a significant SFP accident is very high for particular event scenarios (a so-called cliff-edge effect). Although the staff has used various conservative assumptions in this assessment in order to estimate the potential benefits of reducing the density of spent fuel stored in pools, the expected ability of pools to retain their integrity and the availability of mitigation capabilities leads the staff to conclude that exceeding design basis values associated with SFPs are unlikely to result in such a cliff-edge effect and that the frequency of damage to stored fuel is appropriately low to satisfy overall societal risk goals. Therefore, the staff has not identified this as an area for which it needs to develop new methodologies, guidance, or criteria. In the SRM for SECY-12-0110, "Consideration of Economic Consequences within the U.S. Nuclear Regulatory Commission's Regulatory Framework," the Commission directed the staff to proceed with improvements to the guidance for estimating offsite economic costs. The staff is continuing its efforts and planning related to the SRM and is scheduled to provide the Commission with a paper in December 2013. Factors considered likely to change as a result of the staff's activities (e.g., dollars per person-rem conversion factor) have been addressed in this evaluation through the presentation of additional cases and sensitivity studies.

The staff has concluded that the continued operation of nuclear power plants with high-density loadings in their SFPs does not challenge the NRC's safety goals or related QHOs. Therefore, a regulatory action to require reducing the inventory of spent fuel in the pools would not provide a substantial safety improvement. If the proposed regulatory action did not provide a substantial safety enhancement, the NRC's guidance would instruct the staff to stop the evaluation. In this case, although the staff determined that expedited transfer does not provide a substantial safety enhancement, the staff proceeded to perform a cost-benefit analysis to provide additional information to support the Commission's deliberations.

To address concerns that the discussion is buried within the regulatory analysis, a reference to this specific section and discussion is included in the COMSECY where the topic of using the QHOs is introduced.

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[1] Commission Guidance on Implementation of the NRC's Safety Goal Policy," memorandum from the Secretary of the Commission to the EDO, dated November 6, 1987.

Conclusion

The staff followed established processes, guidance and precedence established in previous evaluations of possible regulatory changes related to SFPs. The addition of more detailed discussion of this issue in the regulatory analysis is an improvement and provides additional information to the Commission for their deliberations.

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6. *Despite the fleet only being bounded by the high estimates (which are shown to be cost-beneficial) and not the base cases, the regulatory analysis concludes the alternative is not cost-beneficial.*

Summary/Discussion

The staff provided within the cost/benefit analysis a number of cases that consist of combinations of assumptions for various parameters or conditions. The calculations including parameters assumed to be at the lower end of their expected ranges were labeled "low estimate" cases. The calculations including conservative assumptions for most parameters (as is typical for generic regulatory assessments) were labeled as "base case." Another set of calculations using bounding assumptions (e.g., conditional failure probabilities of 100%) were labeled "high estimate" cases. The presentation of these cases was not intended to be taken as probability distributions for various parameters such that the high estimate cases actually represented some small number of plants. It is not surprising that the benefits outweigh the costs for high estimate cases given the bounding assumptions compounded upon each other and effectively increased the frequency of releases by more than an order of magnitude in comparison to the values used in Appendix D of the SFP study.

For the purpose of this assessment – which is supporting only a decision on whether or not resources should be spent on additional studies – the staff provided the high estimate cases for the Commission to consider in their deliberations. The low and high estimate case results were presented alongside the base case analysis results to provide an indication of the relative impacts of the assumptions that were made in the analysis. There is also an element of subjective judgment regarding the results of this assessment and the likely outcome of additional studies. If directed by the Commission to proceed with Phase 2 of the Tier 3 plan, the staff would revisit assumptions for various parameters and simplifications (e.g., crediting mitigation for low density pools and not for high density pools) that inflated the benefits of expediting the transfer of spent fuel. In addition, Phase 2 assessments would include the additional risks introduced by additional cask loadings and the risks associated with dry cask storage. The conclusion that additional studies of this issue are unlikely to result in future regulatory actions is based primarily on the base case calculations and the judgment of the staff regarding likely results of future studies.

Actions

To clarify the discussion and acknowledge the role of staff judgment in the assessment, the COMSECY was revised to:

> (Page 7) … Within the enclosed analysis, the staff provides a "base case" which generally used conservative assumptions for key parameters such as conditional probabilities of SFP liner failures and loss of adequate cooling to increase the calculated benefits of expedited transfer of spent fuel (i.e., to skew the calculations towards pursuing additional studies). The benefits calculated for the base case evaluations are less than the estimated costs for requiring expedited transfer of spent fuel to dry cask storage. Although the base case is used as the primary basis for the staff's recommendation, the staff also analyzed additional cases where key parameters are varied to provide low and high estimates of the calculated benefits. The staff used bounding or conservative values in the analysis for several parameters, particularly in the high estimate cases, to ensure that design, operational, and other site variations among the new and operating reactor fleet were addressed and to generally increase the calculated benefits from the proposed action. Sensitivity studies were also conducted on key factors such as the dollars per person-rem conversion factor and consideration of consequences beyond 50 miles to measure each attribute's effect upon the overall result. …
>
> (Page 8) … The cost-benefit analysis also includes sensitivity studies and some combinations of high estimates for important parameters resulting in large economic consequences such that, in some cases, the calculated benefits from expedited transfer of spent fuel to dry cask storage outweigh the associated costs (see Appendix D in enclosed supporting analysis). However, even in these cases, there is not a substantial safety improvement in terms of public health and safety. In the staff's judgment, it is unlikely that individual plants would meet or exceed the most conservative assumptions made in these sensitivity cases within the supporting analysis and the "base case" remains the primary basis for the staff's recommendation. Based on the generic assessment and the other considerations detailed in this paper, the staff finds that additional studies are not needed to reasonably conclude that the expedited transfer of spent fuel to dry cask storage would neither provide a substantial increase in the overall protection of public health and safety, nor sufficient safety benefit to warrant the expected implementation costs. Therefore, the staff finds that additional studies of expedited transfer of spent fuel is not needed.

Conclusion

The possible communication challenges associated with presenting the "high estimate" cases was raised by the staff and the ACRS. The role of judgment within the process needs to be considered and when combined with the conservative and simplifying assumptions, the COMSECY reflects the position of the NRC staff and management that additional studies would be unlikely to justify additional regulatory requirements for SFPs. The revised wording clarified the role of staff judgment in the assessment.

7. *Though the Regulatory Analysis contains an appropriate range of estimates and sensitivity results, both the "Decision Rationale" section of the regulatory analysis and the discussion of the results in the COMSECY transmitting the regulatory analysis fail to provide a balanced view of the range of results. There are several examples of this:*

    - *The COMSECY states that conservative assumptions are used in the regulatory analysis without making it clear that conservatives are primarily to account for variations within the group considered in the high estimates. The base case estimates represent a point estimate and contain a few minor conservatisms. The base case estimates do not bound the group of SFPs.*

    - *The COMSECY states "it is unlikely that individual plants would meet or exceed the most conservative assumptions made in these sensitivity cases within the regulatory analysis." This is highly misleading. The cases referenced are extremely cost-beneficial so a pool even approaching these assumptions would be very cost beneficial.*

    - *The "Decision Rationale" section of the regulatory analysis states there are other considerations discussed in Section 4.5.10 that would further decrease the benefits and make the proposed alternative less cost-justified. Though some of the items discussed would clearly decrease the benefits (e.g. credit for mitigation) others could increase or decrease the benefits. The list omits considerations which would increase the benefits such as relaxing the potentially optimistic assumptions that extensive protective actions are effective following a severe seismic event.*

    - *The analysis concludes that the alternative is not cost-beneficial by apparently focusing on the base case estimate truncated at 50 miles and using $2000/person-rem. Results that are cost-beneficial are downplayed as resulting from combinations of high estimates "sensitivity studies and some combinations of high estimates … such that, in a few cases, the benefits…appear to be cost beneficial." This is inconsistent with the results of the regulatory analysis which are: all high estimates are cost beneficial regardless of what other assumptions are used; and, when considering all consequences and an updated value of $4000/person-rem, all base cases are essentially cost neutral.*

    I have produced several figures and tables below to illustrate the results of the regulatory analysis. They paint a much muddier picture as to whether or not the alternative is cost-beneficial when compared to the COMSECY.

<u>Summary/Discussion</u>

The COMSECY and related analysis present the information mentioned above but in the context of the staff's assessment and the general conclusion that additional studies would be unlikely to lead to additional regulatory requirements for SFPs. Within the assessment, there are various uncertainties related to individual plants but the purpose of this activity is to provide a generic assessment on whether or not additional studies would likely lead to a rulemaking or other imposition of new requirements on licensees regarding SFPs. Much of the individual

points above were previously addressed and so the primary contention is assumed to relate to how the information is presented.

As discussed in the COMSECY, the SFP accident scenarios are low frequency, high consequence events. The fact that the calculated benefits (averted dose and property damage) for the high estimate cases exceed the estimated costs is not surprising given the bounding assumptions for loss of SFP integrity and subsequent zirconium fires effectively raise the calculated frequency of a major release by over one order of magnitude compared to the value used in Appendix D of the SFP study. The staff evaluated and provided its findings related to each step in the regulatory analysis process.

- The first test is whether additional studies were likely to support a finding that expediting the transfer of spent fuel or other actions related to the SFP would be a substantial safety improvement. The COMSECY and regulatory analysis document the staff's finding that additional studies would not support a substantial safety improvement finding. Approaches or presentation of information that would support an alternate view are mentioned but not adopted or recommended. The approach taken is consistent with Commission direction in its SRM related to SECY-12-0110.

- Notwithstanding a finding that expedited transfer of spent fuel or other actions discussed could provide a minor safety improvement but not a substantial safety improvement, the COMSECY and supporting analysis provide a cost/benefit assessment. As previously discussed, the base case assessments (recognizing the conservative and simplifying assumptions) do not show that the benefits of expediting spent fuel movement exceed the associated costs. The bounding assumptions used in the high estimate cases do lead to higher frequencies and consequences from releases and therefore the benefits appear to outweigh the costs. However, if directed to move on with Phase 2 of the Tier 3 plan, the staff would revisit assumptions for various parameters and simplifications (e.g., crediting mitigation for low density pools and not for high density pools) that inflated the benefits of expediting the transfer of spent fuel. In addition, Phase 2 assessments would include the additional risks introduced by additional cask loadings and the risks associated with dry cask storage. The staff and management continue to believe that additional studies of this issue are unlikely to justify requiring a backfit for existing nuclear power plants to expedite the transfer of spent fuel from storage pools to dry cask storage.

- Finally, the staff prepared and provides to the Commission a broader regulatory analysis that includes not only the health benefits associated with averted dose to the public and workers but also broader societal benefits associated with reducing damages to property. Similar to the discussion above for the backfit-related cost/benefit assessment, the broader regulatory analysis is provided in the enclosure to the COMSECY. The points mentioned above for the backfit assessment and mentioned in previous discussions of factors such as dollars per person-rem conversion factors and consequences beyond 50 miles are not repeated here. The staff and management continue to believe that additional studies of this issue are unlikely to support future regulatory actions.

Actions

There were no changes to the COMSECY and related regulatory analysis beyond the previously mentioned additions and revisions to address specific points.

Conclusion

The staff followed established processes and guidance and provided their findings to the Commission for consideration. In preparing the paper and recommendation, the staff followed the normal processes of routinely meeting with the Japan Lessons Learned Steering Committee and ensuring alignment between the Executive Director for Operations and management of the various NRC offices contributing to or affected by the assessment and recommendations. Alternatives and sensitivity studies are provided in the paper to support Commission deliberations on this issue. No additional changes to the paper or regulatory analysis appear to be warranted.