From:	NRRInternationalTravel Resource / OIP
To:	OIP Travel Distribution; Boger, Bruce; Deeds, Erin; Quinones, Lauren; Astwood, Heather
Subject:	FW: Pre-Trip Notification for CNRA SORRT Meeting Paris May 2-5, 2011
Date:	Tuesday, March 15, 2011 8:10:09 AM
Attachments:	043011 Paris Pre-Trip Notification .docx

REL

From: Adams, John NRR

Sent: Tuesday, March 15, 2011 8:09:35 AM

To: NRRInternationalTravel Resource

Subject: Pre-Trip Notification for CNRA SORRT Meeting Paris May 2-5, 2011

Auto forwarded by a Rule

Attached

001

International Travel Pre-Trip Notification

This is due 30 days before trip start date. Please complete the light shaded area.

TRAVEL INFORMATION

• **Traveler Name(s):** (include Office/Division)	John T. Adams	• **Phone #(s):**	301-415-1901
• **E-mail Address(es):**	john.adams@nrc.gov	• **Location(s):**	O12C01
• **Multiple Travelers:**	☒ **Less than 4** ☐ **4 or more** (see below)		
If 4+, Coordinating Office:	N/A		

If 4+, Office Director Approvals:	(1)	**CHOOSE ONE OF THE FOLLOWING**	Date:	
(Office Director approves travelers	(2)	**CHOOSE ONE OF THE FOLLOWING**	Date:	
from his/her office only)	(3)	**CHOOSE ONE OF THE FOLLOWING**	Date:	
	(4)	**CHOOSE ONE OF THE FOLLOWING**	Date:	

ADAMS Accession Number: [If 4+, Submit NRC Daily Note with ML# of pre-trip notification 30 days before trip start date]	N/A
• **Travel Dates** [mm/dd/yyyy]:	04/30/2011 - 05/05/2011
• **Destination(s)** [City, Country]:	Paris, France

• **Framework:**

☐ **Export and Import Licensing** ☒ **Multilateral Cooperation and Assistance**
☐ **Treaties, Agreements, and Conventions** ☐ **International Cooperative Research**
☐ **Bilateral Cooperation and Assistance** ☐ **Other (specify)**

• **International Organization:**

☐ **NEA/CSNI**	☒ **NEA/CNRA**	☐ **NEA/RWMC**	☐ **NEA/MDEP**
☐ **NEA/CRPPH**	☐ **IAEA/NS(TRANSSC)**	☐ **IAEA/NS(WASSC)**	
☐ **IAEA/(NUSSC)**	☐ **IAEA/NS(RASSC)**	☐ **IAEA/Safeguards**	
☐ **IAEA/NS**	☐ **IAEA/NE**	☐ **IAEA/Technical Cooperation**	
☐ **Other:** _____			

• **Purpose of Travel:**	Attend 2nd Meeting of the CNRA Safety of Research Reactor Task Group (SORRTG)
• **Desired Outcome:**	Review results of SORRTG Questionnaire and determine if additional work should be recommended to the CNRA. Include recommendations, if any, in a report for review and action by the CNRA at their June 2011 meeting.
• **Traveler Role(s):**	Participate in the review of completed questionnaires, develop conclusion based on questionnaire content, and contribute to recommendations for areas needing additional work, if justified, to be included in a report to the CNRA for the June 2011 meeting.

• **Is this an NRC Core or Non-Core Trip?** [*Core* means NRC-Funded. *Non-Core* is externally funded or travel to Canada]	☒ Core	☐ Non-Core
• **Is there a speech or presentation to be given?**	☐ Yes	☒ No

[If yes, send ADAMS ML# of presentations in an EDO One Week Look Ahead]	
• **Are policy issues or other items of Commission interest to be raised?**	☐ Yes ☒ No
If yes, how will the Commission be informed?	N/A

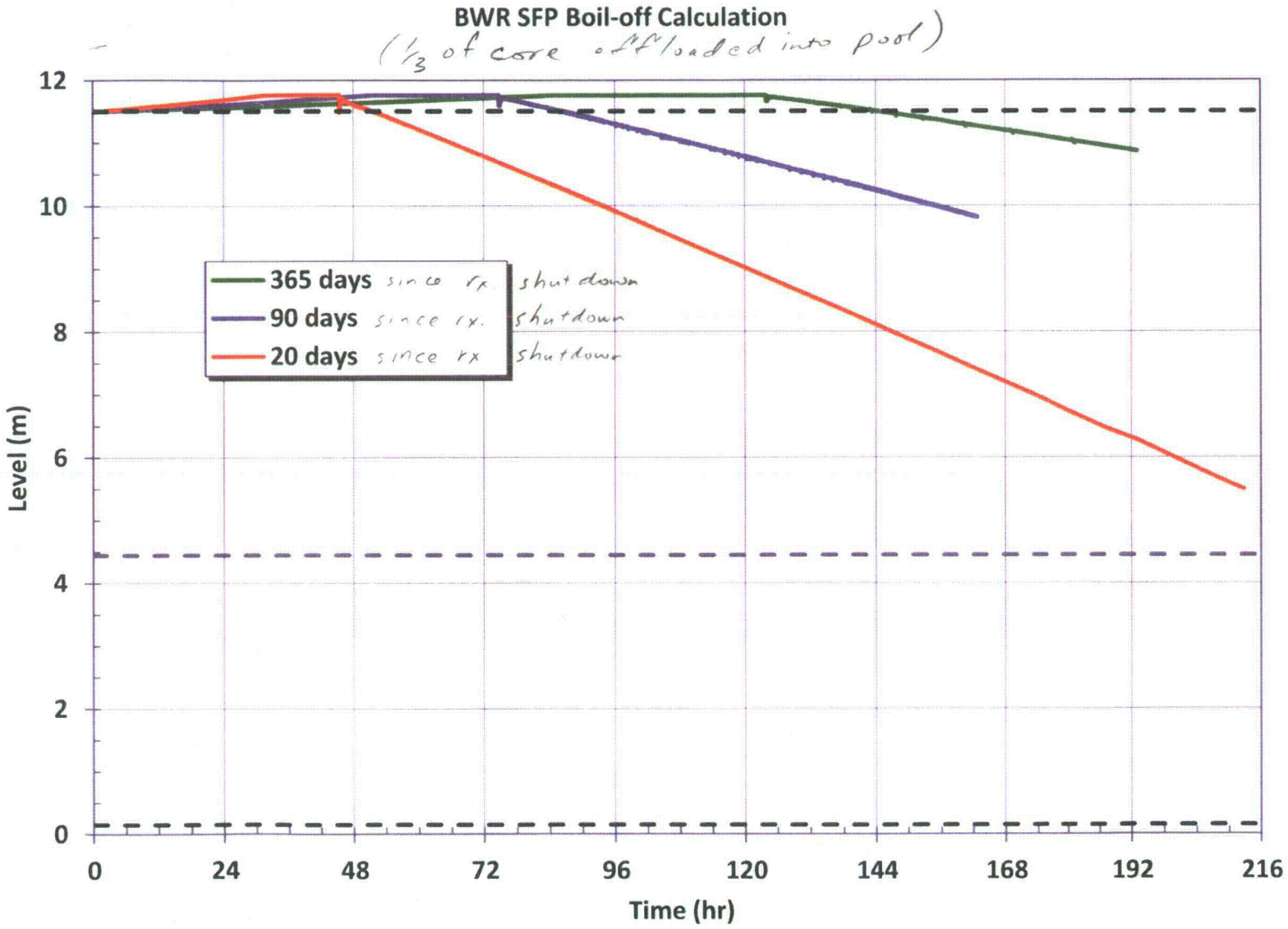
Schaperow, Jason

From: Schaperow, Jason
Sent: Tuesday, March 15, 2011 11:09 AM
To: 'kcw@dycoda.com'
Cc: Tinkler, Charles; Chang, Richard
Subject: telecon
Attachments: SS100_20110315_16131631.pd

Just to confirm our telecom that we had a few minutes ago...

Fukushima Daiichi unit 4 may be having a spent fuel pool meltdown.
Fukushima Daiichi unit 4 has a full core offload in its spent fuel pool.
We are planning to give to the NRC Chairman information on spent fuel pool boil-off times (and maybe spent fuel heatup times).

We need you to do boil-off calculations just like the attached, but for a full core offload into the pool. We need it for 3 decay times (20 days, 90 days, and 365 days). Also, if you could get it to run until the level reaches core mid-plane that would be even better.


BWR SFP Boil-off Calculation
(1/3 of core offloaded into pool)
365 days since rx shutdown
90 days since rx. shutdown
20 days since rx shutdown
Level (m)
Time (hr)
0
2
4
6
8
10
12
0
24
48
72
96
120
144
168
192
216

From: Graves, Herman
To: Case, Michael
Subject: RE: IRC Staffing
Date: Wednesday, March 16, 2011 11:52:36 AM

Mike,

A few SGESEB staff have expressed an interest in the IRC work:

Jose Pires, Structural & Seismic
Rasool Anooshehpoor, Seismologist
Thomas Herrity, Mechanical Engineer

<<Herman>>
<<301.251.7625>>
mail to: Herman.Graves@nrc.gov

From: Case, Michael
Sent: Wednesday, March 16, 2011 7:16 AM
To: Graves, Herman; Hogan, Rosemary; Csontos, Aladar; Koshy, Thomas; Lin, Bruce; Boyce, Tom (RES); Ali, Syed; Murphy, Andrew; Tregoning, Robert; Gavrilas, Mirela; Sydnor, Russell; Lorette, Phillip
Cc: Richards, Stuart
Subject: FW: IRC Staffing

Can you all start to think about this and let me know of any potential names by around noon?

From: Sheron, Brian
Sent: Tuesday, March 15, 2011 5:27 PM
To: Coyne, Kevin; Case, Michael; Coe, Doug; Correia, Richard; Gibson, Kathy; Lui, Christiana; Richards, Stuart; Sangimino, Donna-Marie; Scott, Michael; Uhle, Jennifer; Valentin, Andrea
Cc: Dion, Jeanne
Subject: IRC Staffing

I participated on a conference call with other ODs and led by Michele Evans, acting deputy OD in NSIR at 4 pm today.

The purpose of the conference call was to discuss staffing for the IRC for the near future. The IRC is currently staffed with members of the Reactor safety team, the Protective Measures team, Liaison Team, etc. There is also an ET member there. None of the teams are at their full compliment. What Michele is looking for is people that can staff the IRC and relieve the staff that are currently there. She said they are currently running 3 shifts (11pm-7am, 7am – 3pm, and 3pm to 11 pm). They would like to find staff that can work shifts for 4 days in a row (I think she wants 4 days on, 3 days off). She said the staff do not have to have had IRC training.

Several of us said we would certainly canvas our staff to see who was qualified to work in the IRC and could work there, but we needed to know what technical disciplines they were looking for. Michele did not have a list of needed disciplines, but said she would generate one and send it out. As of 5:15 pm I have not received a list yet.

00/3

However, I am assuming they will be looking for staff with expertise in such areas as systems analysis, severe accidents, radiological dose assessment, etc. In anticipation that these are the technical disciplines of interest, can you please start identifying your staff that you believe have some of the requisite skills needed for the IRC, and start asking if they would be available to work shifts in the IRC if asked to. HR said they would be eligible for normal overtime compensation.

Also, they will be looking for staff to go to Japan and relieve the technical staff that recently went there. There were 2 BWR experts that left over the weekend, and a team of 9 more (6 engineers and 3 OIP staff) left yesterday. The thinking is that the staff that recently went over would come back in 2 weeks, which is when they want to send a replacement team over there. So please check to see if you have any staff with the proper technical credentials, are reasonably good communicators, and would be willing to spend about 2 weeks in Japan as part of the team there.

I will forward the list of desired disciplines as soon as I receive them from Michele. Michele said she will be looking for the list of potential IRC replacements by COB tomorrow (3/16/11), thus, I will need your candidates by mid-afternoon.

For the team that will replace the one that was just sent to Japan, she said she would like us to update the list we previously sent by COB 3/17.

Killian, Lauren

From: Killian, Lauren
Sent: Wednesday, March 16, 2011 4:13 PM
To: Beasley, Benjamin
Cc: Kauffman, John
Subject: RE: As per MSNBC investigative reporter
Attachments: ML08185047719.docx

As requested, here is the part in the GI-199 Communication Plan that discusses the "aggregate" use of plant specific results in GI-199:
(FYI – the most recent GI-199 Communication Plan from ADAMS is also attached)

Q23. Will the NRC release the results of the Safety/Risk Assessment? If so, will plant-specific results be included?

A23. The Safety/Risk Assessment report will be made available on the public NRC Generic Issues Program (GIP) website (http://www.nrc.gov/about-nrc/regulatory/gen-issues.html), on the internal NRC GIP website (http://www.internal.nrc.gov/RES/projects/GIP/), and in the NRC Agencywide Document Access and Management System (ADAMS) under Accession No. ML100270582.

Regarding the plant-specific results, they are included in the Safety/Risk Assessment report (in appendix D), and have been used in the aggregate for the determination that further, plant-specific information and analysis is needed to investigate possible plant-specific improvements. (See the last section, "Safety/Risk Assessment Results - Plants in the GIP "Continue Region," of this Communication Plan.)

Note: Results of the Safety/Risk Assessment confirm that currently operating plants have adequate protection against seismic hazards; however, the results also indicated that GI-199 meets the NRC Generic Issues Program numerical risk threshold for an issue to continue to be evaluated for possible regulatory action (see answer A13). The Safety/Risk Assessment utilized simplifying methods and assumptions to produce plant-specific results to determine trends, not to finalize which plants will or will not be further analyzed.

004

Safety/Risk Assessment Results - Plants in the GIP "Continue Region"

Plant-specific results are included in the Safety/Risk Assessment report (in appendix D) and have been used in the aggregate to determine that further, plant-specific information and analysis is needed to investigate possible plant-specific improvements. Listed below are plants that are currently above the Generic Issues Program (GIP) numerical risk threshold for an issue to continue to be evaluated for possible regulatory action (see answers A13 and A23). (Note that the plants are listed in alphabetical order by NRC region.) During the analysis, this group of plants was referred to as the "plants in the *continue region*."

As more information becomes available and more detailed analysis is performed, this group of plants *will* change. As discussed in answer A4, generic communications on this issue will be addressed to all operating power plants in the United States. More detailed, plant-specific analysis of all plants will allow NRC staff to prioritize plants that may be considered for regulatory action. The need to continue evaluating GI-199 is based on the collective results, not the results for any particular plant.

Region I
Indian Point 2
Indian Point 3
Limerick 1
Limerick 2
Peach Bottom 2
Peach Bottom 3
Seabrook 1

Region II
Crystal River 3
Farley 1
Farley 2
North Anna 1
North Anna 2
Oconee 1
Oconee 2
Oconee 3
Saint Lucie 1
Saint Lucie 2
Sequoyah 1
Sequoyah 2
Summer
Watts Bar 1

From: Beasley, Benjamin
Sent: Wednesday, March 16, 2011 3:35 PM
To: Killian, Lauren
Cc: Kauffman, John
Subject: FW: As per MSNBC investigative reporter

Lauren,

Is there a Q&A in the Comm Plan that addresses Doug's question?

Ben

From: Tifft, Doug
Sent: Wednesday, March 16, 2011 3:24 PM
To: Manoly, Kamal; Scales, Kerby
Cc: Wilson, George; Munson, Clifford; Beasley, Benjamin; Chokshi, Nilesh; Nguyen, Quynh; McNamara, Nancy; Gray, Mel
Subject: RE: As per MSNBC investigative reporter

Thanks, that helps.

But is the reporters statement that our data shows that Indian Point Unit 3 is the plant in the US with the highest risk of suffering core damage from an earthquake an accurate conclusion from our data?

I seem to recall discussions where it was mentioned that the GI-199 data is only valid in aggregate for the nuclear industry, and is not enough data to draw any conclusion about any one reactor site. Is that true?

-Doug

From: Manoly, Kamal
Sent: Wednesday, March 16, 2011 3:19 PM
To: Scales, Kerby
Cc: Wilson, George; Munson, Clifford; Beasley, Benjamin; Chokshi, Nilesh; Nguyen, Quynh; Tifft, Doug; McNamara, Nancy
Subject: RE: As per MSNBC investigative reporter

Kerby,
The list in GI-199 Comm Plan (Q/A23) identifies the plants by name only that are in the continue category for GI-199 (Delta CDF) to request additional information via a generic communication. The GI-199 applies only to plants in Eastern and Central US. The list provided for the top 10 contains interpretation of **total** CDF based on information provided to the reporter concerning all plants in the US. That is why you see Diablo in the list of 10 even though Diablo is not a GI-199 plant.
Kamal

From: Scales, Kerby
Sent: Wednesday, March 16, 2011 1:47 PM
To: Manoly, Kamal
Cc: Wilson, George; Munson, Clifford; Beasley, Benjamin; Chokshi, Nilesh
Subject: FW: As per MSNBC investigative reporter

Kamal,
Can you review the list below and respond back to George and copy me?

Thanks,

From: Nguyen, Quynh
Sent: Wednesday, March 16, 2011 1:39 PM
To: Scales, Kerby
Cc: Tifft, Doug; Meighan, Sean
Subject: FW: As per MSNBC investigative reporter

Can you get somebody to check this quickly?

Thanks,
Q

From: Tifft, Doug
Sent: Wednesday, March 16, 2011 12:56 PM
To: OST05 Hoc; LIA04 Hoc; Nguyen, Quynh
Cc: McNamara, Nancy
Subject: FW: As per MSNBC investigative reporter

Please see below list. I think this is referring to GI 199, but the list of plants is completely different from the list in our GI 199 Comm Plan. Did the reporter mis-interpret the GI 199 documents? If so, where did they go wrong?

See Q23:
http://www.internal.nrc.gov/communications/plans/active/CommPlan_Generic_Issue199.pdf

Doug

From: Giarrusso, John (CDA) [mailto:John.Giarrusso@state.ma.us]
Sent: Wednesday, March 16, 2011 11:48 AM
To: McNamara, Nancy; Tifft, Doug
Subject: Fw: As per MSNBC investigative reporter

From: DeNicola, Fran (CDA) [mailto:Fran.Denicola@state.ma.us]
Sent: Wednesday, March 16, 2011 11:00 AM
To: john.giarrusso@state.ma.us <John.Giarrusso@state.ma.us>; Viveiros, John L. (CDA) <John.L.Viveiros@state.ma.us>; peter.judge@state.ma.us <Peter.Judge@state.ma.us>; linda.lecuyer@state.ma.us <Linda.Lecuyer@state.ma.us>
Subject: As per MSNBC investigative reporter

The top 10

Here are the 10 nuclear power sites with the highest risk of suffering core damage from an earthquake, showing their NRC risk estimates based on 2008 and 1989 geological data. (The full list of 104 reactors is below.)

1. Indian Point 3, Buchanan, N.Y.: 1 in 10,000 chance each year. Old estimate: 1 in 17,241. Increase in risk: 72 percent.

2. Pilgrim 1, Plymouth, Mass.: 1 in 14,493. Old estimate: 1 in 125,000. Increase in risk: 763 percent.

3. Limerick 1 and 2, Limerick, Pa.: 1 in 18,868. Old estimate: 1 in 45,455. Increase in risk: 141 percent.

4. Sequoyah 1 and 2, Soddy-Daisy, Tenn.: 1 in 19,608. Old estimate: 1 in 102,041. Increase in risk: 420 percent.

5. Beaver Valley 1, Shippingport, Pa.: 1 in 20,833. Old estimate: 1 in 76,923. Increase in risk: 269 percent.

6. Saint Lucie 1 and 2, Jensen Beach, Fla.: 1 in 21,739. Old estimate: N/A.

7. North Anna 1 and 2, Louisa, Va.: 1 in 22,727. Old estimate: 1 in 31,250. Increase in risk: 38 percent.

8. Oconee 1, 2 and 3, Seneca, S.C.: 1 in 23,256. Old estimate: 1 in 100,000. Increase in risk: 330 percent.

9. Diablo Canyon 1 and 2, Avila Beach, Calif.: 1 in 23,810. Old estimate: N/A.

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10. Three Mile Island, Middletown, Pa.: 1 in 25,000. Old estimate: 1 in 45,455. Increase in risk: 82 percent.

COMMUNICATION PLAN FOR GENERIC ISSUE 199

September 14, 2010
(ML081850477)

Goal

This plan will guide staff communications and activities with internal and external stakeholders of the United States Nuclear Regulatory Commission (NRC) as they relate to Generic Issue 199 (GI-199), "Implications of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern United States on Existing Plants."

Key Messages

The key messages to be communicated to stakeholders are as follows:

(1) **Operating nuclear power plants are safe:** Plants have adequate safety margin for seismic issues. The NRC's Safety/Risk Assessment confirms that overall *seismic risk* estimates remain small and that adequate protection is maintained.

(2) **Though still small, some seismic hazard estimates have increased:** Updates to seismic data and models indicate increased *seismic hazard* estimates for some operating nuclear power plant sites in the Central and Eastern United States.

(3) **Assessment of GI-199 will continue:** Plants are safe (see key message 1), but the NRC has separate criteria for evaluating whether plant improvements may be imposed. The NRC's Safety/Risk Assessment used readily available information and found that for about one-quarter of the currently operating plants, the estimated *core damage frequency* change is large enough to warrant further attention. Action may include obtaining additional, updated information and developing methods to determine if plant improvements to reduce seismic risk are warranted.

Background

This issue was proposed as a Generic Issue in May 2005 after NRC staff's review of updates to the seismic source and ground motion models provided by applicants in support of early site permits for new reactors. The updated seismic information included new Electric Power Research Institute (EPRI) models to estimate earthquake ground motion and updated models for earthquake sources in seismic regions such as eastern Tennessee, and around both Charleston, South Carolina and New Madrid, Missouri. The new data and models resulted in increased estimates of the seismic hazards for some plants in the Central and Eastern United States (CEUS). The staff evaluated this new information along with preliminary results from a 2004 U.S. Geological Survey (USGS) letter report regarding seismic hazard estimates. From this review the staff concluded that the likelihood of exceeding the seismic hazard values, used in plant design and in previous evaluations (such as the Individual Plant Examination of External Events (IPEEE) Program), may be higher than previously understood for some currently operating CEUS sites.

The staff compared the new seismic hazard data with the earlier evaluations conducted as a part of the IPEEE Program. From this comparison, the staff determined that the seismic designs of operating plants in the CEUS still provide adequate safety margins. At the same time, the staff also recognized that the new seismic data and models could reduce available safety margins due to increased estimates of the probability associated with seismic hazards at some of the currently operating sites in the CEUS.

The licensing basis for currently operating plants is based on deterministic analysis of design basis loads from the maximum earthquake level determined from historical data. The licensing basis does not include a probabilistic assessment of seismic hazards or probabilistic assessment of their potential impact on plant structures, systems, and components.

To maintain consistency with the performance-based approach for assessing seismic hazards for new reactors, the staff determined that the screening analysis should consider seismic hazard data and models besides those available from the USGS. This determination was based on the staff's ongoing interactions with stakeholders to develop a new performance-based approach for assessing seismic hazards for new reactors, as described in a memorandum to the Commission, "A Performance-Based Approach to Define the Safe Shutdown Earthquake Ground Motion," dated July 26, 2006 (ADAMS Accession No. ML052360044). The NRC staff held a public meeting, in February 2008, to engage external stakeholders. During the meeting, the representative from the Nuclear Energy Institute (NEI) expressed their willingness to support a collaborative approach to GI-199. This led to a Seismic Risk Memorandum of Understanding Addendum between EPRI and NRC.

The staff collected and analyzed seismic hazard information from the USGS and from other sources, and seismic risk information from IPEEE analyses. EPRI reported that they calculated mean seismic hazard results for all nuclear power plant sites in the CEUS and used these results to perform an independent evaluation of the implications of changes in seismic hazard estimates. The staff completed the review and analysis of seismic data in support of the Safety/Risk Assessment in June 2009.

Audience and Stakeholders

Internal

Internal stakeholders include the Commission, Office of the Executive Director for Operations (OEDO), Office of Nuclear Regulatory Research (RES), Office of Nuclear Reactor Regulation (NRR), Office of New Reactors (NRO), Office of Nuclear Material Safety and Safeguards (NMSS), Office of Federal and State Materials and Environmental Management Programs (FSME), Region I, Region II, Region III, Region IV, Office of Public Affairs (OPA), Advisory Committee on Reactor Safeguards (ACRS), Office of International Programs (OIP), Office of Congressional Affairs (OCA). (See the "Communications Team" section for a list of specific Communication Team members.)

External

External stakeholders include licensees, EPRI, Nuclear Energy Institute, Congressional members, public interest groups, media, and the public.

Communication Timeline

Detailed Activities to Support Release of the GI-199 Safety/Risk Assessment Report				
Stakeholder Group	**Specific Audience**	**Tool**	**Lead**	**Date**
Internal	Regional Offices	Brief	RES-Kauffman	May 12, 2010 (c)
	NRR Office Director	Brief	RES-Kauffman	May 12, 2010 (c)
	NRO Office Director	Brief	RES-Beasley	May 19, 2010 (c)
	Region I Management	Brief	RES-Kauffman	June 3, 2010 (c)
	EDO, Deputy EDOs	Brief	RES-Kauffman	June 22, 2010 (c)
	Commission offices	Technical Assistants Brief	RES-Kauffman	July 8, 2010 (c)
	NRC Chairman	Brief	RES-Kauffman	August 23, 2010
	Commission offices	EDO Daily Note (with link to documents)	RES-Killian	T* (September 1, 2010)
	EDO	Issue Safety/Risk Assessment Report (goes public after 5 working days)	RES-Sheron	T
External	General Public	Safety/Risk Assessment Report made public in ADAMS		T + 6 days (September 7, 2010)
	General Public	Press Release	OPA-Burnell	T + 6 days (September 7, 2010)
	Public and Licensees	Information Notice	NRR-Manoly	T + 6 days (September 7, 2010)
	Congressional Members/staff (as appropriate)	Phone Calls	OCA-Riley	T + 6 days (September 7, 2010)
	International contacts (as appropriate)	Phone Calls	OIP	T + 6 days (September 7, 2010)
	State/local governments (as appropriate)	Phone Calls	Regional State Liaison Officers Region I- McNamara/Tifft Region II – Trojanowski Region III – Barker Region IV - Maier	T + 6 days (September 7, 2010)

	USGS	Phone Call	OCA-Riley	T + 6 days (September 7, 2010)
	General Public	Public Meeting	RES-Beasley	T + [1 month]
	General Public	Seismic Fact Sheet Update	RES-Killian OPA-Burnell	August 26, 2010

* *"T"* refers to the time that the Director, RES endorses the Safety/Risk Assessment panel recommendation.

Communication Team

Name	Office	Telephone Number	Email ID
Contacts in RES Operating Experience and Generic Issues Branch			
Benjamin Beasley	RES	301-251-7676	Benjamin.Beasley@nrc.gov
John Kauffman		301-251-7465	John.Kauffman@nrc.gov
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Contacts in NRR Division of Engineering			
Patrick Hiland	NRR	301-415-3298	Patrick.Hiland@nrc.gov
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George Wilson		301-415-1711	George.Wilson@nrc.gov
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Contact in NMSS Division of Fuel Cycle Safety and Safeguards			
Marissa Bailey	NMSS	301-492-3264	Marissa.Bailey@nrc.gov
Contacts in NRC Regional Offices			
Wayne Schmidt	Region I	601-337-5315	Wayne.Schmidt@nrc.gov
Robert Carrion	Region II	404-562-4522	Robert.Carrion@nrc.gov
Vijay Meghani	Region III	630-829-9751	Vijay.Meghani@nrc.gov
Thomas Farnholtz	Region IV	817-860-8243	Thomas.Farnholtz@nrc.gov
Communications Assistant in OPA			
Scott Burnell	OPA	301-415-8204	Scott.Burnell@nrc.gov
Contact in OCA			
Tim Riley	OCA	301-415-8492	Tim.Riley@nrc.gov

Additional Communication Tools

The NRC has an internal Generic Issues Program (GIP) website (http://www.internal.nrc.gov/RES/GIP/index.html) and a public GIP website (http://www.nrc.gov/about-nrc/regulatory/gen-issues.html.). These websites include program information and documents, background and historical information, generic issue status information, and links to related programs.

The staff created a Seismic Issue Fact Sheet (http://www.nrc.gov/reading-rm/doc-collections/fact-sheets/fs-seismic-issues.html).

Questions and Answers

Background

Q1. What is the NRC Generic Issues Program?

A1. The Nuclear Regulatory Commission (NRC) Generic Issues Program (GIP) evaluates technical issues that apply to two or more facilities and that may not be covered by existing regulatory processes or criteria. Issues are evaluated for their effect on safety, security, and/or the environment. The GIP is a program by which these issues can be formally assessed to see if they can be dispositioned by existing regulatory processes or if not, to determine their safety and/or risk significance and how best to treat them. Information on the program is available on the public NRC GIP website (http://www.nrc.gov/about-nrc/regulatory/gen-issues.html); information is also available to NRC staff on the NRC internal GIP website (http://www.internal.nrc.gov/RES/projects/GIP/). Management Directive (MD) 6.4, "Generic Issues Program," contains GIP guidance (available at http://www.nrc.gov/about-nrc/regulatory/gen-issues/policy-procedures.html). MD 6.4 was updated in November 2009 to incorporate program changes described in SECY-07-0022 (available at http://www.nrc.gov/reading-rm/doc-collections/commission/secys/2007/).

Q2. What is Generic Issue 199 about?

A2. Generic Issue 199 investigates the safety and risk implications of updated earthquake-related data and models. These data and models suggest that the probability for earthquake ground shaking above the seismic design basis (see answers A8, A9, and A10) for some nuclear power plants in the Central and Eastern United States is still low, but larger than previous estimates (see answer A12).

Q3. Where can I get current information about Generic Issue 199?

A3. The public NRC Generic Issues Program (GIP) website (http://www.nrc.gov/about-nrc/regulatory/gen-issues.html) contains program information and documents, background and historical information, generic issue status information, and links to related programs. The GI-199 section of the NRC internal GIP website (http://www.internal.nrc.gov/RES/projects/GIP/Individual%20GIs/GI-0199.html) contains additional information about Generic Issue 199 (GI-199) and is available to NRC staff. The latest Generic Issue Management Control System quarterly report, which has regularly updated GI-199 information, is publicly available at http://www.nrc.gov/reading-rm/doc-collections/generic-issues/quarterly/index.html. Additionally, the U.S. Geological Survey data is publicly available at http://earthquake.usgs.gov/hazards/products/conterminous/2008/.

Q4. Are all U.S. plants being evaluated as a part of Generic Issue 199?

A4. The scope of the Generic Issue 199 (GI-199) Safety/Risk Assessment is limited to all plants in the Central and Eastern United States. Although plants at the Columbia, Diablo Canyon, Palo Verde, and San Onofre sites are not included in the GI-199 Safety/Risk Assessment, the Information Notice on GI-199 is addressed to all operating power plants in the U.S. (as well as all independent spent fuel storage installation licensees). The staff will also consider inclusion

of operating reactors in the Western U.S. in its future generic communication information requests.

Q5. Does GI-199 affect license renewal?

A5. No. The NRC's regulations for license renewal (10 CFR Part 54) require licensees to manage age-related degradation to ensure that systems, structures, and components (SSCs) will fulfill their safety-related functions, as specified in the current licensing basis, for the period of extended operation. The aging management review conducted by license renewal applicants specifically addresses the impact of age-related degradation on SSC seismic capacity. It should be noted that a plant's licensing basis, including its seismic design basis, is established outside of the license renewal process during initial plant licensing and subsequent license amendments. In addition, the NRC has processes to evaluate the adequacy of plant licensing bases (e.g., the Generic Issues Program) based on new information or operating experience and, if necessary, improve safety (e.g., require plant improvements through the backfit process).

Note: Related to license renewal, the County Executive of Westchester County (New York) and groups from New Jersey submitted a petition for rulemaking on license renewal, including a seismic-related aspect. NRC denied this petition. The petitioners then filed suit in the U.S. Court of Appeals Second Circuit and the court upheld the NRC's position. Details are available on the internal webpage of the Office of General Counsel (under Law Library, Summary of AEC-NRC Litigation, "Spano v. NRC" (2d Cir. 2009): http://www.internal.nrc.gov/ogc/internal/AEC-NRC_Cases.pdf.)

Q6. Are the implications of new seismic hazard estimates being considered for the storage of spent fuel?

A6. Yes, while the GI-199 Safety/Risk Assessment focused solely on operating power reactors in the Central and Eastern U.S., spent fuel storage has been considered by NRC.

The NRC Office of Nuclear Materials Safety and Safeguards (NMSS) was informed of GI-199 and a preliminary screening review was performed in November, 2008 by the NMSS Division of Spent Fuel Storage and Transportation. There is a total of 40 operating independent spent fuel storage installations (ISFSIs) in the Central and Eastern U.S. (CEUS). Except for a wet storage facility at G. E. Morris located in Illinois, the ISFSIs are co-located at the operating and permanently shutdown reactor sites. A review of design earthquakes (DE) used at the existing ISFSI locations in CEUS indicated that the safety margin (defined for ISFSIs as the ratio of DE/SSE, where SSE is the *safe shutdown earthquake* discussed in answer A8) for the cask designs were in the range of 1.20 ~ 3.90. Therefore, NMSS considers that there is significant margin built into the existing designs and has confidence that the ISFSIs can continue to operate safely while the licensees' investigate this issue using their site specific information. Even so, holders of operating license for ISFSIs are included among addressees in the Information Notice on GI-199.

Spent fuel pools (SFPs) were not specifically evaluated as part of GI-199. However, based on their design attributes (as follows), SFPs remain safe. SFPs are constructed of reinforced concrete, several feet thick, with a stainless steel liner to prevent leakage and maintain water quality. Due to their configuration, SFPs are inherently structurally-rugged and are designed to the same seismic requirements as the nuclear plant.

Note: Typically, SFPs are about 40 feet deep and vary in width and length. The fuel is stored in stainless steel racks and submerged with approximately 23 feet of water above the top of the stored fuel. Each plant has a preferred SFP make-up water source (the refueling water storage tank for pressurized water reactors and the condensate storage tank for boiling water reactors). SFPs have alternate means of make-up such as service water systems and the fire water system. SFPs are also typically designed (e.g. with anti-siphon check valves) and instrumented such that leakage is minimized and promptly detected.

Q7. Are the implications of new seismic hazard estimates being considered for fuel cycle facilities?

A7. Yes, while the GI-199 Safety/Risk Assessment focused solely on operating power reactors in the Central and Eastern U.S., fuel cycle facilities have been considered by NRC. Based on preliminary reviews of the updated seismic hazard estimates, NRC staff in the Office of Nuclear Material Safety and Safeguards concluded that, for the fuel cycle facilities within the CEUS, there is no immediate safety concern.

Existing facilities (uranium enrichment, fuel fabrication [high and low enriched]) were mostly built to local building codes. These facilities demonstrate compliance with the performance requirements in 10 CFR 70.61 through their Integrated Safety Analyses (ISAs). 10 CFR Part 70 licensees are required to perform an ISA in which seismic events are addressed (through a combination of design and preventive/mitigative actions). To demonstrate compliance with Part 70, licensees must limit the risk of high and intermediate consequence events, by limiting the likelihood or consequence. It is expected that, in view of this new data, existing facilities will consider the updated information as it relates to the performance requirements and see if additional safety controls are necessary.

In addition to the ISA requirements, new facilities have to meet the even higher baseline design criteria (BDC), which requires the design to provide adequate protection against natural phenomena with consideration of the most severe documented historical events for the site. Three new facilities (LES, USEC ACP, and MOX) are undergoing construction. Conservatism was built into the design of these facilities (i.e., design code factors of safety, elasticity in the structures, and conservatism in the design evaluation) resulting in additional safety margin. All new facilities and new processes at existing facilities are required to meet 10 CFR 70.64(a)(2), which requires adequate protection against natural phenomena.

Note: Regarding some particular facilities, the Paducah Gaseous Diffusion Plant (a 10 CFR Part 76 facility) was designed to meet local building codes at the time of its construction in the early 1950s. Later in the late 1990s, as part of the Certification process, the Paducah plant was evaluated and reinforced to meet a 250 year return earthquake. Honeywell's construction was also consistent with the local building codes when it was built 50 years ago. Later during the 1990s, structural modifications were performed at Honeywell to upgrade the plant so it could withstand a 475-yr recurrence site-specific earthquake.

Q8. How can I learn more about earthquakes?

A8. A fact sheet on seismic issues for existing nuclear power plants is available on the NRC public website at http://www.nrc.gov/reading-rm/doc-collections/fact-sheets/fs-seismic-issues.html. Background information on earthquakes can also be obtained at the U.S. Geological Survey website at http://earthquake.usgs.gov/.

Q9. What do the following terms mean?

- ***Annual exceedance frequency***
- ***Core damage frequency***
- ***Design basis earthquake* or *safe shutdown earthquake***
- ***Ground acceleration***
- ***High confidence of low probability of failure capacity***
- ***Large early release frequency***
- ***Seismic hazard***
- ***Seismic margin***
- ***Seismic risk***

A9. The terms are defined as follows:

- ***Annual exceedance frequency (AEF)*** – Number of times per year that a site's ground motion is expected to exceed a specified acceleration.

- ***Core damage frequency (CDF)*** – Expected number of core damage events per unit of time. *Core damage* refers to the uncovery and heat-up of the reactor core, to the point that prolonged oxidation and severe fuel damage are not only anticipated but also involve enough of the core to result in off-site public health effects if released. *Seismic core damage frequency* refers to the component of total CDF that is due to seismic events.

- ***Design basis earthquake* or *safe shutdown earthquake (SSE)*** – A *design basis earthquake* is a commonly employed term for the *safe shutdown earthquake* (*SSE*); the SSE is the earthquake ground shaking for which certain structures, systems, and components are designed to remain functional. In the past, the SSE has been commonly characterized by a standardized spectral shape associated with a peak *ground acceleration* value.

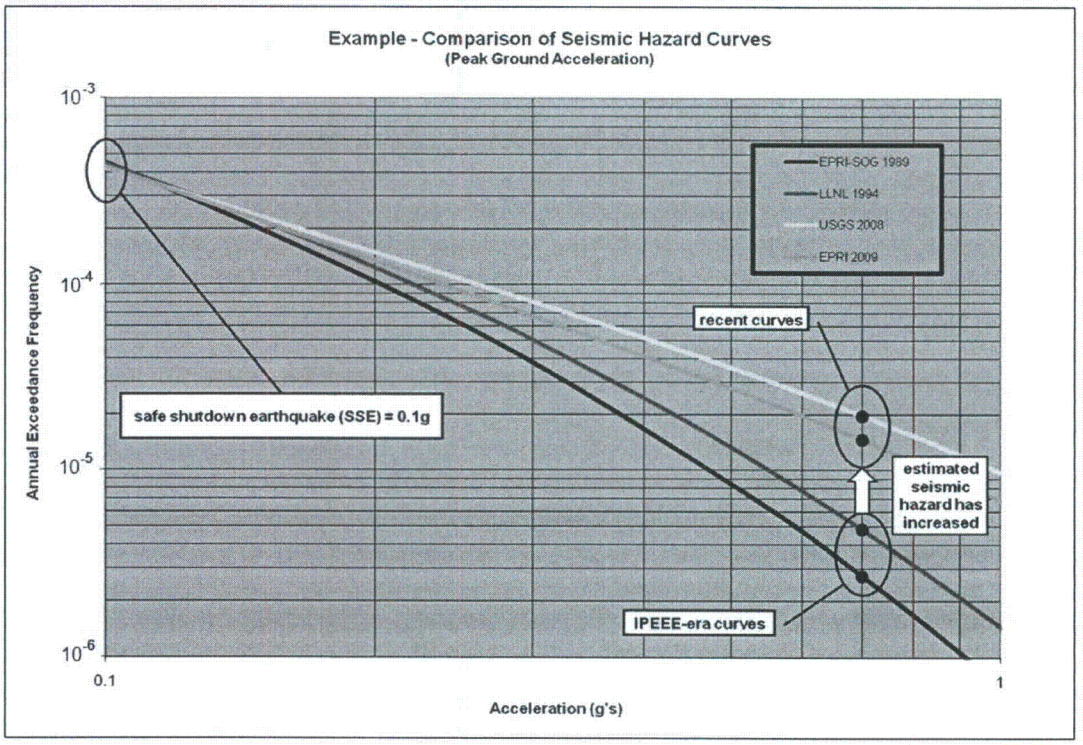
- ***Ground acceleration*** – Acceleration produced at the ground surface by seismic waves, typically expressed in units of *g*, the acceleration of gravity at the earth's surface.

- ***High confidence of low probability of failure (HCLPF) capacity*** – A measure of *seismic margin*. In *seismic risk* assessment, *HCLPF capacity* is defined as the earthquake motion level, at which there is high confidence (95%) of a low probability (at most 5%) of failure of a structure, system, or component.

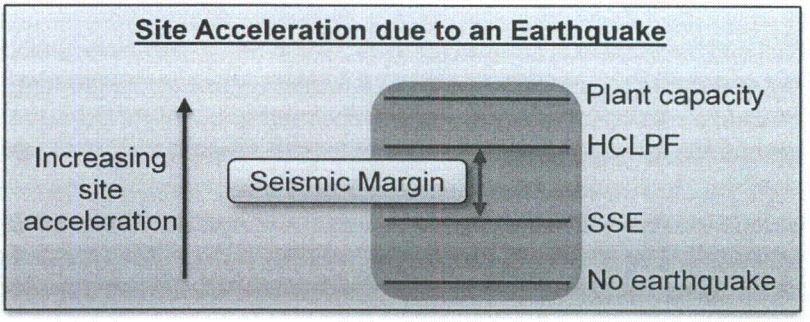
- ***Large early release frequency (LERF)*** – The expected number of large early releases per unit of time. A *large early release* is the rapid, unmitigated release of airborne fission products from the containment building to the environment, occurring before the effective implementation of off-site emergency response and protective actions, such that there is a potential for early health effects. *Seismic large early release frequency* refers to the component of total LERF that is due to seismic events.

- ***Seismic hazard*** – Any physical phenomenon, such as ground motion or ground failure, that is associated with an earthquake and may produce adverse effects on human activities (such as posing a risk to a nuclear facility).

For the representative plant in the chart below, the *annual exceedance frequency* for a 0.7*g* acceleration (e.g., for a large, but highly improbable earthquake) has increased from approximately one in 250,000 years (for IPEEE-era curves) to approximately one in 60,000 years (for recent *seismic hazard* curves). (In other words, the annual exceedance frequency for a 0.7*g* acceleration has increased from about 4 x 10^{-6} (0.000004) per year for IPEEE-era curves to about 1.8 x 10^{-5} (0.000018) per year for recent seismic hazard curves.) Note that the curves in this example are virtually indistinguishable at the SSE (design basis) level, but this is not always the case. Ultimately, GI-199 is about understanding the impact of these seismic hazard changes on reactor risk.



- ***Seismic margin*** – The difference between a plant's *HCLPF* capacity and its seismic design basis (*safe shutdown earthquake, SSE*), as shown in the figure below. (Note that the "plant capacity" label in this figure is the acceleration expected to result in core damage half of the time.) (Also see answer A11.)



- ***Seismic risk*** – The risk (frequency of occurrence multiplied by its consequence) of severe earthquake-initiated accidents at a nuclear power plant. A severe accident is an accident that causes core damage, and, possibly, a subsequent release of radioactive materials into the environment. Several risk metrics may be used to express *seismic risk*, such as seismic *core damage frequency* and seismic *large early release frequency.*

Safety

Q10. How was the seismic design basis for an existing nuclear power plant established?

A10. The seismic ground motion used for the design basis was determined from the evaluation of the maximum historic earthquake within 200 miles of the site, without explicitly considering the time spans between such earthquakes; safety margin was then added beyond this maximum historic earthquake to form a hypothetical *design basis earthquake* (see answer A9). The relevant regulation for currently operating plants is 10 CFR Part 100, Appendix A, "Seismic and Geologic Siting Criteria for Nuclear Power Plants" (http://www.nrc.gov/reading-rm/doc-collections/cfr/part100/part100-appa.html).

Q11. Is there margin above the design basis?

A11. Yes, there is margin beyond the design basis (see answer A9). In the mid to late 1990s, NRC staff reviewed the plants' assessments of potential consequences of severe earthquakes (earthquakes well beyond the safety margin included in each plant's design basis), which licensees performed as part of the Individual Plant Examination of External Events program. From this review, the staff determined that seismic designs of operating plants in the Central and Eastern United States have considerable safety margins, for withstanding earthquakes, built into the designs.

Q12. What do you mean by "increased estimates of seismic hazards" at nuclear power plant sites?

A12. *Seismic hazard* (earthquake hazard) represents the chance (or probability) that a specific level of ground shaking could be observed or exceeded at a given location. Our estimates of seismic hazard at some Central and Eastern United States locations have changed based on results from recent research, indicating that earthquakes occurred more often in some locations than previously estimated. Our estimates of seismic hazard have also changed because the models used to predict the level of ground shaking, as caused by a specific magnitude earthquake at a certain distance from a site, changed. The increased estimates of seismic hazard at some locations in the Central and Eastern United States were discussed in a memorandum to the Commission, dated July 26, 2006. (The memorandum is available in the NRC Agencywide Documents Access and Management System [ADAMS] under Accession No. ML052360044).

Q13. What has the Safety/Risk Assessment found and what does it mean for Generic Issue 199?

A13. Results of the Safety/Risk Assessment confirm that currently operating plants have adequate protection against *seismic hazards* (see Safety/Risk Assessment report transmittal

memorandum). However, based on a separate criterion in the Generic Issues Program, the estimated *core damage frequency* change is still large enough to warrant further attention regarding the possible imposition of plant improvements. Action could include obtaining information and developing methods to complete plant-specific value-impact analyses.

RES staff developed a methodology and implemented it to assess the risk associated with this issue. Overall *seismic risk* estimates remain small in an absolute sense. All operating plants in the Central and Eastern United States have seismic core damage frequency (SCDF) less than or equal to 10^{-4} (0.0001) per year, which is considered safe (see answer to A15). The SCDF changes (the difference in SCDFs calculated using the old and new seismic hazard information) for a number of plants lie in the range of 10^{-4} to 10^{-5} (0.0001 to 0.00001) per year, which meets the NRC Generic Issues Program numerical risk threshold for an issue to continue to be evaluated for possible regulatory action.

Q14. Are the plants safe? If you are not sure they are safe, why are they not being shut down? If you are sure they are safe, why are you continuing evaluations related to this generic issue?

A14. Yes, currently operating nuclear plants in the Central and Eastern United States remain safe, with no need for immediate action. This determination is based on NRC staff reviews associated with Early Site Permits, the conclusions of the Generic Issue 199 Screening Panel (comprised of technical experts), and the conclusions of the Safety/Risk Assessment Panel (also comprised of technical experts).

No immediate action is needed because: (1) existing plants were designed to withstand anticipated earthquakes with substantial design margins, as confirmed by the results of the Individual Plant Examination of External Events; (2) the probability of exceeding the *safe shutdown earthquake* ground motion (see answer A9) may have increased at some sites, but only by a relatively small amount; (3) the increased probability is primarily in the high structural response frequencies, so buildings and equipment should not be affected (seismic amplitudes at lower frequencies are the primary contributors to building and equipment damage); and (4) the Safety/Risk Assessment Stage results indicate that the probabilities of seismic core damage are lower than the guidelines for taking immediate action.

Even though the staff has determined that existing plants remain safe, the Generic Issues Program criteria direct staff to continue their analysis to determine whether any cost-justified plant improvements can be identified to make plants even safer.

Q15. How do you know the plants are safe?

A15. The Safety/Risk Assessment results confirm that plants are safe. The relevant risk criterion for GI-199 is total *core damage frequency* (CDF). The threshold for taking immediate regulatory action (found in NRR Office Instruction LIC-504, see below) is a total CDF greater than or on the order of 10^{-3} (0.001) per year. For GI-199, the staff calculated seismic CDFs of 10^{-4} (0.0001) per year and below for nuclear power plants operating in the Central and Eastern U.S. (CEUS) (based on the new U.S. Geological Survey seismic hazard curves). The CDF from internal events (estimated using the staff-developed Standardized Plant Analysis of Risk models) and fires (as reported by licensees during the IPEEE process and documented in NUREG-1742), when added to the seismic CDF estimates results in the total risk for each plant to be, at most, 4×10^{-4} (0.0004) per year or below. This is well below the threshold (a CDF of

10^{-3} [0.001] per year) for taking immediate action. Based on the determination that there is no need for immediate action, and that this issue has not changed the licensing basis for any operating plant, the CEUS operating nuclear power plants are considered safe. In addition, as detailed in the GI-199 Safety/Risk Assessment and answers A13 and A14 above, there are additional, qualitative considerations that provide further support to the conclusion that plants are safe.

Note: The NRC has an integrated, risk-informed decision-making process for emergent reactor issues (NRR Office Instruction LIC-504, ADAMS Accession No. ML100541776 [not publically available]). In addition to deterministic criteria, LIC-504 contains risk criteria for determining when an emergent issue requires regulatory action to place or maintain a plant in a safe condition.

Despite NRC's determination that plants are safe to operate, MD 6.4, "Generic Issues Program," contains quantitative risk guidelines that place GI-199 into the category of continued evaluation to determine if cost-beneficial backfits can be justified at any plants.

Note: Also, New U.S. Geological Survey seismic hazard information provides ground acceleration likelihoods at each power plant site for both design basis and beyond design basis earthquakes. This seismic hazard information was combined with an estimate of each plant's resistance to earthquakes (seismic fragility) to produce an estimate of the frequency of damage to the reactor core due to earthquakes. This seismic core damage frequency (SCDF) was combined with estimates of the core-damage frequency (CDF) for internal events and fires, and the total CDF was then compared to risk thresholds used by the NRC to assess and assure that nuclear power plants are operated safely. The frequency calculated for all operating nuclear power plants in the CEUS is in the range considered safe.

Q16. Why are new nuclear plants being built to different seismic design requirements than existing nearby plants? Why are the currently operating plants not required to meet the new standards?

A16. Currently operating plants have been determined to adequately protect the public; new plants are designed to different requirements in order to meet the Nuclear Regulatory Commission's expectation that the new plants will provide enhanced margins of safety (see "Regulation of Advanced Nuclear Power Plants; Statement of Policy" 59 FR 35461 at http://www.nrc.gov/reading-rm/doc-collections/commission/policy/#power). There are two primary ways of determining safety: deterministic assessments (based on past events and engineering judgment) and probabilistic assessments. New plants employ probabilistic methods. Existing plants were built to older standards, based on deterministic assessments. Those standards have been monitored, and were found to be sufficient and appropriate. In order to impose new requirements on existing plants, the NRC must be able to justify the new requirements in accordance with the "Backfit Rule" (10 CFR 50.109, available at http://www.nrc.gov/reading-rm/doc-collections/cfr/part050/part050-0109.html). The NRC needs additional information to justify any new requirements, and the Safety/Risk Assessment Panel recommended taking action to acquire the information.

Q17. How does the occurrence of a new earthquake in the Central or Eastern United States affect Generic Issue 199?

A17. The effect of a single earthquake is small on the estimated *seismic hazard* (defined in answer A9) and hence on Generic Issue 199, unless it occurs in an area not previously recognized as being capable of producing earthquakes, or is larger than previously believed possible in a region. In a seismic hazard study, the seismic source zones are specifically delineated to include a sufficient number of earthquakes to provide a stable estimate of the seismicity rate and are thus relatively insensitive to the addition of a single earthquake. If an earthquake does occur in an area not previously recognized as being capable of producing earthquakes or if an earthquake occurs that is larger than previously believed possible in a region, changes to the seismic hazard model used to develop seismic hazard estimates would be required.

Note: The magnitude 5.2 earthquake that occurred on April 18, 2008 in southeastern Illinois provides a good example of the potential impact of a single earthquake. This earthquake occurred in an area recognized as being capable of producing significant earthquakes (the Wabash Valley seismic source zone) and was smaller than the maximum magnitude event defined for the zone based on geologic investigations (maximum magnitude of 7-7.5). The addition of a single event of this magnitude to the earthquake database for this area would likely change the activity rate by less than a few percent and thus have a very small impact on the estimated seismic hazard at any of the nuclear facilities in the area.

Schedule

Q18. What has been done about this issue since it was identified as a generic issue in the Generic Issues Program?

A18. The following summarizes what has been done on Generic Issue 199 (GI-199):

Prioritization and Screening

- *June 2005:* The issue was logged into the Generic Issues Program (GIP) and, based on the NRC determination that the seismic design of plants in the Central and Eastern United States still provided an adequate level of protection, the Agency decided that this issue was a relatively low priority.
- *November 2005 – February 2007:* The Agency awarded a contract to screen this issue and determine whether it should continue to be evaluated under the GIP. In 2006, the contractor notified RES of problems obtaining information that the contractor wanted to perform its task.
- *April 2007:* The NRC decided to use Agency staff to complete the screening analysis using guidance provided in Management Directive (MD) 6.4 and SECY-07-0022, "Status Report on Proposed Improvements to the Generic Issues Program," dated January 30, 2007. MD 6.4 outlines the seven GIP criteria for use in determining whether proposed generic issues should be designated generic issues (the screening process) and proceed to the Safety/Risk Assessment Stage of the GIP.
- *September 2007:* An initial screening analysis was completed.
- October 2007: For consistency with the performance-based approach for assessing *seismic hazards* for new reactors, the staff determined that the screening analysis should consider seismic hazard data and models besides those available from the U.S.

Geological Survey.

- *February 2008:* The NRC completed the GIP screening with the GI-199 Screening Panel concluding that the issue should proceed to the Safety/Risk Assessment Stage under the GIP. The NRC staff held a public meeting to engage external stakeholders. During the meeting, the representative from NEI expressed their willingness to support a collaborative approach to GI-199. (This led to a Seismic Risk Memorandum of Understanding Addendum between the Electric Power Research Institute and the NRC Office of Nuclear Regulatory Research (RES).)

Safety/Risk Assessment Stage

- GI-199 then entered the Safety/Risk Assessment Stage of the GIP. RES staff collected and analyzed seismic hazard information from the U.S. Geological Survey and other sources, and *seismic risk* information from Individual Plant Examination of External Events analyses.
- *November 2008:* The NRC Office of Nuclear Material Safety and Safeguards (NMSS) performed a preliminary review related to independent spent fuel storage installations (ISFSIs). A review of design earthquakes (DE) used at the existing ISFSI locations in Central and Eastern U.S., indicated that there is significant margin built into the existing designs and NMSS determined that they have confidence that the ISFSIs can continue to operate safely while GI-199 is processed.
- *June 2009:* In support of the Safety/Risk Assessment, the staff completed the review and detailed analysis of seismic data for 96 plants.
- *July 2009 – March 2010:* Several Safety/Risk Assessment Panel meetings were held to determine recommendations in light of stakeholder input that was received.
- *April 2010 – August 2010:* The Safety/Risk Assessment report is finalized. Internal briefings and communications are carried out (to build NRC consensus and to prepare for the release of the Safety/Risk Assessment report and associated public meeting).
- During the process of resolving GI-199, staff responded to Freedom of Information Act requests and held numerous meetings with internal and external stakeholders.

Q19. Why is it taking the NRC so long to process Generic Issue 199?

A19. This is a complicated issue involving the intersection of the probabilistic risk analysis and seismic disciplines. Obtaining data, developing methods, and performing analyses are all required to address the issue. Analyzing a few representative plants for this issue (as is normally done in the Generic Issues Program) is inappropriate because the *seismic hazard* and associated impact to the power plant are very site-specific; so analysis for 96 separate plants is required. (Refer to A14 for a summary of what has been done on GI-199 since it was first identified.) GI-199 has also been a communication-intensive generic issue because it affects many parts of the NRC and industry, and because it is important to NRC and all stakeholders that the Safety/Risk Assessment results are properly conveyed.

Q20. What will happen next regarding Generic Issue 199?

A20. The next step is for the staff to complete the Safety/Risk Assessment Stage of the Generic Issues Program (GIP). The Safety/Risk Assessment report will soon be published, followed by an information notice being sent to all licensees of nuclear power reactors and independent spent fuel storage installations. A public meeting will be held to discuss the results of the Safety/Risk Assessment and the next steps for GI-199. After the Safety/Risk Assessment

Stage, further action regarding GI-199 will be pursued (such as obtaining more detailed, plant-specific information and performing analysis to determine whether plant-specific improvements are warranted). NRC staff will also make presentations to the Advisory Committee on Reactor Safeguards.

Q21. Aside from evaluations for GI-199, what is the NRC's expectation regarding the use of updated probabilistic seismic hazard information in regulatory applications?

A21. It is expected that all NRC licensees that are required to analyze risks and hazards impacting their operations will use the most current seismic hazard information.

Regarding currently operating nuclear power plants, there is no requirement that the plants re-evaluate their seismic design basis (10 CFR 100, Appendix A, "Seismic and Geologic Siting Criteria for Nuclear Power Plants"), but plants do need to use the most updated information available in the case of risk-informed licensing amendments.

Note: The NRC guidance for using probabilistic risk assessment (PRA) in risk-informed decisions on plant-specific changes to the licensing basis is provided in RG 1.174. The scope, level of detail, and technical acceptability of the PRA are to be commensurate with the application for which it is intended and the role that the PRA results play in the integrated decision process. One over-riding requirement is that the PRA should realistically reflect the actual design, construction, operational practices, and operational experience of the plant and its owner. RG 1.200 provides further guidance concerning the technical adequacy of PRAs and states that seismic hazard analysis should include current information. Consistent with this guidance, the staff expects that licensees will use the most recent seismic hazard information available for risk-informed regulatory applications.

Regarding seismic requirements for dry cask storage systems and independent spent fuel storage installations (ISFSIs), the staff also expects that licensees will use the most recent seismic hazard information available for risk-informed regulatory applications.

Note: NRC regulations (in 10 CFR Part 72) require licensees to perform written evaluations to establish that, for their site-specific conditions, the conditions set forth in the Certificate of Compliance (CoC) have been met. They must also perform evaluations showing that cask storage pads and areas have been designed to adequately support the static and dynamic loads of the stored casks, considering potential amplification of earthquakes through soil-structure interaction as well as soil liquefaction potential or other soil instability due to vibratory ground motion.

Stakeholder Interest

Q22. Has the NRC received any requests from government officials regarding seismic issues?

A22. Yes. On November 15, 2007, the NRC received a letter (available in the NRC Agencywide Documents Access and Management System, ADAMS, under Accession No. ML0732500954) from the Attorneys General of six states (Connecticut, Delaware, Illinois, Kentucky, New York, and Vermont). The letter encouraged the NRC to consider siting and safety requirements, including geographic and seismic issues, in the regulatory process for license renewal. The NRC reviewed this letter and responded that the items of concern are

addressed in "ongoing regulation [that]... occurs throughout the life of the license... [and that] expand[ing] the scope of license renewal to cover...[the] issues raised in [the] letter...[would be] duplicating the Commission's responsibilities..." (ADAMS Accession No. ML073400603). Additionally, several Freedom of Information Act requests were received, and NRC staff responded to the requests; the U.S. Geological Survey data related to these requests is publicly available under ADAMS Accession No. ML072880133.

Also, the County Executive of Westchester County (New York) and groups from New Jersey submitted a petition for rulemaking on license renewal, including a seismic-related aspect. The NRC denied this petition. The petitioners then filed suit in the U.S. Court of Appeals Second Circuit and the court upheld the NRC position. Details are available on the internal webpage of the NRC Office of General Counsel (under Law Library, Summary of AEC-NRC Litigation, "Spano v. NRC" (2d Cir. 2009): http://www.internal.nrc.gov/ogc/internal/AEC-NRC_Cases.pdf.)

Q23. Will the NRC release the results of the Safety/Risk Assessment? If so, will plant-specific results be included?

A23. The Safety/Risk Assessment report will be made available on the public NRC Generic Issues Program (GIP) website (http://www.nrc.gov/about-nrc/regulatory/gen-issues.html), on the internal NRC GIP website (http://www.internal.nrc.gov/RES/projects/GIP/), and in the NRC Agencywide Document Access and Management System (ADAMS) under Accession No. ML100270582.

Regarding the plant-specific results, they are included in the Safety/Risk Assessment report (in appendix D), and have been used in the aggregate for the determination that further, plant-specific information and analysis is needed to investigate possible plant-specific improvements. (See the last section, "Safety/Risk Assessment Results - Plants in the GIP "Continue Region," of this Communication Plan.)

Note: Results of the Safety/Risk Assessment confirm that currently operating plants have adequate protection against seismic hazards; however, the results also indicated that GI-199 meets the NRC Generic Issues Program numerical risk threshold for an issue to continue to be evaluated for possible regulatory action (see answer A13). The Safety/Risk Assessment utilized simplifying methods and assumptions to produce plant-specific results to determine trends, not to finalize which plants will or will not be further analyzed.

Safety/Risk Assessment Results - Plants in the GIP "Continue Region"

Plant-specific results are included in the Safety/Risk Assessment report (in appendix D) and have been used in the aggregate to determine that further, plant-specific information and analysis is needed to investigate possible plant-specific improvements. Listed below are plants that are currently above the Generic Issues Program (GIP) numerical risk threshold for an issue to continue to be evaluated for possible regulatory action (see answers A13 and A23). (Note that the plants are listed in alphabetical order by NRC region.) During the analysis, this group of plants was referred to as the "plants in the *continue region.*"

As more information becomes available and more detailed analysis is performed, this group of plants *will* change. As discussed in answer A4, generic communications on this issue will be addressed to all operating power plants in the United States. More detailed, plant-specific analysis of all plants will allow NRC staff to prioritize plants that may be considered for regulatory action. The need to continue evaluating GI-199 is based on the collective results, not the results for any particular plant.

Region I
Indian Point 2
Indian Point 3
Limerick 1
Limerick 2
Peach Bottom 2
Peach Bottom 3
Seabrook 1

Region II
Crystal River 3
Farley 1
Farley 2
North Anna 1
North Anna 2
Oconee 1
Oconee 2
Oconee 3
Saint Lucie 1
Saint Lucie 2
Sequoyah 1
Sequoyah 2
Summer
Watts Bar 1

Region III
Dresden 2
Dresden 3
Duane Arnold
Perry 1

Region IV
River Bend 1
Wolf Creek 1

Q24. There was a recent Part 21 (60-Day Interim Report) Notification concerning seismic input for control rods that might lead to a failure to scram at Boiling Water Reactors (BWRs). Was this information included in the GI-199 Safety/ Risk Assessment (S/RA)? Could this information change the results of the S/RA?

A24. On September 3, 2010, General Electric Hitachi (GEH) Nuclear Energy submitted a 10CFR50 Part 21 Notification regarding a failure to include seismic input in reactor control blade customer guidance for BWRs. BWRs remain safe because (1) control rods are expected to fully or partially insert even with channel-control rod interference, (2) operators will still have the ability to manually scram partially inserted rods, and (3) the limited time spent at conditions where the failure to scram could occur (low reactor pressure). NRR has been following this Part 21 issue and has determined that the GEH has provided effective interim guidance to the affected licensees that experience channel-control rod interference, and that additional guidance detailed in the Part 21 notice, provides licensees with conservative strategies to assist in the insertion of control rods under low reactor pressure conditions.

The GI-199 S/RA was completed in August, prior to the Part 21 notice. Considering the above, information from the new Part 21 notice would not be expected to change the conclusions of the GI-199 S/RA. Information from this Part 21 Notification will be considered in future efforts to address GI-199.

From:	Jessup, William
To:	Leeds, Eric; Grobe, Jack; Boger, Bruce; Hiland, Patrick; Skeen, David; Giitter, Joseph; Howe, Allen; Nelson, Robert; Holian, Brian; Galloway, Melanie; Evans, Michele; Lubinski, John; Thomas, Brian; "Cunningham, Mark"; Lee, Samson; Ruland, William; Bahadur, Sher; Brown, Frederick; Cheok, Michael; McGinty, Tim; Blount, Tom; Quay, Theodore
Cc:	Khanna, Meena; Franke, Mark; Sykes, Marvin; Rich, Daniel; Lake, Louis; Farzam, Farhad; Saba, Farideh; Thomas, George; Wilson, George; Lupold, Timothy; Auluck, Rajender; Kuntz, Robert; Bernardo, Robert; Bamford, Peter; Rezai, Ali; Ashar, Hansraj; White, John; Brand, Javier; Sheikh, Abdul; Manoly, Kamal; "nausdj@ornl.gov"; Graves, Herman; Broaddus, Doug; Jolicoeur, John; Orders, William; Hipschman, Thomas; Castleman, Patrick; Snodderly, Michael; Franovich, Mike
Subject:	OUO/SI: Status of Crystal River Unit 3 Containment Delamination Issue (March 18, 2011)
Date:	Friday, March 18, 2011 3:15:33 PM
Attachments:	Crystal River Unit 3 Delamination Locations.pdf

OFFICIAL USE ONLY/SENSTIVE INFORMATION

Crystal River Delamination Issue Update – March 18, 2011

- On March 14, 2011, Progress Energy identified indications of an additional separation or gap (delamination) resulting from the repair work on the containment wall of Crystal River Unit 3.
- Further re-tensioning activities, which were being conducted as part of the repair, have been suspended.
- Indications of a potential anomaly were recorded during repair activities by acoustic monitors and strain gauges. The licensee's initial non-destructive testing (impulse response (IR) testing) confirmed a delamination in Bay 5-6 of the containment.
- In contrast to the initial delamination that was identified by the licensee in Bay 3-4 in 2009, Bay 5-6 is adjacent to the fuel transfer building to the spent fuel pool and contains the two fuel transfer tubes (**site graphic with bay locations identified is attached**). The location of the delamination in Bay 5-6 will increase the difficulty associated with any repair.
- The total extent of delamination is not yet known. The licensee is continuing to perform root cause and extent of condition assessments.
- The licensee is in the process of conducting IR testing on the other containment bays.
- For the current mode (Mode 5), containment integrity relies only on an intact containment liner. The liner was inspected with no problems noted. There is currently no impact on the spent fuel pool. NRC inspectors have walked down the outside of Bay 5-6 and did not identify any noticeable cracking on the surface.
- On March 15, 2011, the licensee submitted a voluntary non-emergency event notification and issued a press release.
- The licensee indicated that this additional delamination issue will cause a "significant delay" in their restart schedule.
- The March 22, 2011, public meeting to discuss plant restart has been cancelled.
- An internal Petition Review Board meeting related to the 2.206 petition is scheduled for March 26, 2011.

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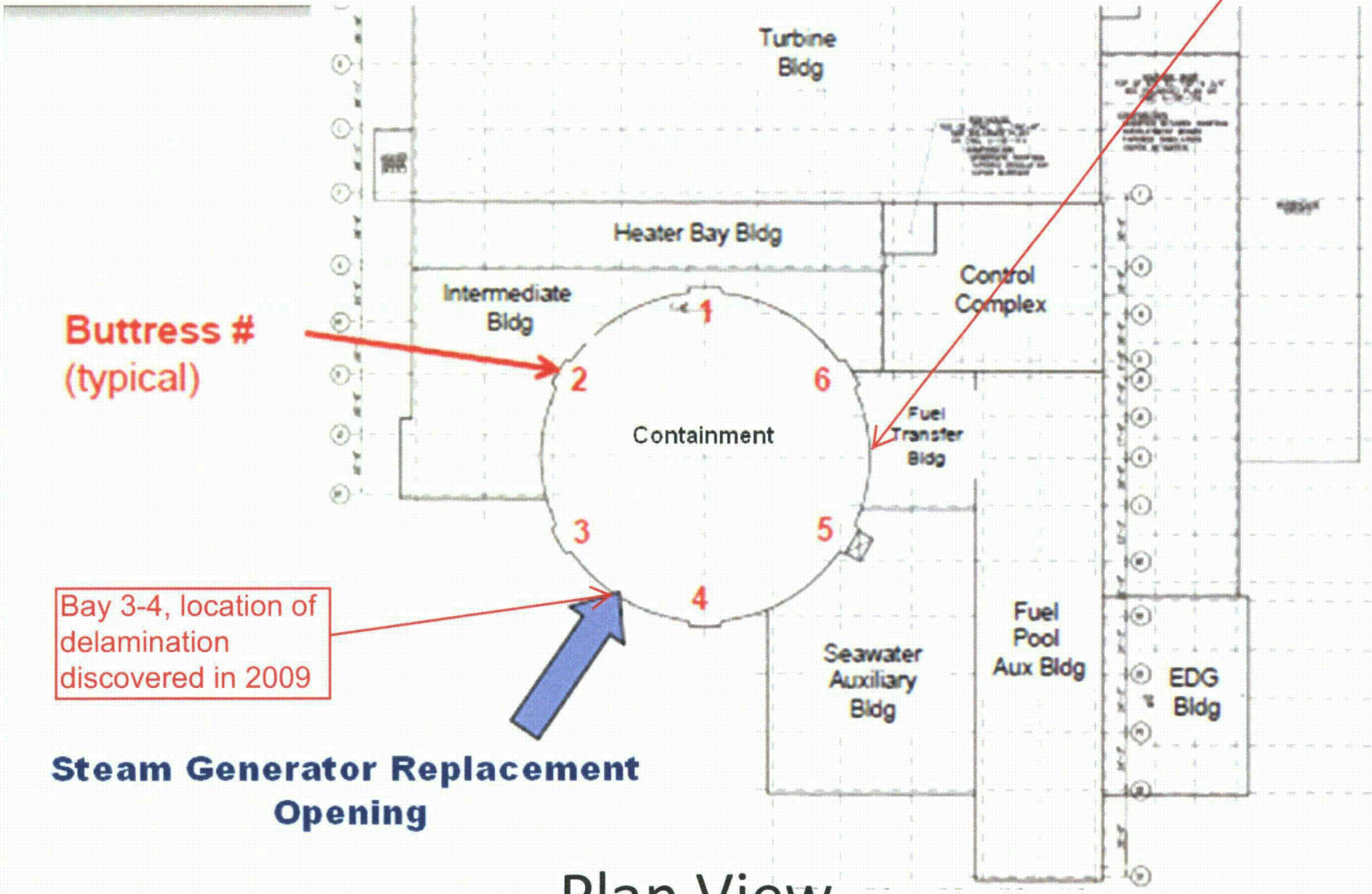
Overview



Plan View

Kanney, Joseph

From: Kanney, Joseph
Sent: Friday, March 18, 2011 4:46 PM
To: See, Kenneth; Nicholson, Thomas; Dion, Jeanne; Cook, Christopher
Cc: Randall, John; Ott, William; Chokshi, Nilesh; Raione, Richard
Subject: RE: Revision to PMP Narrative

A reasonable question that an observer of the tsunami in Japan might have goes something like this: I observed the tremendous amount of debris being carried along with the water. Are NPPs in the US sufficiently protected from flood-associated debris flow?

Joe

From: See, Kenneth
Sent: Friday, March 18, 2011 4:34 PM
To: Kanney, Joseph; Nicholson, Thomas; Dion, Jeanne; Cook, Christopher
Cc: Randall, John; Ott, William; Chokshi, Nilesh; Raione, Richard
Subject: RE: Revision to PMP Narrative

I agree. I think the term "most" of the existing plants were designed for the PMP/PMF is applicable, but not all.

Ken

From: Kanney, Joseph
Sent: Friday, March 18, 2011 4:26 PM
To: Nicholson, Thomas; See, Kenneth; Dion, Jeanne
Cc: Randall, John; Ott, William; Chokshi, Nilesh; Raione, Richard
Subject: RE: Revision to PMP Narrative

I think perhaps we are getting carried away with our focus on PMP. PMP is only one part of the flooding analysis and probably the not weakest link. Recall that the design basis floods for many of the existing plants were determined from area-discharge envelope curves, not from an analysis that routed the PMP through the basin and down to the site. I'm more comfortable with the level of conservatism in PMP than I am with the level of conservatism in the envelope curve approach.

-- Joe

Joseph Kanney, Ph.D.
Hydrogeologist
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Mail Stop CSB-2A07M
11555 Rockville Pike
Rockville, MD 20852-2738

joseph.kanney@nrc.gov
Phone: 301-251-7600
Fax: 301-251-7422

0016

From: Nicholson, Thomas
Sent: Friday, March 18, 2011 4:01 PM
To: See, Kenneth; Dion, Jeanne
Cc: Randall, John; Kanney, Joseph; Ott, William; Chokshi, Nilesh; Raione, Richard
Subject: Revision to PMP Narrative

Ken:

I just spoke to John England, U.S. Bureau of Reclamation concerning our response to the NRO information request. He would like a one-word addition to the narrative to say the Southeast U.S. rather than just the U.S.

This is his suggested wording:

Some of the Reports from the National Weather Service used to estimate the design precipitation are 30-40 years old. Are these estimates still valid?

The NRC has funded research by the U.S. Bureau of Reclamation to review the information and methods developed by the National Weather Service and the U.S. Army Corps of Engineers (HMR 51), focusing on South and North Carolina. To date, reviews of precipitation records from extreme storm events (e.g., tropical storms, hurricanes) since the publication of HMR 51 does not indicate any exceedance or potential for exceedance of those precipitation (PMP) estimates in this region. We have not seen any information or data that would indicate that HMR precipitation (PMP) estimates for the **Southeast** U.S. have been exceeded.

Please make whatever changes are possible at this time.

Thanks ..... Tom

****
Thomas J. Nicholson, Senior Technical Advisor
U.S. Nuclear Regulatory Commission
Office of Nuclear Regulatory Research
Mail Stop CSB 2-A07
11555 Rockville Pike
Rockville, MD 20852
Tel: (301) 251-7498
Fax: (301) 251-7422
E-mail: Thomas.Nicholson@nrc.gov
****

Lee, Richard

From: Gauntt, Randall O [rogaunt@sandia.gov]
Sent: Monday, March 21, 2011 8:57 PM
To: Lee, Richard
Subject: Re: Japan related contracting to be oversee and managed by FSTB

Give me a call at 505 263 6849

You probably called my personal cell phone which I lost last week!

----- Original Message -----
From: Lee, Richard [mailto:Richard.Lee@nrc.gov]
Sent: Monday, March 21, 2011 06:54 PM
To: Gauntt, Randall O
Subject: Japan related contracting to be oversee and managed by FSTB

Randy:

I left message on your cell phone. So far, I have not heard back. A new contract will be initiated and funded this week from our Branch. I need to know what analysis is on going and what been planned in the very near future that will serve the basis for writing up the tasks. There should be a task for on-call assistant to answer questions we are getting everyday.

If you think you do not need to contact us, then I assume you do not need us to do the funding. You can deal with whoever you are dealing with now.

Richard

00/7

Lee, Richard

From: Gauntt, Randall O [rogaunt@sandia.gov]
Sent: Monday, March 21, 2011 9:52 PM
To: Lee, Richard
Subject: RE: Japan related contracting to be oversee and managed by FSTB

Richard,

Analyses that are ongoing or needed:

1) Evaluation of Fukushima SBO sequences. These sequences are long-term station blackout sequences with variations between reactors that have isolation condenser designs or turbine-driven RCIC designs. Analyses are needed to evaluate effect of inital loss of offsite power, shutdown and emergency generator operation followed by flooding induced site blackout with station battery depletion or interruption for battery replacement, accounting for duration of RCIC operation. Analyses are required to evaluate operator actions and for suppression pool saturation and extent of core damage, hydrogen generation, and fission product release expected. In addition, evaluation of steam, hydrogen and fission product release to reactor building for containment venting operations, and source term release to environment is required in order to validate best estimate models of broad class of US BWR fleet.

2) Based on insights gained from task 1, perform evaluation of select representative reactors in US fleet of BWR's, initially for Mark-1 containments (with/without isolation condenser design) and subsequently for Mark-III designs that do not have SBO hydrogen controls in non-inerted containments.

3) Perform accident analysis of spent fuel pool damage in Fukushima Unit 4 pool using fission product inventories available from GE or ORNL considering possible seismic damage and water loss that may have accelerated boildown and fuel damage. Evaluate releases from the pool accidents and compare to available data on dose rates and environmental releases, atmospheric transport, land contamination and health effects.

4) Based on task 3, project this understanding to the relatively more densly populated US fuel pools and evaluate potential source term implications of current pools considering various options for transferring fuel to dry storage.

From: Lee, Richard [Richard.Lee@nrc.gov]
Sent: Monday, March 21, 2011 6:54 PM
To: Gauntt, Randall O
Subject: Japan related contracting to be oversee and managed by FSTB

Randy:

I left message on your cell phone. So far, I have not heard back. A new contract will be initiated and funded this week from our Branch. I need to know what analysis is on going and what been planned in the very near future that will serve the basis for writing up the tasks. There should be a task for on-call assistant to answer questions we are getting everyday.

If you think you do not need to contact us, then I assume you do not need us to do the funding. You can deal with whoever you are dealing with now.

Richard

Lee, Richard

From: Gauntt, Randall O [rogaunt@sandia.gov]
Sent: Tuesday, March 22, 2011 7:42 PM
To: Humphries, Larry Laron
Cc: Pickering, Susan Y; Orrell, Stanley A; Tinkler, Charles; Lee, Richard
Subject: RE:

Short answer is - we are not likely to hand over models to ORNL. Between you and me, a large reason is that they will try to run the models and either a) have a lot of instructional and explainational questions requiring our nursing, b) have no end of questions on the code predictions and why, or c) generate public issues potentially in contrast with our ongoing work requiring our energy to explain the differences. All dilute our limited resources to respond to an unacceptable level.

This is the reality - in addition to the fact that we don't want our work over the past 10 years to be credited to ORNL.

You could conver to Juan that Sandia maintains tight configuration control over the decks and cannot disperse decks without NRC direction and participation. Prioliferation of decks, even within our own ranks, has to be managed. Plus, all of the pragmatic reasons cited above. None of this would seem reasonable to ORNL, and I don't know how to respond aside from ignoring the requests mainly due to distraction.

Randy

From: Humphries, Larry Laron
Sent: Tuesday, March 22, 2011 3:52 PM
To: Gauntt, Randall O
Subject: FW:

Perhaps you can help Juan or if you like, refer him to Wagner.

-Larry

From: Carbajo, Juan J. [mailto:carbajojj@ornl.gov]
Sent: Saturday, March 19, 2011 12:22 PM
To: Humphries, Larry Laron
Subject:

Hi Larry,
It has been some time since we communicated, after my trip to SNL last January.
I was waiting for the advice of Randy Cole on modeling the split of blowdown stream into water and steam.
I am contacting you this time to ask you for a different favor.
We know here at ORNL that SNL has been involved in MELCOR calculations for the Fukushima reactor, in particular for the spent fuel pool of Unit 4.
I am sure you know all of this work.
We are here also doing some work for the Japanese reactors.
Will it be possible to get the actual dimensions of the pool of that reactor? And the fuel heat load?
Do you have a MELCOR model of the spent fuel in the pool?
Thanks and best regards.

Juan J. Carbajo
Oak Ridge National Laboratory
carbajojj@ornl.gov
(865)-574-5856

Lee, Richard

From: Basu, Sudhamay
Sent: Wednesday, March 23, 2011 12:06 PM
To: 'Farmer, Mitchell T.'; Tinkler, Charles
Cc: Lee, Richard; Grandy, Christopher
Subject: RE: Response to your questions..

Debris depth looks about right based on 50% porosity.

From: Farmer, Mitchell T. [mailto:farmer@anl.gov]
Sent: Wednesday, March 23, 2011 11:01 AM
To: Tinkler, Charles
Cc: Lee, Richard; Basu, Sudhamay; Grandy, Christopher
Subject: Response to your questions..

Hi Charlie,
Sorry for my delay; got your phone message and have been really swamped. I think I had two questions, I added a bit more based on aniticpation of your questions:

1) what is debris depth in Pool 4:

Input: 1300 assemblies, or about 2.3 full core loads (584 per load). Total Fuel mass per load is ~ 150 MT, toss in 15% more for cladding, so fuel + cladding = 170 tons. So, total weight is 2.3*170 = 390 tons. Fuel Density of Fuel + cladding ~ 9000 kg/m3, and so the volume of fuel + cladding is about 43 m**3. Pool is 35 x 40 ft which is 10 x 12 m = 120 m**2. So, theoretical depth is about 35 cm; but if rubbalized about 70 cm.

2) Decay heat and fluxes and coolability -

3 MW decay heat right now /120 m**2 = 25 kW/m**2 heat flux effective in bed assuming its uniformly distributed over the bottom of the pool. The rubbilized material will be exceptionally porous and so will have a very high dryout limit (based lipinki correlatoin, easliy 1 MW/m**2 dryout limit) So if you can keep water on this it should be readily coolable. I

3) What if it goes dry - porous bed can convect to air as well, but if it were to go into mcci the quasi-steady relationship between ablation rate and heat flux applied ot concrete is

(conc. Density)*(concrete decompositoin enthalpy)*(erosion rate) = heat flux into concrete.

We have a lot of test data that says about 2/3 heat goes up and 1/3 goes down, and so the heat flux down is going to be about (1/3)*25 ~ 8 kW/m2. For siliceous concrete, density ~ 2300 kg/m**3, decomposition enthalpy ~ 1.6 MJ/kg, and so working through the math yields an erosion rate of about 19 cm/day (0.6 ft/day)

Note of caution though: for siliceous concrete the heat flux tends to get focused to the side more than down. So the lateral rate could be up to 1.8 ft/day.

If you want to discuss please let me know.

Mitch

00/10

Lee, Richard

From: Wagner, Katie
Sent: Wednesday, March 23, 2011 5:24 PM
To: Lee, Richard
Subject: RE: RST Requested Evaluations to be Performed by RES
Attachments: image001.gif

Ok

From: Lee, Richard
Sent: Wednesday, March 23, 2011 5:22 PM
To: Wagner, Katie
Subject: RE: RST Requested Evaluations to be Performed by RES

Katie:

I have responded to all 3 items. I plan to send additional info. on #4. Will let you know after I sent them.

Richard

From: Wagner, Katie
Sent: Wednesday, March 23, 2011 4:50 PM
To: Lee, Richard
Subject: RE: RST Requested Evaluations to be Performed by RES

Richard - Is the item below "Completed" or still "Pending"? – Thanks, Katie

36	3/23/2011	Ops Center (RST) ! NEW	Email: RST01 Hoc	Richard Lee x7526	FSTB	To investigate the topics related to queries 1, 3, and 4 from the NRC Site Team located over in Japan. 1. Sea water injection continues to reactors 1-3. NISA is concerned about the radiolytic disassociation of H2 and O2. NISA would like NRC's perspective on the significance of this concern and how to treat this concern as they transition to freshwater injection. 3. NISA is conducting simulations to project the extent of damage to fuel in the reactors. Has NRC developed any views on the extent of fuel damage? 4. NISA is interested to obtain any reference material regarding core-concrete interaction (not because they think they have a current issue but against that eventuality) including the conditions under which that occurs and any associated data.	Respond with answers to the 3 questions raised.	Pending

---

From: Lee, Richard
Sent: Wednesday, March 23, 2011 11:07 AM
To: RST01 Hoc
Cc: Wagner, Katie
Subject: RE: RST Requested Evaluations to be Performed by RES

This is the response to item 4. Please forward this to our team in Japan. If we get more information on this subject, we will send them later.

From: RST01 Hoc
Sent: Wednesday, March 23, 2011 5:39 AM

To: Lee, Richard
Cc: Gibson, Kathy; RST01 Hoc
Subject: FW: RST Requested Evaluations to be Performed by RES

From: Hoc, RST16
Sent: Wednesday, March 23, 2011 5:37 AM
To: RST01 Hoc
Subject: RST Requested Evaluations to be Performed by RES

RST Coordinator,

Please forward the e-mail text below to Richard Lee and cc: Kathy Gibson from RES.

Thank you,

RST Communicator

****

Dear Mr. Lee,

The Reactor Safety Team (RST) in the Operations Center has been given your name as the lead Point of Contact (POC) for the Office of Research to manage all incoming requests from the RST.

As such, here is a request to investigate the topics related to queries 1, 3, and 4 from our Site Team located over in Japan. FYI, this request was entered into the WEB EOC Task Tracker software as Record # 2098 on 3/22/2011 @ 11:42 AM. Items 2 and 5 have been evaluated by our team and other sources and have been determined to be complete and require no further action.

Per Site Team e-mail, they would like RST to provide input on the following:

1. Sea water injection continues to reactors 1-3. NISA is concerned about the radiolytic disassociation of H2 and O2. NISA would like NRC's perspective on the significance of this concern and how to treat this concern as they transition to freshwater injection.
2. At what point does salt deposits become a problem for flow during pending freshwater injection?
3. NISA is conducting simulations to project the extent of damage to fuel in the reactors. Has NRC developed any views on the extent of fuel damage?
4. NISA is interested to obtain any reference material regarding core-concrete interaction (not because they think they have a current issue but against that eventuality) including the conditions under which that occurs and any associated data.
5. In addition to the H2/O2 disassociation in item 1 above, they are concerned that there may be residual H2 in the containments and welcome NRC's thoughts on how to treat such a condition.

The RST requests an progress update via e-mail, at least once daily in order for us to be able to track your progress for each of the three issues.

Thank you for your time and consideration.

Bill Roggenbrodt
RST Communicator
301-816-5504

Lee, Richard

From: Aissa, Mourad
Sent: Wednesday, March 23, 2011 4:51 PM
To: Wagner, Katie
Cc: Lee, Richard
Subject: RE: Ops Center Action Item for Ticketing

Katie,
I don't mind taking this one. I will talk with Richard about it.
Thanks
Mourad

From: Wagner, Katie
Sent: Wednesday, March 23, 2011 2:05 PM
To: Aissa, Mourad
Cc: Lee, Richard
Subject: FW: Ops Center Action Item for Ticketing

Mourad,

Are you able to answer the following question from the Ops Center?:

-What should the Japanese be considering with respect to criticality prevention and decay heat removal during the entombment period?

I think Richard is in a meeting and this is urgent, otherwise I would let him take the lead on figuring what belongs to who.

Thanks,
Katie

From: Gibson, Kathy
Sent: Wednesday, March 23, 2011 1:53 PM
To: Lee, Richard; Wagner, Katie
Subject: FW: Ops Center Action Item for Ticketing

More info

From: Bowman, Gregory
Sent: Wednesday, March 23, 2011 1:45 PM
To: Richards, Stuart
Cc: Case, Michael; Gibson, Kathy; Scott, Michael; Rini, Brett
Subject: RE: Ops Center Action Item for Ticketing

Sorry if my e-mail wasn't clear...we only need a response to the four questions from the 11:23 am e-mail from RST01 Hoc.

I would recommend coordinating with George Deegan and Larry Camper in FSME. I was CC'd on the attached e-mail from Larry a little while ago, and it looks like they might be best equipped to cover most (or all) of the questions.

0912

From: Richards, Stuart
Sent: Wednesday, March 23, 2011 1:07 PM
To: Bowman, Gregory
Cc: Case, Michael; Gibson, Kathy; Scott, Michael
Subject: RE: Ops Center Action Item for Ticketing
Importance: High

Greg

Are we trying to respond to all of the issues in Dan Dorman's e-mail, or just the items highlighted in the 11:23 am e-mail from RST01 Hoc ?

Thanks
Stu

From: Bowman, Gregory
Sent: Wednesday, March 23, 2011 12:08 PM
To: Case, Michael; Richards, Stuart; Gibson, Kathy; Scott, Michael
Subject: FW: Ops Center Action Item for Ticketing
Importance: High

FYI – I just sent this information request from the Ops Center to Brett. I sent it to him because I wasn't sure which division in RES would be the right one to help with this, but I figured I'd pass it along to you, as well, given that the Ops Center is looking for a response by the end of the day.

From: Bowman, Gregory
Sent: Wednesday, March 23, 2011 11:51 AM
To: Rini, Brett; Deegan, George
Cc: Frazier, Alan; Brock, Kathryn
Subject: FW: Ops Center Action Item for Ticketing
Importance: High

Brett and George,

We got the request below from the Ops Center. We think there should be one coordinated response back to the Ops Center from RES and FSME, but none of us are sure which division would be best able to respond. Can you help with this? Note that the Ops Center has asked for a response by 18:00 tonight.

Greg

From: RST01 Hoc
Sent: Wednesday, March 23, 2011 11:23 AM
To: Andersen, James; Muessle, Mary
Cc: Brown, Frederick
Subject: Ops Center Action Item for Ticketing

Jim and Mary,

Per Fred Brown, RST Director here in the Ops Center, Please ticket the following item to RES and FSME:

"Respond to Dan Dorman's email on long-term issue questions from Japan. Provide responses or estimates of when the responses can be expected to Dan by 18:00 EDT. If additional information is needed, let the site team know of any questions that can be brought back to NISA.

-Regarding the best type of enclosure for the plant, does NRC have any thoughts? Do we have any regulations applicable to this condition or thoughts on the role of the regulatory authority in this decision?

-What licensing requirements apply to decommissioning and regulatory review of the decommissioning plan?

-What should the Japanese be considering with respect to criticality prevention and decay heat removal during the entombment period?

-The NRC's TMI fact sheet notes that the first manned entry into the Unit 2 reactor building occurred after a venting of Krypton in July 1980 (16 months after the accident). WHat actions occurred during those 16 months that could inform their planning?"

Dan Dorman's email pasted below:

From: Dorman, Dan
Sent: Tuesday, March 22, 2011 3:05 AM
To: OST01 HOC; Casto, Greg; Monninger, John; ET07 Hoc
Subject: RE:

Additional tasks from meeting with NISA et al this morning. Lower prirority than the Cabinet level issues we just discussed on the phone, but any responses available by 1800 EDT on 3/22 would be greatly appreciated along with an estimate of when the remainder may be expected. If you need additional info, please identify any questions we can bring back to NISA (keeping in mind please that their plant data is also very limited, i.e., keep your data expectations modest).

1. Sea water injection continues to reactors 1-3. NISA is concerned about the radiolytic disassociation of H2 and O2. NISA would like NRC's perspective on the significance of this concern and how to treat this concern as they transition to freshwater injection.
2. At what point does salt deposits become a problem for flow during pending freshwater injection?
3. NISA is conducting simulations to project the extent of damage to fuel in the reactors. Has NRC developed any views on the extent of fuel damage?
4. NISA is interested to obtain any reference material regarding core-concrete interaction (not because they think they have a current issue but against that eventuality) including the conditions under which that occurs and any associated data.
5. In addition to the H2/O2 disassociation in item 1 above, they are concerned that there may be residual H2 in the containments and welcome NRC's thoughts on how to treat such a condition.

NISA is beginning to look at long term issues and has the following Qs in this area (note some of these may only apply to Japan's regulatory framework, but if we have insights from our post-TMI actions they would be greatly appreciated):
6. Regarding the best type of enclosure for the plant, does NRC have any thoughts? Do we have any regulations applicable to this condition or thoughts on the role of the regulatory authority in this deicision?
7. What licensing requirements apply to decommissioning and regulatory review of the decommissioning plan?
8. What should they be considering with respect to criticality prevention and decay heat removal during the entombment period?
9. The NRC's TMI Fact Sheet notes that the first manned entry into the Unit 2 reactor building occurred after a venting of Krypton in July 1980 (16 months after the accident). What actions occurred during those 16 months that could inform their planning?

Regarding the spent fuel pools, NISA asserted that the Unit 1 SFP is above TAF with over 20 days margin due to low decay heat. They are not injecting to the UNit 1 SFP. For Unit 2, they are injecting seawater to the SFP via installed piping. For Units 3 and 4, they are spraying from pumper trucks within the RBs to put water on the top of the pools (In response to a question, they indicated that these sprays were put in place after the explosive events in those buildings.) Based on this information, NISA is assuming that the SFPs are all below 100C. The team here has questions relative to the latter buildings and other information available, for example, lack of visual evidence of steaming on Unit 4. We would appreciate HQ's thoughts on the SFPs and apparent inconsistencies with the status provided by NISA.

Dan Dorman

Thanks,
Eric Thomas
RST Coordinator

Lee, Richard

From: RST01 Hoc
Sent: Wednesday, March 23, 2011 5:39 AM
To: Lee, Richard
Cc: Gibson, Kathy; RST01 Hoc
Subject: FW: RST Requested Evaluations to be Performed by RES

From: Hoc, RST16
Sent: Wednesday, March 23, 2011 5:37 AM
To: RST01 Hoc
Subject: RST Requested Evaluations to be Performed by RES

RST Coordinator,

Please forward the e-mail text below to Richard Lee and cc: Kathy Gibson from RES.

Thank you,

RST Communicator

****

Dear Mr. Lee,

The Reactor Safety Team (RST) in the Operations Center has been given your name as the lead Point of Contact (POC) for the Office of Research to manage all incoming requests from the RST.

As such, here is a request to investigate the topics related to queries 1, 3, and 4 from our Site Team located over in Japan. FYI, this request was entered into the WEB EOC Task Tracker software as Record # 2098 on 3/22/2011 @ 11:42 AM. Items 2 and 5 have been evaluated by our team and other sources and have been determined to be complete and require no further action.

Per Site Team e-mail, they would like RST to provide input on the following:

1. Sea water injection continues to reactors 1-3. NISA is concerned about the radiolytic disassociation of H2 and O2. NISA would like NRC's perspective on the significance of this concern and how to treat this concern as they transition to freshwater injection.
2. At what point does salt deposits become a problem for flow during pending freshwater injection?
3. NISA is conducting simulations to project the extent of damage to fuel in the reactors. Has NRC developed any views on the extent of fuel damage?
4. NISA is interested to obtain any reference material regarding core-concrete interaction (not because they think they have a current issue but against that eventuality) including the conditions under which that occurs and any associated data.
5. In addition to the H2/O2 disassociation in item 1 above, they are concerned that there may be residual H2 in the containments and welcome NRC's thoughts on how to treat such a condition.

The RST requests an progress update via e-mail, at least once daily in order for us to be able to track your progress for each of the three issues.

Thank you for your time and consideration.

0013

Bill Roggenbrodt
RST Communicator
301-816-5504

Marksberry, Don

From: Helton, Donald
Sent: Wednesday, March 23, 2011 3:06 PM
To: Tinkler, Charles; Marksberry, Don; Salay, Michael; Schaperow, Jason; Esmaili, Hossein
Subject: RE: Questions from our Hallway Discussion
Attachments: Suggested Questions of Interest for the RST Severe Accident Analyst - 032311.docx

All – An updated version based on input from Charlie.

Charlie – I included input from your markup and your email that was sent simultaneous to me sending out the earlier version. I did not understand why you deleted the venting question, so I left it in, but moved it further down the list

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From: Helton, Donald
Sent: Wednesday, March 23, 2011 2:36 PM
To: Tinkler, Charles; Marksberry, Don; Salay, Michael; Schaperow, Jason; Esmaili, Hossein
Subject: Questions from our Hallway Discussion

All,

These are the questions that I compiled from our hallway discussion this morning…

Don

Don Helton
Senior Reliability & Risk Engineer
Division of Risk Analysis
NRC Office of Nuclear Regulatory Research
Physical address: 21 Church Street, CSB4-C9, Rockville, MD 20850
Postal address: US NRC / MS CSB4-C7M / Washington, DC 20555
Ph: 301 251-7594, Donald.Helton@nrc.gov

00/14

Suggested Questions of Interest for the RST Severe Accident Analyst:

Reminder: The RST Severe Accident Analyst should touch base with the PMT RASCAL analyst at the beginning of each shift, and periodically throughout the shift (recall that the RST/PMT liaison positions are not being staffed).

Reactors:

- (Potential) primary containment failure:
  - What is the basis for reports that the Unit 3 primary containment may be failed? Reports had previously indicated that the Unit 3 primary containment was pressurized, well after the initiating events and the hydrogen deflagration/detonation in the reactor building.
  - What is the current estimate on containment integrity for Units 1 and 2?
- Hydrogen:
  - What particular concern do the Japanese have regarding hydrogen accumulation on Unit 2? Is this a concern about primary containment becoming de-interted in the long-term, or due to containment failure, or is it a concern regarding a deflagration/detonation in the reactor building?
- Water injection:
  - Various RPV water levels in Units 1-3 have been reported, many suggesting that the level is being maintained around core mid-plane. Is this the case, and if so, what is the rationale?
  - What is the status (and if possible timeline) regarding containment flooding, including current water level?
- Water spraying:
  - Is water spraying also intended to cool the outside of the containment (assuming the primary purpose is spent fuel pool makeup)?
- Venting:
  - What were the vent paths used for the primary containment venting: drywell versus suppression pool airspace?
- Data requests:
  - For each reactor, could we get a plot (or enough data points to make a plot) for each of the critical indications listed below? If not, could we at least get a current, verified reading for these?
    - Reactor vessel pressure
    - Reactor vessel water level
    - Primary containment pressure
    - Primary containment (drywell/wetwell) hydrogen concentration

Spent Fuel Pools:

- Coolant inventory:
  - How much water is in each of the spent fuel pools for units 1, 2, 3, and 4? Are there any reliable temperature measurements?

- Please provide any timeline information available regarding fuel uncovery and makeup/spray injection.
- Did the concrete pump used to put water in the unit 4 spent fuel pool have a camera on it? If so, what did the video show?
  - Why was the pump stopped after delivering 150 tons of water? 150 tons is not enough to fill the spent fuel pool.

Other:

- Are any of the NRC in-country representatives stationed in the TEPCO or NISA Emergency Operations Facilities? If so, are they in a position to relay some of the measurements that are available (such as ones that would answer the above questions)?
- Has any assessment been done of the I-131/Cs-137 ratios that are being observed via field measurements, to determine if they originated from a reactor release versus a spent fuel pool release?
- What is the status of replacing spray/injection sources with potable water, rather than continuing to use seawater?
- Has water been observed draining from the loading bay door (or other areas) of any of the reactor buildings, coincident with the spraying operations?

Marksberry, Don

From: Marksberry, Don
Sent: Thursday, March 24, 2011 4:48 PM
To: Arndt, Steven
Subject: Some questions

1. Any concerns that Unit 3 primary containmet may be failed? (we heard both ways)

2. What is the zero reference for RPV level indication that has been reported by NISA press releases? (The press release has a table note that indicates that zero is top of active fuel---this is different that typical U.S. GE designs. Indication of -1,700 mm is below 2/3 core for zero of TAF)

3. Is water spraying also intended to cool the outside of the containment?

4. What were the vent paths used for primary containment venting: drywell vs. suppression pool airspaces?

5. Any recent concerns aboout spent fuel pool coolant inventory?

6. Has any assessment been done of the I-131/Cs-137 ratios that are being observed via field measurements (i.e., eactor vs. spent fuel releases)

00/15

Lee, Richard

From: Sheron, Brian
Sent: Friday, March 25, 2011 1:05 PM
To: Lee, Richard
Subject: FW: Tank Waste Retrieval, Processing, and On-site Disposal at Three Department of Energy Sites
Attachments: SEEPEX transfer pump.jpg

From: Per F. Peterson [mailto:peterson@nuc.berkeley.edu]
Sent: Friday, March 25, 2011 12:31 PM
To: Brinkman, Bill; Hurlbut, Brandon; Sheron, Brian; Poneman, Daniel; 'harold.mcfarlane@inl.gov'; Adams, Ian; 'jholdren@ostp.eop.gov'; Kelly, John E (NE); 'john.grossenbacher@inl.gov'; Owens, Missy; 'peterson@nuc.berkeley.edu'; Lyons, Peter; 'phillip.finck@inl.gov'; 'rgarwin@ostp.eop.gov'; 'RJBudnitz@lbl.gov'; 'ronaldo.szilard@inl.gov'; SCHU; Aoki, Steven; Koonin, Steven; 'Steven_A._Fetter@ostp.eop.gov'; DAgostino, Thomas
Subject: Tank Waste Retrieval, Processing, and On-site Disposal at Three Department of Energy Sites

Following up on the discussion during our conference call, I wanted to point everyone to some information about the technologies that were developed in the U.S. to pump, store, and treat radioactive effluents from defense reprocessing facilities. I've had a lot of experience with these technologies, since one of my my first jobs before going to grad school was to work at Bechtel on the design of the Defense Waste Processing Facility at SRS.

An important capability that will be needed for the clean up and remediation of Units 1-3 at Fukushima will be the capacity to flush fresh water through the reactor vessel and containment and discharge it into some type of collection tank, similar to the waste tanks that exist at SRS, Hanford, and INL. The practice at these sites is to simply bury the transfer lines 1 to 2 m below ground, which provides appropriate shielding for personnel during transfers of radioactive fluids. At Fukushima, the simpler approach may be to stack shielding material on top of transfer lines.

An evaporator system will be needed to concentrate the salt and solids in these flush water streams, and the resulting water can be recollected to use for further flushing and cooling operations. The tanks will need to be designed so that heat removal can be provided, as is the case at SRS. Eventually the material collected in these tanks will need to be further processed into appropriate waste forms for disposal, and methods for doing that can be derived from the similar technologies developed at SRS for processing salt and sludge from waste tanks. Sufficient flushing of the primary system and containment should help bring radiation levels down and allow access to remove damaged fuel from the reactors.

Chapter 3, "Tank Waste Retrieval", in the following National Academies report provides a brief overview of the technology.

"Tank Waste Retrieval, Processing, and On-site Disposal at Three Department of Energy Sites: Final Report," National Academies Press, 2006.

http://www.nap.edu/catalog.php?record_id=11618#toc

Here is a photo from the report showing a typical transfer pump. It is designed like a sump pump and can be lowered down into a tank to pump out and transfer radioactive fluids.

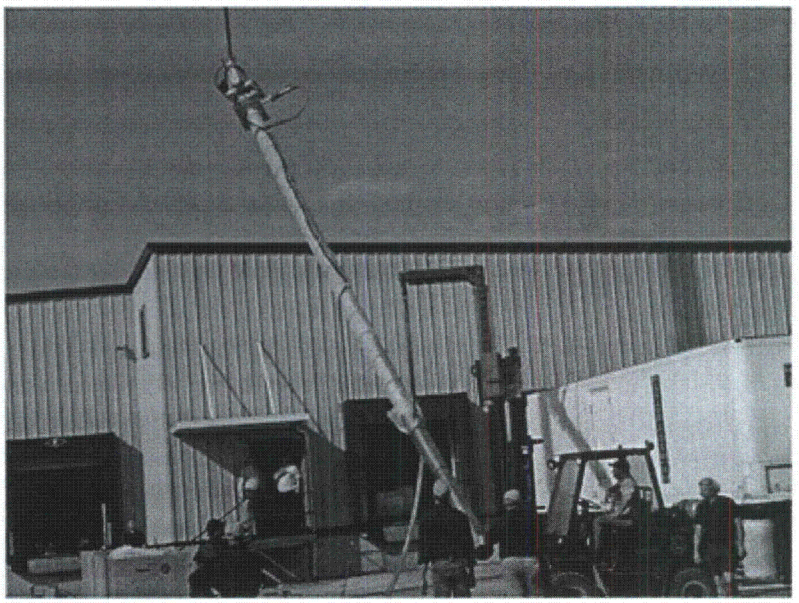

FIGURE III-1 The SEEPEX transfer pump before installation (pg. 35).

I see the key issues being how to figure out how to perform let down and injection to flush the salt and contaminated fluids out of the reactor and containment systems. I like very much Dick's idea that one can just use pressure to force fluids out. Injection of fresh water can be done at the same time. The key thing that is needed is a nice big tank to discharge the effluent into and the capability to evaporate water away to concentrate it, and to recycle the water back into the flushing operation. There is no rocket science here, since this is something that was done extensively in the U.S..

-Per
--

Per F. Peterson
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http://www.nuc.berkeley.edu/People/Per_Peterson

Lee, Richard

From: Binder, Jeffrey L. [binderjl@ornl.gov]
Sent: Friday, March 25, 2011 7:48 AM
To: 'farmer@anl.gov'; 'johnE.Kelly@nuclear.energy.gov'; 'Bill.McCaughey@nuclear.energy.gov'; 'TOM.MILLER@nuclear.energy.gov'
Cc: 'Douglas.Burns@inl.gov'; 'Phillip.Finck@inl.gov'; 'elizabeth.connell@inl.gov'; Basu, Sudhamay; Tinkler, Charles; Lee, Richard; 'corradin@cae.wisc.edu'; Flanagan, George F.
Subject: RE: Results of Nuclide Analysis

Mitch

Have you seen the various containment reports? They are bit conflicting. There also was a report last that one of the RPV pressures was sub atmosperic.

I will call you in a bit.

Jeff

-----Original Message-----
From: Farmer, Mitchell T. [mailto:farmer@anl.gov]
Sent: Friday, March 25, 2011 07:42 AM Eastern Standard Time
To: Binder, Jeffrey L.; johnE.Kelly@nuclear.energy.gov; McCaughey, Bill; Miller, Tom
Cc: Douglas E Burns; Phillip J Finck; Connell, Elizabeth; Basu, Sudhamay; 'Tinkler, Charles'; 'Lee, Richard'; 'Michael Corradini'; Flanagan, George F.
Subject: FW: Results of Nuclide Analysis

You should proabably take a look at this. They are even detecting noble metals areound the site (La, Ru) which are indicative of fuel melting. I'd ask Dana Powers to have a 5 minute look at these if you could Richard/Charlier/Sud; we need his opinion. Maybe he can tell from teh La/Ru ratio whether there has been air ingress (if not, I think the ratio shoud be about what it is in irradiated fuel in general; if so, I'd expect Ru to be high as RuOx is quite volatile).

I also don't like the 0.355 MPa pressre reading on unit 1 yesterday that we talked about. Can't be much water going in as that's about teh head teh head teh fire pump pushes. It might have stopped pressurizing because teh core uncovered and so there's no more steam productino?? Or leak rate of containment has matched steam production at pressure (don't containments usually leak 0.5 % /day at pressure).

These are some questions, none of them good.

Thanks
Mitchr

From: Grandy, Christopher
Sent: Friday, March 25, 2011 12:04 AM
To: Farmer, Mitchell T.
Cc: Taiwo, Temitope A.; Yang, Won Sik
Subject: FW: Results of Nuclide Analysis

00/17

They are also seeing Ru isotope in the seawater and in the air now.

Chris

From: Yang, Won Sik
Sent: Thursday, March 24, 2011 5:57 PM
To: Taiwo, Temitope A.
Cc: Grandy, Christopher
Subject: Results of Nuclide Analysis

Temi,

More results of nuclide analysis are in page 11 to 16. They have detected even La isotopes, indicating fuel melting according the correlation of group release and fuel temperature.

Won Sik

From: Taiwo, Temitope A.
Sent: Thursday, March 24, 2011 5:34 PM
To: Yang, Won Sik; Hoffman, Edward A.; Kim, Taek K.; Adduci, Joseph J.
Subject: FW: Check out these photos

FYI.

Temitope Taiwo
Argonne National Laboratory
9700 S. Cass Avenue,
Argonne, IL 60439-4842, U.S.A.
Taiwo@anl.gov
(630) 252 1387

From: Grandy, Christopher
Sent: Thursday, March 24, 2011 2:51 PM
To: Taiwo, Temitope A.; Sofu, Tanju; Seidensticker, Ralph W.; Farmer, Mitchell T.
Subject: Check out these photos

Lee, Richard

From: Richard L Garwin [rlg2@us.ibm.com]
Sent: Saturday, March 26, 2011 11:38 PM
To: Aoki, Steven
Cc: Larzelere, Alex; DL-NITsolutions; Binkley, Steve
Subject: RE: FW: Water Level Sensors and Deployment Concepts

It's been said, but it can't be said often enough

Could be a fire hose with a heavy weight on its end, draped iinto the pool.

Dick Garwin

From Steve Aoki:
If this hasn't been addressed already, perhaps we should push to ensure that the Japanese retain the ability to top up the spent fuel pools, even if radiation levels on the ground near the reactor buildings preclude access by human operators or continued use of a man-operated crane. At its simplest, this could would be no more than a pipe fixed to the pool, with the other end at ground level fitted with a robot-operable quick connect to which a hose could be attached. Some redundancy would be desirable, since one would not want to lose the ability to introduce water to the pool if there were to be damage to a single fill pipe .

00118

Lee, Richard

From: Kelly, John E (NE) [JohnE.Kelly@Nuclear.Energy.Gov]
Sent: Sunday, March 27, 2011 10:05 PM
To: Lee, Richard
Subject: RE: missed the firts few mintues of US - Japan disussions

Summary is that if it continues to go badly, what could they do if they had to evaluate site because of high hard levels. He is going to recommend that a group be formed in Japan to work this.

From: Lee, Richard (NRC)
Sent: Sunday, March 27, 2011 6:58 PM
To: Kelly, John E (NE)
Subject: missed the firts few mintues of US - Japan disussions

John:

I missed the first few minutes of the conference call since the call in number was changed. I got in the end of where Dr. Kondo was going through a list of proprieties that they are addressing. Please let me know what they are.

Thanks,
Richard

00/19

Lee, Richard

From: Per F. Peterson [peterson@nuc.berkeley.edu]
Sent: Sunday, March 27, 2011 5:25 PM
To: DL-NITsolutions
Subject: Fwd: reactor#3 and others

I am forwarding this email from Professor Tom Devine, a colleague in Materials Science and Engineering at UC Berkeley who has extensive experience in corrosion processes in light water reactor systems. He expresses strong concern about the likelihood of very rapid stress corrosion cracking in the reactor primary system (0.8 cm/day), given the high concentration of chloride in the reactor coolant. He believes that it is urgent to begin flushing salt water out of these systems. I think that it is worthwhile to take this concern seriously.

-Per

>Date: Fri, 25 Mar 2011 09:57:48 -0700
>Subject: reactor#3 and others
>From: devine@berkeley.edu
>To: Peterson@nuc.Berkeley.edu
>
>----------
>Per,
>
>I'm troubled by the report I just heard on CNN, which indicated that Co
>was in the ocean adjacent to the plant and in the water that burned the
>three workers. Apparently the workers were exposed to Co-containing
>water while in the turbine room. The presence of Co at these two
>locations suggests that water from the core is releasing into the ocean
>and into the turbine room.
>
>The cause of the leak(s) might be pipes that were cracked during the
>hydrogen explosions. Alternatively, the leak(s) might be due to
>corrosion and/or stress corrosion cracking. The possibility of
>corrosion and scc must be urgently addressed.
>
>The email that I sent to you one week
>ago was prompted by our parking-lot discussion in which you mentioned
>the amount of salt water that was being used to cool the reactors. My
>concern then was that the chloride would cause stress corrosion
>cracking of the stainless steel cladding that coats the inside of the
>RPV and of stainless steel piping that is part of the cooling system.
>I indicted that an upper limit SCC velocity of about 0.8 cm/day in
>stainless steel exposed to hot aqueous chloride. Hot aqueous chloride
>would severely corrode, and possibly crack, low alloy steel and carbon
>steel, especially if oxygen (from air) is also present.
>
>The only sure way of stopping SCC is to remove the stress. In this
>case removing the stress might not be possible because the highest
>stresses are most likely residual. Furthermore, the carbon steel and
>low alloy steel are susceptible to very high corrosion rates in high
>temperature aqueous chloride, so if cracks have penetrated the RPV
>cladding then corrosion of the low alloy steel is as much of a potential problem as is SCC
>

00120

>The steam lines going from the RPV to the turbine are carbon steel, so
>hot aqueous chloride can be expected to severely corrode and possibly
>crack the steam lines.
>
>At this point the best remedial action to take is to get rid of the salt.
>Probably the only way to do it is by dilution: flooding the reactor
>with salt-free water. In my view it is extRemely urgent that the
>chloride be removed asap. Can you communicate this message to someone in authority?
>
>Tom

--

Per F. Peterson
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Lee, Richard

From: Richard L Garwin [rlg2@us.ibm.com]
Sent: Sunday, March 27, 2011 11:24 PM
To: Binkley, Steve
Cc: Brinkman, Bill; 'Binder, Jeffrey L.'; Hurlbut, Brandon; Sheron, Brian; Poneman, Daniel; 'Elizabeth.Connell@inl.gov'; 'Harold McFarlane'; 'Harold Denton'; Adams, Ian; 'John Holdren'; 'JOE H. PAYER'; Kelly, John E (NE); 'John Grossenbacher'; Owens, Missy; 'Per Peterson'; Lyons, Peter; 'Phil Finck'; 'Dick Garwin'; Lee, Richard; 'Bob Budnitz'; 'Rolando Szilard'; SCHU; Aoki, Steven; Koonin, Steven; 'Steve Fetter'; Binkley, Steve; DAgostino, Thomas
Subject: Conversion factor from becquerel to siervert/hr

I think it was Harold Denton on the call today who asked the conversion between Bq and Sv/hr in water.

Steve Koonin immediately figured that 10^6 Bq of 1 MeV per disintegration corresponded to about 500 milliSv/hr, and I concur:

[10^6 Bq/cc] [1 MeV/Bq-s] [1.6 x 10^-13 J/MeV] [3600 s/hr] [1000 cc/kg] = 0.5 J/kg-hr or 500 mSv/hr, given the definition of 1 Sv = 1 J/kg.

It is easy to see that a deep pool of contaminated water bearing 10^6 Bq/cc overlain by uncontaminated air of water would provide half this dose rate-- 250 mSv/hr-- immediately above the surface of the contaminated water. To prove this, understand that dose rate is linear in the source term of Bq/s, multiplied by some number depending on geometry. Given the linearity, the dose at the boundary between contaminated water below $x<0$ and clean water above, $x>0$, is the same as if the contaminated water and pure water were reversed. Hence it would be precisely doubled if both bodies of water were contaminated.

So the dose rate just above a deep pool of water with 10^6 Bq/s of 1-MeV gamma emitter is 250 mSv/hr or 25 R/hr. The dose rate is linear with the product of source strength in Bq and disintegration energy in MeV.

Dick Garwin

00121

Lee, Richard

From: Per F. Peterson [peterson@nuc.berkeley.edu]
Sent: Sunday, March 27, 2011 1:29 PM
To: Kelly, John E (NE)
Cc: Adams, Ian; Aoki, Steven; Binkley, Steve; Brinkman, Bill; Budnitz, Bob; DAgostino, Thomas; Finck, Phillip; Garwin, Dick (EOP); Garwin, Dick (IBM); Grossenbacher, John (INL); Hurlbut, Brandon; John Holdren; Kelly, John E (NE); Koonin, Steven; Lee, Richard; Lyons, Peter; McFarlane, Harold; Miller, Neile; Mustin, Tracy; NITSolutions; Owens, Missy; Peterson, Per; Poneman, Daniel; SCHU; Sheron, Brian; Steve Fetter; Szilard, Ronaldo; Binkley, Stev
Subject: Re: Fw: Status & Recommendations 3-26-11 21 hrs.docx

John,

This NRC status document is helpful to review.

One thing that it does not include is an assessment of what the water levels are in the drywell volume and in the torus of the 3 units. This would be helpful information to have, particularly what the water inventory is in the dry well and where the water elevation is relative to the top of fuel elevation. This would be particularly useful to know on Units 2 and 3, where the recirculation pump seals are suspected to have failed. In this case, if the reactor is depressurized relative to the containment and the dry well is filled above the the top of active fuel, one can also do the same inside the reactor vessel.

Also, it would be helpful to have some status report on the floor drain and sump/drain-tank systems in Units 1-4, particularly any level indication from the sumps and status on the operability of the drain system pumps. Most of the water that they have been injecting that has missed the spent fuel pools, plus the water that sloshed out of the pools, has probably accumulated there and any that might be leaking from the pools. It will be important to know whether it has been enough has accumulated to flood the pump motors and other equipment. If so, this will complicate the clean up effort, since the capability to pump water out of the drain system sumps for treatment will be an important element in the overall effort to flush out the reactor systems and get radiation and contamination levels reduced in the reactor building.

I will be on an airplane this afternoon, so will not be able to join for the telecon today.

-Per

Spke with nrc japan this morning and recommendation contiues to evolve
John E Kelly

From: Taylor, Robert <Robert.Taylor@nrc.gov>
To: Kelly, John E (NE)
Cc: Cherry, Ron
Sent: Sun Mar 27 08:26:37 2011
Subject: FW: Status & Recommendations 3-26-11 21 hrs.docx

Dr. Kelly

00122

I received a â¤¦mail undeliverableâ¤ù response for the email to Mr. Holdren. Perhaps it is a challenge related to our network here in Japan. Could I kindly ask you to forward the email along to Mr. Holdren to ensure he received it.

Best regards,

Rob

From: Taylor, Robert
Sent: Sunday, March 27, 2011 8:22 AM
To: 'John_P_Holdren@ostp.eop.gov'
Cc: 'JohnE.Kelly@nuclear.energy.gov'; 'cherryRC@state.gov'; Scott, Michael; Casto, Chuck
Subject: FW: Status & Recommendations 3-26-11 21 hrs.docx

Mr. Holdren,

We were asked to email you the NRCâ¤□s latest assessment document regarding the status of the Fukushima Daiichi nuclear power plants. We plan to provide the attached to our Japanese counterparts at NISA tomorrow so that we can continue our dialogue on the ongoing efforts to bring this issue to a safe closure. This continues to be a living document that we will periodically update as new information becomes available.

Please let us know if you have any questions.

Best regards,

Rob Taylor

NRC Japan Team

From: RST01 Hoc
Sent: Saturday, March 26, 2011 10:56 PM
To: Blamey, Alan; Casto, Chuck; Dorman, Dan; GE Hitachi; Giessner, John; INPO ERC Main; INPO ERC Tech; John Kelly - DOE; Monninger, John; Richard Stark - DOE; Rob Versluis - DOE; RST01B Hoc; RST03 Hoc; Sal Golub - DOE; Scott, Michael; Taylor, Robert
Cc: Ruland, William; RST07 Hoc; RST08 Hoc; RST09 Hoc; RST03 Hoc; LIA07 Hoc; Orr, Mark
Subject: Status & Recommendations 3-26-11 21 hrs.docx

Team:

Attached is the RSTâ¤□s Consensus Assessment of Fukushima Daiichi Units with input from the team.

Mark Orr

RST Coordinator

Content-Type: application/vnd.openxmlformats-officedocument.wordprocessingml.document;
name="Status & Recommendations 3-26-11 21 hrs.docx"
Content-Description: Status & Recommendations 3-26-11 21 hrs.docx
Content-Disposition: attachment;
filename="Status & Recommendations 3-26-11 21 hrs.docx"; size=39221;
creation-date="Sat, 26 Mar 2011 22:51:52 GMT";
modification-date="Sat, 26 Mar 2011 22:51:52 GMT"

Attachment converted: Macintosh HD:Status & Recommend#25F66D4.docx (/)
(025F66D4)

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Lee, Richard

From: Binder, Jeffrey L. [binderjl@ornl.gov]
Sent: Sunday, March 27, 2011 5:06 PM
To: Kelly, John E (NE); Connell, Elizabeth; Lee, Richard
Cc: Peltz, James; tom.miller@hq.doe.gov
Subject: RE: Diagram of Fukushima No. 1

Richard:
I am going to send you a bunch of drawings that Tom Miller was able to obtain. There will be a series of emails.
Jeff

From: Kelly, John E (NE) [JohnE.Kelly@Nuclear.Energy.Gov]
Sent: Sunday, March 27, 2011 4:54 PM
To: Connell, Elizabeth; Binder, Jeffrey L.
Cc: Peltz, James
Subject: FW: Diagram of Fukushima No. 1

if we have this can one of you take care of it

From: Lee, Richard (NRC)
Sent: Friday, March 25, 2011 5:33 PM
To: Kelly, John E (NE)
Subject: Diagram of Fukushima No. 1

John:

Ie would like to get details electronics drawings of plant and the spent fuel pools of all 4 nits.

ıanks, Richard

00123

Lee, Richard

From: Powers, Dana A [dapower@sandia.gov]
Sent: Monday, March 28, 2011 12:02 PM
To: Lee, Richard
Subject: RE: Input to DOE science council

Richard, This looks fine. Dana

From: Lee, Richard [mailto:Richard.Lee@nrc.gov]
Sent: Monday, March 28, 2011 7:04 AM
To: Powers, Dana A; Dana Powers
Subject: Input to DOE science council

Dear Dana:

Please review and revise the attached note so I can offer our input to the ongoing discussions.

Thanks, Richard

00124

Lee, Richard

From: Powers, Dana A [dapower@sandia.gov]
Sent: Monday, March 28, 2011 3:03 PM
To: Lee, Richard
Subject: RE: Salt Beds and Core Melts

Cry!?! No, I will need immediate therapy by overpriced psychologists of international reputation. Dana

-----Original Message-----
From: Lee, Richard [mailto:Richard.Lee@nrc.gov]
Sent: Monday, March 28, 2011 1:00 PM
To: Powers, Dana A
Subject: RE: Salt Beds and Core Melts

No call today. I know you going to cry!

-----Original Message-----
From: Powers, Dana A [mailto:dapower@sandia.gov]
Sent: Monday, March 28, 2011 2:52 PM
To: Lee, Richard
Subject: RE: Salt Beds and Core Melts

I'm sure pictures have suffered the fate of the Dodo bird, since the test was done back in the CRBR days. I think we gave a conference paper on the test. I will look, but it may take a while.

Did I understand correctly that we are NOT doing the call-in today?
Dana

-----Original Message-----
From: Lee, Richard [mailto:Richard.Lee@nrc.gov]
Sent: Monday, March 28, 2011 10:56 AM
To: Powers, Dana A
Subject: RE: Salt Beds and Core Melts

Dana:

Do you have summary documents and pictures from these test? If you can compose a page or 2, I can send it to the DOE Science Council. Thx.

Richard

-----Original Message-----
From: Powers, Dana A [mailto:dapower@sandia.gov]
Sent: Monday, March 28, 2011 12:34 PM
To: Lee, Richard
Subject: Salt Beds and Core Melts

Richard, A usual core melt scenario for a BWR has individual assemblies slumping into the lower head where they quench because of all the water and steel structures (CRDs and instrument tubes) in the lower head. Then, if coolability cannot be maintained there is remelting beginning with the metals (stainless steel alloyed with residual cladding and

channel box alloys). It is this metal melt that attacks the lower head usually at an instrumentation tube.

The presence of a large salt bed might alter this. Salt melts at a relatively low temperature with a middle of the road heat of fusion. The molten salt would act as a working fluid to dissipate the decay heat. We did a fairly large scale test of this type with borax (sodium borate) rather than salt. We got beautiful Bernard convection cells - lovely pentagrams. The steel within the molten borax had a solid crust and was molten within this crust. Wonder if the same behavior would be realized in salt? Dana

Lee, Richard

From: Binkley, Steve [Steve.Binkley@science.doe.gov]
Sent: Tuesday, March 29, 2011 5:50 PM
To: Lee, Richard
Subject: Re: When do we have a conference call with Japan

6:00 pm, same number as for the call we just completed.

Steve Binkley

From: Lee, Richard <Richard.Lee@nrc.gov>
To: Adams, Ian; Binkley, Steve
Sent: Tue Mar 29 17:31:11 2011
Subject: When do we have a conference call with Japan

Hi, Steve and Ian:

Please let me know when the call to Japan will take place.

Richard

00126

Lee, Richard

From: Bisconti, Giulia [Giulia.Bisconti@nuclear.energy.gov]
Sent: Tuesday, March 29, 2011 12:26 PM
To: PWG; DL-NITsolutions
Cc: Bisconti, Giulia
Subject: Bisconti TDY-Tokyo

Dear all:

As requested, this is an update of how I am helping in Tokyo for the week. My main duty is to be embedded with the NRC team at the Embassy. I am also performing other duties where I can be helpful to Ron and Aleshia. They have both been very welcoming.

Giulia

Here are some items of interest:

--Two PNNL experts to visit Japan (at the request of Japan)to help on water decontamination and storage issues.
--Japanese government is seeking private sector experts on fuel rod/pool issues with hands-on TMI experience (per NRC meetings).
--Japanese government is thrilled with NNSA's airborne monitoring cooperation (I joined MOFA/MEXT meeting with Alan).
--Met with Toshiba and B&W. Toshiba has hundreds of employees at the accident site and the TEPCO emergency control room. Toshiba is deploying equipment and resources. Toshiba and Hitachi are both in the emergency control room, and TEPCO is heavily relying on them. Toshiba offered to be an information resource to our specialists.
--6.3 quake in Northeastern Japan today--no damage reported to facilities.
--Aleshia and I met today with METI Vice Minister Okada (at his invitation). Okada mentioned that Japan is thinking about a "cover" for the Fukushima plants in the coming months. He and his colleagues expressed very deep appreciation for assistance from DOE and its National labs and everyone's hard work and long hours. They appreciated DOE recommendations on the salt/fresh water issue. Okada offered to personally work with DOE on any matter related to the Fukushima response and to help overcome any barrier. Although, he mentioned that information flow is much better now (the government is better organized to receive and respond to inquiries) and the mechanisms seem to be working. He noted that Japan will be looking for assistance-including on the issue of water decontamination (10,000 tons ? of contaminated water). He said that the Japanese government would seek input from DOE and its labs, including PNNL, Idaho, Livermore, others...
--Participated in NRC meeting. Issues: remove heat from the reactor. Structural concerns for the pools. Controlling releases. Water management is a big issue. Are the Japanese workers wearing adequate protective clothing? Flooding--continued leakages? Need to establish the water level of the pools--want to get water above the rods, maybe 3-4 feet above.

00127

Lee, Richard

From: Powers, Dana A [dapower@sandia.gov]
Sent: Tuesday, March 29, 2011 4:12 PM
To: Lee, Richard
Subject: RE: Re: Salt water corrosion

I think the accelerated corrosion is actually fatique corrosion cracking in pressure vessel steel. I have stuff on that. I'll put something together. Dana

-----Original Message-----
From: Lee, Richard [mailto:Richard.Lee@nrc.gov]
Sent: Monday, March 28, 2011 7:22 PM
To: danapowers@msn.org; Powers, Dana A
Subject: FW: Re: Salt water corrosion

Dana:

At this rate, the vessel should disappear soon!!

Richard

From: Per F. Peterson [peterson@nuc.berkeley.edu]
Sent: Monday, March 28, 2011 7:59 PM
To: SCHU; Garwin, Dick (IBM); Binkley, Steve; Brinkman, Bill; Binder, Jeff; Hurlbut, Brandon; Sheron, Brian; Poneman, Daniel; Connell, Elizabeth; McFarlane, Harold; 'Harold Denton'; Adams, Ian; John Holdren; 'JOE H. PAYER'; Kelly, John E (NE); Grossenbacher, John (INL); Owens, Missy; Peterson, Per; Lyons, Peter; Finck, Phillip; Garwin, Dick (EOP); Lee, Richard; Budnitz, Bob; Szilard, Ronaldo; Aoki, Steven; Koonin, Steven; Steve Fetter; Binkley, Steve; DAgostino, Thomas
Subject: Fwd: Re: Salt water corrosion

As a follow-up to Steve Chu's note on oxygen, Tom Devine at UC Berkeley also identifies oxygen as a key problem for accelerated corrosion. Corrosion will emerge as the most important longer-term risk factor, since integrity of the reactor containment is essential to maintaining flooding of the drywell and heat removal from the reactor vessel.

Now that the transition to fresh water has been achieved (something we identified early on as an important risk-reduction step), one of the most important risk-reduction opportunities now available is to expedite the installation of equipment to strip oxygen from water injected into the reactor vessels and primary containments.

I note an important in Tom's email, that CO2 can have a beneficial and protective effect for corrosion. This suggests that it would be better to use CO2 as an inerting agent to inject into the reactor feedwater, rather than nitrogen. CO2 can serve as an inerting agent, but also should have useful effects in inhibiting corrosion.

-Per

>Date: Mon, 28 Mar 2011 08:07:02 -0700
>Subject: Re: Salt water corrosion
>From: devine@berkeley.edu
>To: "Per F. Peterson" <peterson@nuc.berkeley.edu>
>
>Per,

00128

>
>The critical parameters that will determine the appropriate material
>for storing aqueous waste are: the water's: dissolved oxygen concentration,
>temperature, pH and chloride ion concentration. For example, carbon
>steel can be protected against corrosion in 65°C water that contains
>0.4M chloride (i.e., greater than sea water) and is saturated with CO2
>(so it is acidic). However, in this case it is critically important
>that the water is free of oxygen (as it is saturated with CO2) and
>organic corrosion inhibitors have been added. This example illustrates
>the interplay between the environment's chemistry (in this case, the
>absence of dissolved oxygen permits the use of carbon steel along with
>corrosion inhibitors, despite the presence of high chloride
>concentration and a pH of about 5), the appropriate material and the
>required protective measure (organic inhibitors).
>
>As you have pointed out, titanium would be a good choice. In over 70
>years of use of titanium in a range of environments and applications
>there has never been a single reported instance of crevice corrosion or
>pitting corrosion of titanium at temperatures below 80°C.
>
>The family of Ni-Cr alloys should be considered. Recall that Alloy C22
>(a Ni-Cr-Mo alloy) was selected by DOE for long-term storage of waste
>at Yucca Mtn. Cobalt-chromium alloys would also be effective, but
>should be ruled out on the basis of cost/availability.
>
>Depending on the storage time, there are stainless steels that should
>also be considered.
>
>In summary, there are a number of options available. IN particular,
>there are a number of corrosion-resistant alloys that should be
>considered as well as carbon steel that is protected by organic
>inhibitors, or cathodic protection. I can suggest specific alloys
>and/or methods of protecting carbon steel once I know the chemistry of the water.
>
>Tom
>
>> Tom,
>>
>> One of the major elements of remediating the reactors at Fukushima
>> will involve flushing high activity, corrosive fluids out of the
>> reactor vessel into some type of storage tank(s), with an evaporator
>> system to concentrate the fluids and to recycle the condensate for
>> further flushing.
>>
>> Currently the turbine condensers are being used for storage of
>> low-activity fluids (see below).
>>
>> The storage tanks for high activity fluids will need to have life
>> spans of at least a few years. This raises the question of what
>> materials can be used an how corrosion can be managed. Commonly for
>> salt water service one would use coatings, but I expect that these
>> would not survive well in a high radiation environment. Titanium has
>> excellent salt-water resistance, but I'm not sure if it is a
>> practical material to use to use for tanks that need to be acquired
>> on an expedited basis. An alternative is to attempt to use existing
>> tanks at the site; these might be carbon steel or stainless steel.
>>

>> Could you provide advice on potential materials to use for such
>> storage tanks? If carbon steel or stainless steel end up being the
>> only practical options, are there methods that might be used to
>> mitigate corrosion (e.g., cathodic protection), and what would the
>> thickness need to be to assure several years of service life?
>>
>> -Per
>>
>>
>>
>>>Date: Sun, 27 Mar 2011 20:50:09 -0700
>>>To: "Gehin, Jess C." <gehinjc@ornl.gov>
>>>From: "Per F. Peterson" <peterson@nuc.berkeley.edu>
>>>Subject: Re: Hydrogen
>>>
>>>Jess,
>>>
>>>Below is my assessment of the major actions needed to remediate these
>>>reactors. Getting the refueling crane out of the way in Unit
>>>3 will be important (I'm pretty sure that the 100 ton reactor crane
>>>would not have been parked above the pool). Hopefully the pool is
>>>not leaking and can be refilled enough to reduce radiation levels to
>>>assist with the effort to remove this debris, including cutting up
>>>the crane system so it can be lifted out.
>>>
>>>The turbine condensers are an ok place to flush low-activity fluids
>>>to. In the longer term evaporators will be needed to concentration
>>>these solutions and produce condensate that can be used in additional
>>>flushing operations. Shielded tanks (probably new) will be needed
>>>for high activity fluids flushed out of the reactors and
>>>containments.
>>>
>>>-Per
>>>
>>>
>>>
>>>Initial remediation efforts will involve flushing corrosive and
>>>radiaoctive fluids out of the primary systems and containments into
>>>storage tanks (potentially new tanks, similar to the waste tanks at
>>>Hanford and SRS), and the use of evaporators to concentrate this salt
>>>solution in the tanks and return condensate for additional
>>>flushing. Later on the salts can be processed into suitable waste
>>>forms using similar technologies to those used for the supernate
>>>fluids in U.S. defense waste tanks.
>>>
>>>In parallel remotely operated construction cranes will be used to
>>>remove debris from above and around spent fuel pools, and then the
>>>cranes will be used to transfer the fuel assemblies into lightly
>>>shielded baskets to be lowered into shielded casks on the ground.
>>>
>>>Once the spent fuel is moved and the systems have been extensively
>>>flushed to reduce radiation levels and to remove mobile
>>>contamination, core debris will be removed as was done at TMI.
>>>
>>>Currently this is my best projection of the sequence of actions
>>>needed to remediate these reactors.
>>>

>>>-
>>>
>>>>Per,
>>>>
>>>>Interesting theory. I happen to be watching NHK right now and there
>>>>seems to be a lot of discussion by a government official (chief
>>>>cabinet secretary Edano) about when the venting was ordered and when
>>>>it occurred. I was looking at the timelines on the BBC and
> >>>OECD sites and they indicate that there was venting before the
>>>>explosion in unit 1 and unit 3.
>>>>
>>>>He is also saying that the leak of the water to the TG building was
>>>>a "one-time" event. They are working to pump the water into the
>>>>condensers, but for unit 1 and 2 the condensers are full and they
>>>>need to dump them first.
>>>>
>>>>They have a university of tokyo professor that says the crane in
>>>>unit 3 has fallen into the pool and may have damaged some fuel.
>>>>They have some close-up video and images from what looks to be a
>>>>helicopter with a good zoom lens.
>>>>
>>>> -- Jess
>>>>
>>
>> --
>>
>> ---
>> Per F. Peterson
>> Professor and Chair
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>> University of California
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>> ---
>>

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Lee, Richard

From: Per F. Peterson [peterson@nuc.berkeley.edu]
Sent: Tuesday, March 29, 2011 6:16 AM
To: SCHU; Garwin, Dick (IBM); Binkley, Steve; Brinkman, Bill; Binder, Jeff; Hurlbut, Brandon; Sheron, Brian; Poneman, Daniel; Connell, Elizabeth; McFarlane, Harold; 'Harold Denton'; Adams, Ian; John Holdren; 'JOE H. PAYER'; Kelly, John E (NE); Grossenbacher, John (INL); Owens, Missy; Peterson, Per; Lyons, Peter; Finck, Phillip; Garwin, Dick (EOP); Lee, Richard; Budnitz, Bob; Szilard, Ronaldo; Aoki, Steven; Koonin, Steven; Steve Fetter; Binkley, Steve; DAgostino, Thomas
Subject: Millstone Unit 1 trip report

Here is a brief summary of observations from our visit to Millstone Unit 1 (a shutdown 660 MWe GE reactor with an isolation condenser, that entered service in 1971).

The plant staff reviewed two important areas of experience that are relevant to the Fukushima accident.

First, they noted experience that at low pressure differences, check valves in the control rod drive mechanisms can leak (they seal well at 1000psi, but not at 50 psi). Millstone had implemented an early plant modification that would provide a leak path for primary coolant through the check valves to the main feedwater line, and thus back to the feedwater pumps in the turbine building where leakage could occur through the feedwater pump check valves and seals into a seal water drain tank that could then overflow. They note that this could be the source of contamination found in the Fukushima turbine buildings.
Closing the feedwater pump isolation valves would eliminate this leakage path.

Second, early in its operating history Unit 1 experienced a substantial intrusion of salt water due to failure of condenser tubes. Substantial and rapid stress corrosion cracking of stainless steel occurred. In particular, chloride got into the condensate storage tank and then into the shell-side water of the isolation condenser. Rapid stress corrosion cracking resulted in leaking tubes within a few weeks (I think this was the time frame). John Kelly has a report on the saltwater intrusion event provided by the plant staff that provides more details. The important point is that there are now much higher chloride levels in the Fukushima units, raising the risk for rapid stress corrosion cracking. I think that lower reactor vessel penetrations, and stainless-steel lines penetrating the primary containment, are at particular risk. Leakage in the lower reactor vessel head or the primary containment could jeopardize the capacity to continue heat removal from the reactor cores, which argues for an expedited effort to treat water for injection (oxygen removal, corrosion inhibitor addition) and to flush salt water from the reactors and primary containments.

Plant staff reviewed potential flow paths that might be used to establish once-through cooling, where fresh water injection would provide evaporative cooling and flushing, with periodic bleeding of steam for heat removal combined with let-down of excess flow to flush contaminated salt water from the system.

They recommended a single flow path, with feed injection to the reactor vessel at sufficient rates to overflow water to the steam lines and through the safety relief valves to the suppression pool.
Let-down from the suppression pool could then occur from the low-pressure safety injection system, which takes suction from the bottom of the torus.

This provides one option for flushing salt from the reactor and primary containment systems. A systematic review of potential once-through cooling and flushing options is warranted. If a flow pattern can be established where fresh water is fed from the top, and salt water

03/29

flushed from below, stable stratification of denser salt water below lighter fresh water could increase the effectiveness and speed of flushing.

Also discussed were options for level measurement and for use of radiation hardened cameras.

Finally, in touring the reactor and turbine buildings of Unit 1, it was impressive to think of the Japanese operators who were doing their work in those buildings by flashlight.

-Per

--

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Lee, Richard

From: Per F. Peterson [peterson@nuc.berkeley.edu]
Sent: Tuesday, March 29, 2011 7:42 AM
To: SCHU; Garwin, Dick (IBM); Binkley, Steve; Brinkman, Bill; Binder, Jeff; Hurlbut, Brandon; Sheron, Brian; Poneman, Daniel; Connell, Elizabeth; McFarlane, Harold; 'Harold Denton'; Adams, Ian; John Holdren; 'JOE H. PAYER'; Kelly, John E (NE); Grossenbacher, John (INL); Owens, Missy; Peterson, Per; Lyons, Peter; Finck, Phillip; Garwin, Dick (EOP); Lee, Richard; Budnitz, Bob; Szilard, Ronaldo; Aoki, Steven; Koonin, Steven; Steve Fetter; Binkley, Steve; DAgostino, Thomas
Subject: Fwd: CO2 not good

Importance: High

I need to reverse my earlier comment about CO2 being beneficial for suppressing corrosion. As long as one is removing oxygen from the feedwater, CO2 is not good.

Nitrogen (or argon) is likely to be the better inerting agent to inject into feedwater.

-Per

>Date: Mon, 28 Mar 2011 20:43:25 -0700
>Subject: CO2 not good
>From: devine@berkeley.edu
>To: "Per F. Peterson" <peterson@nuc.berkeley.edu>
>
>Per,
>
>CO2 should NOT be added. CO2 removes oxygen (which is good) but
>creates carbonic acid (which is not good).
>
>Tom
>

--

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00130

Lee, Richard

From: Per F. Peterson [peterson@nuc.berkeley.edu]
Sent: Wednesday, March 30, 2011 2:40 PM
To: Kelly, John E (NE); Richard L Garwin
Cc: Caponiti, Alice; DL-NITsolutions; Golub, Sal
Subject: Fwd: Re: FW: DRAFT Proposal for Rev. 1 of RST Assessment Document

A couple of additional points.

If nitrogen injection can be done into the pump discharge flow to the RPV cooling water injection, this should be considered to increase the inerting of the primary containment rather than attempting to use the normal nitrogen injection system, where the valves that need to be operated are apparently in a high radiation area that cannot be accessed.

On the suggestion to reestablish residual heat removal, I am concerned that the RHR pumps may not operate reliably with high salinity, high activity fluids. Replacement of failed pump seals will be very challenging due to the high activity in the fluids. Thus if residual heat removal is restarted, I recommend that it be done with only one of the two trains of RHR equipment, and the second train of equipment be reserved for use after substantial flushing has been performed to reduce salinity and activity levels. All of this may be academic anyhow, since the RHR pumps are at a low elevation in the reactor building and may be flooded due to the large amounts of water that have gone into the reactor building from spent fuel sloshing and refill activities.

Also, in the status reports the level of water in the drywell is not listed. This is an important parameter to report, as the likely/known/unknown level of water in the reactor vessel is reported.

-Per

To: "Kelly, John E (NE)" <JohnE.Kelly@Nuclear.Energy.Gov>
CC: "Caponiti, Alice" <Alice.Caponiti@Nuclear.Energy.Gov>, DL-NITsolutions <DL-NITsolutions@nnsa.doe.gov>, "Golub, Sal" <sal.golub@Nuclear.Energy.Gov>
Subject: Re: FW: DRAFT Proposal for Rev. 1 of RST Assessment Document
X-KeepSent: 0A915D84:CCD7AFF7-85257863:0060ADE3;
type=4; name=$KeepSent
From: Richard L Garwin <rlg2@us.ibm.com>
Date: Wed, 30 Mar 2011 13:42:17 -0400

Thanks, John.

In this Proposal for Rev. 1 of RST I think we find,

1. Maintain RPV injection / core cooling
2. Flood drywell to a level above minimum debris submergence level (approximately 4ft)
3. Continue to increase drywell level to TAF in a controlled manner, giving considerations to venting / containment atmosphere conditions and the integrity of primary containment.
4. Establish long term recirculation cooling.

00131

It seems to me that the proposed wording,

➢ Begin flooding primary containment using drywell spray path. Do not lower RPV injection below MDRIR.

does not begin to capture these priorities, whether or not it is the right thing to do.

Flooding the dry well to 4-ft level is a very different matter compared with flooding to top of active fuel (TAF).

Specifically, I propose that we should, if we agree, include the 4-point recommendation and reassess after Point 2.

Dick Garwin

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Lee, Richard

From: Powers, Dana A [dapower@sandia.gov]
Sent: Wednesday, March 30, 2011 4:57 PM
To: Lee, Richard
Subject: RE: Processes to remove radiocesium from salt solutions

sure

-----Original Message-----
From: Lee, Richard [mailto:Richard.Lee@nrc.gov]
Sent: Wednesday, March 30, 2011 2:42 PM
To: Powers, Dana A
Subject: RE: Processes to remove radiocesium from salt solutions

Dana:

Do you want me to send your message to the DOE Science CXoincil?

Richard

-----Original Message-----
From: Powers, Dana A [mailto:dapower@sandia.gov]
Sent: Wednesday, March 30, 2011 4:30 PM
To: Lee, Richard
Subject: RE: Processes to remove radiocesium from salt solutions

Hanford and Savannah River folks have a lot more experience with ion exchange than I do. I think it is tricky to do in this situation because you have micromolar quantities of cesium and half molar sodium ions. You really have to have a lot of selectivity to overcome the concentration difference. But the Hanford and savannah river wastes had equally high concentrations of sodium nitrate. You also have a substantial amount of divalent cations in the salt water that love to grab onto resins. Crystalline silicontitanates seem to perform well. You do have to figure out what to do with it all. Nenoff was the developer of this method. She is still here at Sandia. I will send her a note and see what she thinks. Dana

-----Original Message-----
From: Lee, Richard [mailto:Richard.Lee@nrc.gov]
Sent: Wednesday, March 30, 2011 11:40 AM
To: Dana Powers (Home); Powers, Dana A
Subject: FW: Processes to remove radiocesium from salt solutions

Dana:

Ion-exchange to remove Cs, Sr and actinides....

Richard

-----Original Message-----
From: Per F. Peterson [mailto:peterson@nuc.berkeley.edu]
Sent: Wednesday, March 30, 2011 1:20 PM
To: Brinkman, Bill; 'Binder, Jeffrey L.'; Hurlbut, Brandon; Sheron, Brian; Poneman, Daniel; 'Elizabeth.Connell@inl.gov'; 'Harold McFarlane'; 'Harold Denton'; Adams, Ian; 'John Holdren'; 'JOE H. PAYER'; Kelly, John E (NE); 'John Grossenbacher'; Owens, Missy; 'Per Peterson'; Lyons, Peter; 'Phil Finck'; 'Dick Garwin'; Lee, Richard; 'Bob Budnitz'; 'Rolando Szilard'; SCHU; Aoki, Steven; Koonin, Steven; 'Steve Fetter'; Binkley, Steve; DAgostino, Thomas

00132

Subject: Processes to remove radiocesium from salt solutions

I have attached below references to four National Academy reports on methods to remove radiocesium and other water-soluble radionuclides from sodium salt solutions.

I would point in particular to the ion exchange options for removal of strontium and actinides, and removal of cesium. Cesium was separated from tank wastes at Hanford and is now in storage at Hanford, and has also been studied extensively for use at SRS, INL, and Hanford, for example:

"Alternatives for High-Level Waste Salt Processing at the Savannah River Site," Chapters on "Strontium and Actinide Removal (35-42)"
and "Crystalline Silicotitanate Ion Exchange (pp. 55-64)."

In discussion with Terry Todd at INL, one of the most promising ion exchange materials may be crystalline silicotitanate (CST), developed by Sandia National Laboratory and Texas A&M University. It has high selectivity for cesium in solutions that have high concentrations of sodium. It also can recover strontium, although the use of monosodium titanate (MST) may be more effective in stripping strontium and actinides. A mixed bed of MST and CST might be best.
CST and MST are also appropriate materials to feed into waste glass, so this would provide a path to get cesium and strontium from Fukushima incorporated into waste glass that could be produced at Rokkashomura.

I think that the goal for using ion exchange should be to achieve 90 to 99% reductions in radioactivity levels in the salt solutions, to reduce the shielding and reliability requirements for tanks and evaporator equipment. If ion exchange columns are set up in series, even higher decontamination is possible. Radiation monitoring on the outlet of the lead column can be used to determine when it is saturated, and it can then be valved out and the next column in the series can become the lead column. Further clean up and processing of these lower activity salt solutions coming out of the ion exchange can be deferred while attention is focused on remediation of the reactors.

I do remain concerned about the longer-term integrity of the primary containment vessels, since they are only 3-cm of carbon steel and are subjected to a very corrosive environment. A substantial leak in the torus of a reactor containment would make it impossible to maintain flooding of the bottom of the reactor vessels and would challenge the coolability of the reactor cores, inviting further damage and relocation of core material. It is important to note that the addition of caustic to reduce the pH of the salt solutions has been suggested to reduce corrosion rates, and that reducing the pH will also improve the effectiveness of the ion exchange.

Flushing the reactor vessels and primary containments can reduce corrosion rates, and it also reduces the source term for fluid that could be discharged into the reactor building if the primary containment leaks. Flushing also prepares the reactors to reestablish residual heat removal and for removal of the damaged fuel in the cores.

In parallel with flushing, cranes need to be used to remove debris from above and around the spent fuel pools and reactor refueling cavities. Flooding of the reactor cavities, once debris is gone, can allow the reactor vessel heads to be removed and core debris transferred into baskets and then into shielded casks. The heads are heavy, so the crane capability to make these transfers is an issue.
There is discussion of reenclosing the reactor buildings, but some hatches are needed above the reactor cavity to allow an external crane to be used.

To estimate the amount of CST and MST needed to clean up the reactors, it's probably reasonable to assume that 30 to 90 percent of the cesium and strontium that was in the

reactor cores is now in the salt solutions. The total inventory of cesium and strontium can be estimated based upon the integrated power production that has occurred with the fuel in the reactors. Fairly quickly, it will be important to get some estimates of how much ion exchange material would be needed to clean up the reactors, and to begin addressing the engineering and logistical challenges of setting up an ion exchange system at Fukushima. It may also be necessary to increase the rate of commercial production of these ion exchange materials. This is something that merits a near-term scoping study by INL and SRL.

-Per

Alternatives for High-Level Waste Salt Processing at the Savannah River Site, National Academy Press, 2000.
http://www.nap.edu/catalog.php?record_id=9959

Also

Research and Development on a Salt Processing Alternative for High-Level Waste at the Savannah River Site, National Academy Press, 2001.
http://www.nap.edu/catalog.php?record_id=10170

Evaluation of Criteria for Selecting a Salt Processing Alternative for High-Level Waste at the Savannah River Site: Interim Report, National Academy Press, 2001.
http://www.nap.edu/catalog.php?record_id=10088

Interim Report -- Committee on Cesium Processing Alternatives for High-Level Waste at the Savannah River Site, National Academy Press, 1999.
http://www.nap.edu/catalog.php?record_id=9703
--

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Lee, Richard

From: Powers, Dana A [dapower@sandia.gov]
Sent: Wednesday, March 30, 2011 6:37 PM
To: Lee, Richard
Subject: RE: white smoke

. don't know. It is very likely steam. Some of the cable fires will produce white smoke. So if it is really particulate it may be cables or other organic but not lub oil. My suspicion is that it is steam. Dana

---

From: Lee, Richard [mailto:Richard.Lee@nrc.gov]
Sent: Wednesday, March 30, 2011 4:16 PM
To: Dana Powers (Home); Powers, Dana A
Subject: white smoke

Dana:

From the daily DOE situation report:

Do you have any idea what these white smoke continues to see coming from Unit 1, 2 & 3, and Unit 4 spent fuel pool? Is it just steam? The weather is still cold there.

Richard

00133

Lee, Richard

From: Powers, Dana A [dapower@sandia.gov]
Sent: Wednesday, March 30, 2011 4:56 PM
To: Lee, Richard
Subject: RE: Mar 30 S-1 Science Council

Do you know how to open the file entitled DOE Perspective on Corrosion?

From: Lee, Richard [mailto:Richard.Lee@nrc.gov]
Sent: Wednesday, March 30, 2011 2:45 PM
To: Dana Powers (Home); Powers, Dana A
Subject: FW: Mar 30 S-1 Science Council

fyi

From: Binkley, Steve [mailto:Steve.Binkley@science.doe.gov]
Sent: Wednesday, March 30, 2011 4:45 PM
To: Adams, Ian; Aoki, Steven; Binkley, Steve; Bob Budnitz; Sheron, Brian; Brinkman, Bill; DAgostino, Thomas; Dick Garwin; Dick Garwin; Harold Denton; Harold McFarlane; Hurlbut, Brandon; JOE H. PAYER; John Grossenbacher; John Holdren; Kelly, John E (NE); Koonin, Steven; Lyons, Peter; Owens, Missy; Per Peterson; Phil Finck; Poneman, Daniel; Lee, Richard; Rolando Szilard; SCHU; Steve Fetter
Subject: FW: Mar 30 S-1 Science Council

Science Call briefing materials attached

From: Larzelere, Alex
Sent: Wednesday, March 30, 2011 4:42 PM
To: DL-NITsolutions
Subject: Mar 30 S-1 Science Council

Everybody,

Attached are the slides and document for discussion at today's Science Council meeting.

Regards,

Alex

00134

Lee, Richard

From: Richard L Garwin [rlg2@us.ibm.com]
Sent: Wednesday, March 30, 2011 6:09 PM
To: Binkley, Steve
Cc: Brinkman, Bill; Hurlbut, Brandon; Sheron, Brian; Poneman, Daniel; Harold McFarlane; Harold Denton; Adams, Ian; John Holdren; JOE H. PAYER; Kelly, John E (NE); John Grossenbacher; Owens, Missy; Per Peterson; Lyons, Peter; Phil Finck; Dick Garwin; Lee, Richard; Bob Budnitz; Rolando Szilard; SCHU; Aoki, Steven; Koonin, Steven; Steve Fetter; Binkley, Steve; DAgostino, Thomas
Subject: Determining the water level in the dry well by exploiting the (closed) organ-pipe resonance of the air in the dry well.

Exploiting the (closed) organ-pipe resonance of the air in the dry well, which is a function of the water level.

Here we are allowed to put a ;high-power shaking transducer on the building and to drive it for many cycles of oscillation in the range of the dry-well resonance (swept over the range of assumed water levels). Or we could use repetitive thumps and stack the responses from a receiving geophone on the building. And do FFT to determine the Green's function.

I will estimate this.

Dick Garwin

00135

Lee, Richard

From: Per F. Peterson [peterson@nuc.berkeley.edu]
Sent: Thursday, March 31, 2011 7:36 PM
To: Sheron, Brian; Lee, Richard
Subject: Fwd: RE: Spent fuel pool level indication

Dear Brian and Richard,

Just to queue this issue for the U.S. lessons-learned effort, the capability to measure spent fuel pool level, as well as level in sumps and low points in reactor and turbine buildings, has clearly emerged as a major problem in responding effectively to the accident in Japan.

One back fit that may be warranted for U.S. plants would be a system to provide continuous level indication for these locations, with the simplest instrument probably being a bubbler tube. It would need to have an anti-syphon valve if the pressure gage is at an elevation below the maximum water level expected (e.g., anti-syphon would be needed for BWR spent fuel pools, but probably not for reactor building sumps).

Regards, Per

Date: Wed, 30 Mar 2011 21:18:58 -0700
To: "Bedesem, Peter" <PWB2@pge.com>
From: "Per F. Peterson" <peterson@nuc.berkeley.edu>
Subject: RE: Spent fuel pool level indication

Cc: Kelly.John

Pete,

Thank you. It is useful to know that this spent fuel pool level instrumentation provides a high/low indication.

The inability to measure the spent fuel pool level from a remote location is turning out to be a substantial issue at Fukushima. I expect that one of the lessons learned will involve installing level sensors that can detect level continuously to a very low inventory. One option would be a bubbler tube with an antisyphon valve, where the pressure needed to inject air into the tube (causing it to bubble out the end) gives one a measurement of the level.

-Per

Per,

Sorry I was out of town.

00136

Here's the device we use - it's works on buoyancy. From what I can tell, We have a simple HI and LO alarms:

http://us.magnetrol.com/products.aspx?product=748

Pete

-----Original Message-----
From: Per F. Peterson [mailto:peterson@nuc.berkeley.edu]
Sent: Tuesday, March 22, 2011 7:35 AM
To: Bedesem, Peter
Subject: Spent fuel pool level indication

Pete,

I'm involved in a group advising DOE on approaches to help the Japanese.

Could you help me get a quick answer to this question, how is level measured in the spent fuel pools at Diablo Canyon? One cannot have penetrations at the bottom of the pool to do this by pressure, I assume, so an alternative approach must be taken.

-Per

--

--

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--

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Lee, Richard

From: Richard L Garwin [rlg2@us.ibm.com]
Sent: Thursday, March 31, 2011 6:05 PM
To: Bob Budnitz; Adams, Ian
Cc: Brinkman, Bill; Narendra, Blake; Hurlbut, Brandon; Sheron, Brian; Butnitz, Bob (pacbell.net); Smith, Haley; McFarlane, Harold; Adams, Ian; Kelly, John E (NE); Grossenbacher, John (INL); Pitzer, Karrie S.; Chambers, Megan (S4); Owens, Missy; Miller, Neile; Fitzgerald, Paige; Peterson, Per; Lyons, Peter; Finck, Phillip; Garwin, Dick (EOP); Lee, Richard; Budnitz, Bob; Szilard, Ronaldo; Steve Fetter; Aoki, Steven; Binkley, Steve; Mustin, Tracy
Subject: RE: Nuclear science group call - Today at 4:00pm EDT

The Millstone-1 BWR refueling plugs just lie in place, according to this response from Millstone.

So removal of the plug and flooding of the top of the steel drywell liner is a real possibility. But need to analyze the heat transfer impedance by a permanent gas bubble in the drywell.

Dick Garwin

****

----- Forwarded by Richard L Garwin/Watson/Contr/IBM on 03/31/2011 06:03 PM -----
From: Skip J Jordan <skip.j.jordan@dom.com>
To: Richard L Garwin/Watson/Contr/IBM@IBMUS
Cc: Jeff D Semancik <jeff.d.semancik@dom.com>, "SChu@hq.doe.gov" <SChu@hq.doe.gov>
Date: 03/31/2011 05:42 PM
Subject: RE: Urgent question.

Dick,

The shield plugs above the reactor head were interlocking and stepped out slightly in diameter to fit into place. The Refueling Deck crane on the 108 foot level was capable of removing the shield blocks.

From: Richard L Garwin [mailto:rlg2@us.ibm.com]
Sent: Thursday, March 31, 2011 4:51 PM
To: Skip J Jordan (Generation - 6)
Cc: Jeff D Semancik (Generation - 4); SChu@hq.doe.gov
Subject: Urgent question.

Skip, regarding the shield plug above the reactor head. These need to be removed for refueling; are they bolted down or can they just be lifted by the main crane (or a massive construction crane)?

Thanks very much.

Dick Garwin

00137

Lee, Richard

From: Binder, Jeffrey L. [binderjl@ornl.gov]
Sent: Thursday, March 31, 2011 10:37 PM
To: Lee, Richard
Subject: RE: SFP at Unit 3

Heard a rumor that there was a problem. Kelly mentioned it on the phone today so I thought I would ask.

-----Original Message-----
From: Lee, Richard [mailto:Richard.Lee@nrc.gov]
Sent: Thursday, March 31, 2011 10:23 PM Eastern Standard Time
To: Binder, Jeffrey L.
Subject: RE: SFP at Unit 3

Jeff:

As of 1800 03/31/2011, we understood Unit 3 pool is getting fresh water injection via the cooling and purification line. Did something happen?

Richard

From: Binder, Jeffrey L. [binderjl@ornl.gov]
Sent: Thursday, March 31, 2011 4:59 PM
To: Lee, Richard
Subject: SFP at Unit 3

Richard:

Do you know anything about the status?

Jeff

00138

Lee, Richard

From: Powers, Dana A [dapower@sandia.gov]
Sent: Thursday, March 31, 2011 7:43 PM
To: Lee, Richard
Subject: Briefing slides

Richard I received the briefing slides - lots of pictures - about 12 MB. I am going to turn them into a pdf. I will also break them down into 3 power point packages labeled "331A", 331B, and 331C" and send them that way. Let me know which works so if we have to do this again we'll know what to do. Dana

00139

Lee, Richard

From: Richard L Garwin [rlg2@us.ibm.com]
Sent: Friday, April 01, 2011 7:57 PM
To: Adams, Ian
Cc: Brinkman, Bill; Narendra, Blake; Hurlbut, Brandon; Sheron, Brian; Butnitz, Bob (pacbell.net); Smith, Haley; McFarlane, Harold; Kelly, John E (NE); Grossenbacher, John (INL); Pitzer, Karrie S.; Chambers, Megan (S4); Owens, Missy; Miller, Neile; Fitzgerald, Paige; Peterson, Per; Lyons, Peter; Finck, Phillip; Garwin, Dick (EOP); Lee, Richard; Budnitz, Bob; Szilard, Ronaldo; Steve Fetter; Aoki, Steven; Binkley, Steve; Mustin, Tracy
Subject: Major TEPCO action reported.

Regarding my characterization of 16,000-gallon (60 ton) tanks as "tiny," I am glad to see in the current SITREP

"World Nuclear News reports Tepco plans to construct a 6000 tonne water tank as well as a 4000 tonne pond. These will work in conjunction with a 20 tonne per hour treatment facility to handle water from drainage canals around all six reactors at the plant. The tank and pond should be complete around the middle of this month, with the treatment facility following about two weeks later. The set-up should let the company mitigate the discharges to sea by safely storing and sampling the water and only discharging it after treatment. (1800, 4/1 SITREP)"

Hope it is true.

TEPCO seems to be taking some reasonable action in transferring dosimeters from other plants to Fukushima Daiichi.

Dick Garwins

00140

Lee, Richard

From: Richard L Garwin [rlg2@us.ibm.com]
Sent: Tuesday, April 05, 2011 4:53 PM
To: Larzelere, Alex
Cc: DL-NITsolutions; Lee, Richard
Subject: Suggestions re Tuesday, April 5 "Experts meeting."

1. Steve Chu is right to propose "junk shot" with graded size materials. Much more feasible here than when thousands of psi pressure difference was involved.

2. Does "blasting" mean penetration by US Army Engineers shaped charges. I will give an illustration of the hole made by 15-lb and 40-lb shaped charge. I do believe it goes through rebar, too.

3. Abrasive-loaded water jet does need mechanical motion, and can be combined with low-pressure jet and sucking to take out abrasive.

Dick Garwin

00141

Lee, Richard

From: Kelly, John E (NE) [JohnE.Kelly@Nuclear.Energy.Gov]
Sent: Tuesday, April 05, 2011 4:58 PM
To: Lee, Richard
Subject: RE: Call to Japan

I will - Lyons is in Japan and we didn't want to conflict with his appointments

---

From: Lee, Richard (NRC)
Sent: Tuesday, April 05, 2011 4:55 PM
To: Kelly, John E (NE)
Subject: RE: Call to Japan

Thx, John:
Please let me know future ones?

From: Kelly, John E (NE) [mailto:JohnE.Kelly@Nuclear.Energy.Gov]
Sent: Tuesday, April 05, 2011 4:13 PM
To: Lee, Richard
Subject: Re: Call to Japan

No calls scheduled
John E Kelly

From: Lee, Richard (NRC)
To: Kelly, John E (NE)
Sent: Tue Apr 05 15:41:15 2011
Subject: Call to Japan

Hi, John:

Do you a call to Japan today?. If yes, what time is it scheduled for?

Thx, Richard

Lee, Richard

From: Powers, Dana
Sent: Tuesday, April 05, 2011 5:47 PM
To: Lee, Richard
Subject: RE: per strikes

They are getting beaten to death in the press over deliberate releases of radioactive water to free up storage space for more contaminated water. They need to be careful or the government may fall over all this. Dana

---

From: Lee, Richard
Sent: Tuesday, April 05, 2011 5:43 PM
To: Powers, Dana
Subject: RE: per strikes

Let's send some of the DOE Sci. Council advisors to have this done. I have at least 2 suggestions!

From: Powers, Dana
Sent: Tuesday, April 05, 2011 5:41 PM
To: Lee, Richard
Subject: RE: per strikes

The rarefaction off the backside of the concrete is the way we kill people inside bunkers. Dana

---

From: Lee, Richard
Sent: Tuesday, April 05, 2011 5:39 PM
To: Powers, Dana
Subject: RE: per strikes

Do you want to know who else I tune off too?

From: Powers, Dana
Sent: Tuesday, April 05, 2011 5:38 PM
To: Lee, Richard
Subject: RE: per strikes

Now we are seriously discussing using shaped charges in the vicinity of the head – madness. Dana

---

From: Lee, Richard
Sent: Tuesday, April 05, 2011 5:37 PM
To: Powers, Dana
Subject: RE: per strikes

I have automatically tuned off as soon as Per starts talking.

From: Powers, Dana
Sent: Tuesday, April 05, 2011 5:35 PM
To: Lee, Richard
Subject: RE: per strikes

No, stipping if the try to cool the outside of the drywell. In fact, the AP1000 flow even with the engineering stipes and you get streaks of wet and dry – without the engineering you will get a bunch of uneven streaks with low heat transfer. Dana

00143

From: Lee, Richard
Sent: Tuesday, April 05, 2011 5:33 PM
To: Powers, Dana
Subject: RE: per strikes

You are talking about sand vs. water. Correct?

From: Powers, Dana
Sent: Tuesday, April 05, 2011 5:28 PM
To: Lee, Richard
Subject: per strikes

Once again, Per can find fault with any suggestion. I think he is right in this case. dana

Lee, Richard

From: Richard L Garwin [rlg2@us.ibm.com]
Sent: Thursday, April 07, 2011 4:51 PM
To: Larzelere, Alex
Cc: Busby, Jeremy T.; DL-NITsolutions; Douglas.Burns@inl.gov; Lee, Richard; Schneider, Steve
Subject: Garwin comments on WaterJet performance.

I am very glad to see the performance of the WJ in cutting 20" concrete. Who is responsible for this progress?

One of my ad hoc colleagues has done some tests also, reported here, but I think you are doing well.

Dick Garwin

Status Report, morning 04/07:

I managed to get my hands on one 4" diameter x 8" long test plug of high pressure concrete. I felt this would be adequate for initial tests based upon advice from water jet manufacturer. Manufacturer states an ability to cut only 6" deep with our 49,000 psi jet.

Test 1:

Machine was set to pierce with low pressure (32,000 psi) and cut with high pressure (49,000) using 120 grit garnet.
Omax machines use a machining index of numerical value as a variable in machining speed calculations.
I selected a limestone setting with machinability of 6,300. (predicted cut time = 18 minutes)
Program was written such that the path was concentric rings graduating in size in increments of 0.050" until 0.550" diameter then increasing in graduation to 0.075" until achieving a hole diameter of 0.700"

When the hole diameter was approximately .250", the jet perforated the material. The test was allowed to progress until complete whereupon it was discovered that the hole was cut through with a reduction in diameter of about 28%. Results are useful only as reference as the goal was to observe waterjet performance evacuating a blind pocket.

Test 2:

Machine was set to pierce with low pressure (32,000 psi) and cut with high pressure (49,000) using 120 grit garnet.
Omax machines use a machining index of numerical value as a variable in machining speed calculations.
I selected a custom setting with machinability of 10,000. (predicted cut time = 10.5 minutes)
Program was written such that the path was concentric rings graduating in size in increments of 0.050" until 0.550" diameter then increasing in graduation to 0.075" until achieving a hole diameter of 0.700"

An obstruction in the garnet hopper prevented garnet flow. Test was allowed to continue to demonstrating performance of water only cutting. Resulting hole was tapered to a near point approximately 4" deep. Finish throughout the bore was rough and wave like. It appears the various stones have an increased negative effect on the high pressure jet when no garnet is used.

00144

Test 3:

(Settings identical to Test 2)
Machine was set to pierce with low pressure (32,000 psi) and cut with high pressure (49,000) using 120 grit garnet.
Omax machines use a machining index of numerical value as a variable in machining speed calculations.
I selected a custom setting with machinability of 10,000. (predicted cut time = 10.5 minutes)
Program was written such that the path was concentric rings graduating in size in increments of 0.050" until 0.550" diameter then increasing in graduation to 0.075" until achieving a hole diameter of 0.700"

Having garnet in the jet greatly increased cutting potential. When path diameter was nearing 0.450", the jet again broke through the full 8 inch thickness of the material. This is promising and suggests much greater travel speed and larger step-over distances are possible.

Test 4 will increase stepover values to 0.150" and 0.200". Machinability index will be increased to 15,000 also.

This was in response to email from Dick Garwin:

I need to do more research on US Army Engineer shaped charges of 40 lb mass, but let's scope the abrasive-loaded waterjet approach.

The advantage of the WJ IS just that it cuts a narrow kerf, so let's see how we can do that in a deepening cylindrical kerf. I have seen WJ cut through 6 inches of steel, so lets plan on that. initial depth. We can speed the process by detecting the presence of reinforcing rod (probably by the sound (or ultrasound) of the jet, and slowing the rotary feed for the steel and adding garnet at that time. But the kerf might be only 1-mm thick and 6 inches deep. Assume that we are boring a 6-in diam (15 cm) hole.

Then we could reset the jet to make a similar cylindrical kerf at 12 cm diam, 15-cm deep. We could then put a 1-cm ring of DetaSheet at the base of the kerf and break off the cylindrical ring of thickness about 13 mm and remove it in one piece or many.

Now comes the trick. Assume that deep cuts parallel to the axis must use a jet that is similarly parallel to the axis but emerges from a nozzle of OD 1 cm. We make room for the nozzle at 6-cm depth by angling it at 45 deg and chewing out a ring-shaped chamber of 1-cm radial depth and 2-cm axial extent, both outside and inside the cylindrical kerf.. That will allow repositioning the nozzle (a right-angle nozzle) so that the center of the nozzle can again be lined up with the outside of the normal cylindrical kerf to jet another 15-cm depth. And similarly for the 12-cm diameter kerf. Then we put in the DetaSheet ring and remove another 6 cm of core.

If there happens to be an axial rebar to keep the core from coming out, one can turn the right-angle nozzle to cut radially inward to sever the core cleanly.

And so it goes through wall after wall and through the primary containment ("dry well").

Here is a pretty good report of the accident sequence, http://www.seyth.com/ressources/quake/AREVA-Document.pdf
See pp. 4 and 7 for a sketch of the configuration.

Please criticize my WJ scenario.

Look forward to the call.

Dick Garwin

Lee, Richard

From: Lyons, Peter [Peter.Lyons@Nuclear.Energy.gov]
Sent: Friday, April 08, 2011 1:35 AM
To: Larzelere, Alex; DL-NITsolutions
Subject: Re:
Attachments: image001.jpg

Fyi-I discussed the possibility of an acetylene explosion with senior tepco officials and was assured that NO acetylene was present at unit 4. The work in progress involved cutting out the existing shroud to prepare for replacement. I was told that ONLY EDM was used for cutting and no acetylene torch was present.
Pete lyons

From: Larzelere, Alex
To: DL-NITsolutions
Sent: Wed Apr 06 15:09:00 2011
Subject:

Everyone,

Attached is a report that was prepared assessing the Unit 4 explosion. This might be useful in our discussions later today. Also attached is a summary report from the IAEA that has some good diagrams (including one of the leak into the ocean) that you might find useful.

Regards,

Alex

Alex R. Larzelere
Director, Advanced Modeling and Simulation Office
Office of Nuclear Energy (NE-71)
U.S. Department of Energy
202-586-1906
Alex.Larzelere@nuclear.energy.gov

AMSO
ADVANCED MODELING & SIMULATION OFFICE

00145

Lee, Richard

From: Larzelere, Alex [alex.larzelere@nuclear.energy.gov]
Sent: Friday, April 08, 2011 7:28 AM
To: Sheron, Brian; Lee, Richard
Cc: Kelly, John E (NE)
Subject: RE: Email Contact

Brian,

I am going to talk with Richard today to track this down. I sent out another couple of emails last night and I do not think they go through to Richard. I have never gotten a bounce email so it does not seem that I have a bad address or that his mailbox it full.

I have a funny suspicion that my address may have gotten on his spam list that that is the reason I am being blocked. I did get one email from Richard last night that apparently he had to send twice. In it was a bounce that said:

'each some or all of the intended recipients.

)ject: VGs
it: 4/7/2011 6:06 PM

The following recipient(s) cannot be reached:

'Larzelere, Alex ' on 4/7/2011 6:06 PM
None of your e-mail accounts could send to this recipient.

This is now a challenge and I will be working to figure out the answer.

Thanks for the help.

Regards,

Alex

From: Sheron, Brian
Sent: Friday, April 08, 2011 7:13 AM
To: Larzelere, Alex; Lee, Richard (NRC)
Cc: Kelly, John E (NE)
Subject: RE: Email Contact

It is weird because I receive the slides with no problem.

From: Larzelere, Alex [mailto:alex.larzelere@nuclear.energy.gov]
Sent: Thursday, April 07, 2011 6:16 PM
To: Lee, Richard
Cc: Kelly, John E (NE); Sheron, Brian
Subject: Email Contact

Richard,

I am completely baffled by this – but I am going to find the problem. Please confirm that this is a good address. I am sending this email as a reply to one you sent, so hopefully this will work.

If so, I will next start sending files of different sizes to see what gets through.

00146

I am really sorry for the trouble.

Regards,

Alex

From:	phil
To:	Bonaccorso, Amy; OPA Resource
Subject:	how to get the uranium out of the water and concrete, while still in the reactor.
Date:	Monday, April 04, 2011 6:16:06 AM
Attachments:	Uranium removal from water and concrete.pdf

Read attached, running the radioactive runoff through iron, and any pile of iron, might cement the uranium, and prevent it's total loss to the sea. Yes, really.

I've added the apparently required additional clarifications for water absorbing polymers.

Also, how to get the uranium out of the water and concrete, while still in the reactor.

That would allow the recovery of expensive reactor fuel, reduce the danger from the site, and calm the public a great deal.

Attached, sorry, didn't have time to prune it, lots more is available, proven attached, but, there are keys there to speed the choice of action by a competent nuclear chemist/physicist.

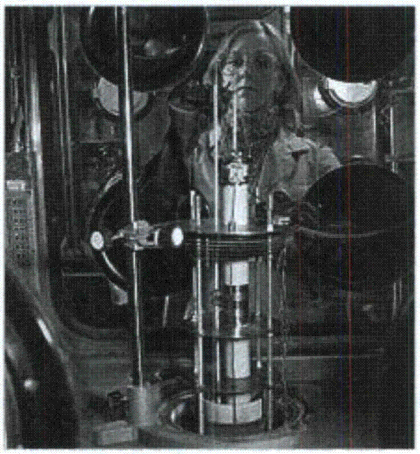
Please hurry.

00147

From:	phil
To:	Bonaccorso, Amy; OPA Resource
Subject:	Recovering nuclear fuel electrlytically, from the reactor tank directly, slow, but it will take out the exposed fuel first.
Date:	Monday, April 04, 2011 12:50:22 AM

http://en.wikipedia.org/wiki/Electrowinning

Adapt for in situ



Electrorefining technology converting spent commercial nuclear fuel into metal.

Emergency graphite electrodes : ?
use standard legging nylon stockings, stuffed with carbon from burnt wood, submerge halfway into water pool, and keep top dry, but embed copper wires in the dry top part of the carbon.

Use lots of them in parallel, to keep the individual currents down for many reasons.

That will reduce copper pollution of the pool, and current limit the electrolysis to reduce gas production rate, and double layer heating and resistance.

Lee, Richard

From: Powers, Dana A [dapower@sandia.gov]
Sent: Thursday, March 31, 2011 11:31 AM
To: Lee, Richard
Subject: RE: Recommendation

I cannot help but wonder if the oxygen removal they seek to reduce corrosion will not be frustrated by the radiolytic formation of hypochlorate ion - an oxidant of well known power. It may be that the ambient hydrogen partial pressure suppresses hypochlorate formation which would be good. I need to get a radiolysis model set up and check this. Dana

-----Original Message-----
From: Lee, Richard [mailto:Richard.Lee@nrc.gov]
Sent: Thursday, March 31, 2011 8:25 AM
To: Dana Powers (Home); Powers, Dana A
Subject: FW: Recommendation

fyi

-----Original Message-----
From: Per F. Peterson [mailto:peterson@nuc.berkeley.edu]
Sent: Wednesday, March 30, 2011 11:49 PM
To: Brinkman, Bill; Binder, Jeff; Hurlbut, Brandon; Sheron, Brian; Poneman, Daniel; Connell, Elizabeth; McFarlane, Harold; 'Harold Denton'; Adams, Ian; John Holdren; 'JOE H. PAYER'; Kelly, John E (NE); Grossenbacher, John (INL); Owens, Missy; Peterson, Per; Lyons, Peter; Finck, Phillip; Garwin, Dick (EOP); Lee, Richard; Budnitz, Bob; Szilard, Ronaldo; Aoki, Steven; Koonin, Steven; Steve Fetter; Binkley, Steve; DAgostino, Thomas
Subject: Fwd: Recommendation

Following up on Steve Chu's email on commercial equipment that can remove oxygen from boiler feedwater, this note from Tom Devine in our Materials Science and Engineering department also identifies the reduction of oxygen in feedwater as the single most effective step that can be taken to reduce corrosion of stainless and carbon steel in the reactor vessels and primary containment.

Given this information, it would appear to be prudent to move now to procure commercial equipment to strip oxygen from the water being injected into the Fukushima reactors (probably using the commercial membrane equipment is better than using hydrazine, an option that Tom mentions). The equipment might be installed at the dam that is currently providing the fresh water supply, since the conditions are less challenging there and access should be easier to get the equipment set up.

One way of looking at this problem is that essentially all of the oxygen that is injected into these reactors will end up reacting with and corroding metal structures. Just as stopping the flow of salt into the reactors was an important action to take on an expedited basis, now stopping the injection of oxygen is likely to also be a prudent measure. (Once recycled condensate is available from evaporators, it will also have relatively low oxygen content, but this will not be available for weeks to a few months).

-Per

>
>Date: Wed, 30 Mar 2011 14:19:08 -0700
>Subject: Recommendation

00142

>From: devine@berkeley.edu
>To: peterson@nuc.berkeley.edu
>
>Per,
>
>The following is my recommendation (and explanation) of what should be
>done regarding treatment of water inside the reactors at Fukushima. In
>brief, my specific recommendation is to add sulfite or hydrazine to the
>water inside the reactors. My explanation is as follows.
>
>At present, seawater inside the reactor is saturated with air. The
>combination of a high water temperature, a high chloride ion
>concentration (i.e., that of sea water or higher), and dissolved oxygen
>provides a very corrosive electrolyte to stainless steel (e.g., cooling
>pipelines and cladding on the RPV), and to the nonstainless steels, consisting of carbon
>steel (steam lines), and low alloy steel (RPV). In the present state of
>affairs inside the reactors, stainless steel is susceptible to
>localized corrosion, such as pitting corrosion and crevice corrosion
>(also intergranular corrosion if the stainless steel is sensitized,
>which might be the case for a region close to a weld). Because of
>residual stresses, the stainless steel components are also susceptible
>to environmentally assisted cracking, such as stress corrosion cracking
>(SCC). Non-stainless steel components are susceptible to general corrosion.
>
>Generally, increases in temperature, dissolved oxygen concentration,
>and chloride ion concentration increase the general corrosion rate of
>nonstainless steels and increase the probability of localized corrosion
>of stainless steels. However, there is a limit to how aggressive the
>water can become because the three factors of temperature, chloride ion
>concentration and dissolved oxygen concentration are interdependent.
>Specifically, in an open system the concentration of oxygen that is
>dissolved in water initially increases with increasing temperature, but
>it reaches a maximum value at about 80°C. As temperature increases
>above 80°C, the dissolved oxygen concentration decreases. As a result,
>the corrosion rate in boiling water (not seawater) is about the same as
>the corrosion rate at 30°C. Sodium chloride also limits the amount of
>oxygen that can be dissolved in the water. As the concentration of
>salt dissolved in the water increases, the solubility of oxygen decreases.
>
>The corrosion rate of nonstainless steels when the rate determining
>step is oxygen diffusion approximately doubles for every 30°C increase
>in temperature (this is likely the case inside the reactors). Thus,
>the corrosion rate in seawater at 65°C is approximately 0.12 mm/y. So,
>an approximate value of the maximum corrosion rate of a nonstainless
>steel in high temperature (80°C) seawater is about 0.24 mm/y.
>
>The main risk to stainless steel components is crevice corrosion and
>stress corrosion cracking. If the stainless steel is galvanically
>coupled to nonstainless steel, then the stainless steel is unlikely to
>fail by pitting corrosion but might be susceptible to hydrogen assisted
>cracking if the electrolyte in contact with the stainless steel is in
>an occluded volume.
>
>What is to be done?
>
>Undoubtedly, the most effective action that can be taken to lower the
>probability of a corrosion-related failure inside the reactors is to

>lower the water's dissolved oxygen concentration. This can be achieved
>by adding hydrazine to the water:
>
>$N_2H_4 + O_2 \rightarrow N_2 + 2H_2O$.
>
>Alternatively, sodium sulfite (esp in the presence of a cobaltous salt
>as
>catalyst) can be added as an oxygen scavenger:
>
>$2SO_3^{=} + O_2 \rightarrow 2SO_4^{=}$.
>
>Hydrazine is a toxic, flammable liquid at room temperature (boiling
>point is 114°C) and is sometimes employed as a rocket fuel. However,
>nuclear plant operators might be familiar with handling and using
>hydrazine to deoxygenate water (hydrazine is employed in the procedure
>known as Hydrogen-Water Chemistry, which is employed by some operators of BWRs).
>The products of the hydrazine deoxygenation reaction are benign, while
>the sulfate products of the sulfite deoxygenation reaction might
>contribute to corrosion. However, the corrosive effect of sulfate is
>greatly outweighed by the decrease in dissolved oxygen concentration.
>
>Corrosion inhibitors act either to passivate the nonstainless steel, or
>to diminish the rate of the reduction reaction. Passivating
>nonstainless steel components (e.g., by adding nitrite or phosphate)
>inside the reactors runs the risk of the steel subsequently failing by
>localized corrosion, such as pitting corrosion. In my view, the better
>approach is to try to lower the corrosion rate by inhibiting against
>the electrochemical reduction reaction.
>
>Corrosion is the most likely cause of failure of nonstainless steel
>components. In the near term, the greatest risk to the stainless steel
>components is some form of environmentally assisted cracking. The
>susceptibility of the stainless steel components to SCC is a
>consequence of the high temperature and high chloride ion
>concentrations, as well as the dissolved oxygen concentration.
>Deoxygenating the water will lower the susceptibility to SCC.
>
>In summary, the greatest benefit to both stainless steel and
>nonstainless steel components will be derived from deoxygenating the
>water (such as by the addition of hydrazine or sulfite). In addition,
>you might also consider the addition of an inhibitor that will act to
>lower the rate of the reduction reaction. I'll recommend specific
>inhibitors in a separate message.
>
>Best regards,
>Tom

--

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Lee, Richard

From: Per Peterson [peterson@nuc.berkeley.edu]
Sent: Sunday, March 27, 2011 9:43 PM
To: Richard L Garwin
Cc: Binkley, Steve; Brinkman, Bill; Binder, Jeffrey L.; Hurlbut, Brandon; Sheron, Brian; Poneman, Daniel; Elizabeth.Connell@inl.gov; Harold McFarlane; Harold Denton; Adams, Ian; John Holdren; JOE H. PAYER; Kelly, John E (NE); John Grossenbacher; Owens, Missy; Lyons, Peter; Phil Finck; Dick Garwin; Lee, Richard; Bob Budnitz; Rolando Szilard; SCHU; Aoki, Steven; Koonin, Steven; Steve Fetter; DAgostino, Thomas
Subject: Re: Reducing oxygen in external water supply-- nitrogen sparging.

When water is injected for evaporative cooling, bleeding of steam is also required. This venting can be done periodically, and as venting reduces the containment pressure steam flashes from the suppression pool and thus cools the water. This steam released by venting should also carry hydrogen and oxygen out. So im not sure about the concern on oxygen, unless some alternative path exists for residual heat removal besides venting.

Per

Sent from Per's iPhone

On Mar 27, 2011, at 5:42 PM, Richard L Garwin <rlg2@us.ibm.com> wrote:

> A moment on the web gives me this:
> Nitrogen Sparging and Blanketing of Water Storage Tanks
>
> at http://www.steamcycle.com/nitrogen.htm
>
> Please read and evaluate.
>
> Dick Garwin

00149