From:	Helton, Donald
To:	Flanagan, Michelle
Cc:	Esmaili, Hossein; Lee, Richard
Subject:	RE: Nitriding writeup
Date:	Tuesday, March 12, 2013 10:22:00 AM

Thanks Michelle. Sounds like things are moving along well. Safe travels.

Don

-----Original Message-----
From: Flanagan, Michelle
Sent: Monday, March 11, 2013 4:56 PM
To: Helton, Donald
Cc: Esmaili, Hossein; Lee, Richard
Subject: RE: Nitriding writeup

Don,

We prepared a preliminary write up addressing all of the cladding phenomena you identified a few weeks ago. Before we sent it over, we wanted Sandia to review it to comment on the significance or consideration of these issues in MELCOR modeling. We haven't heard back from them yet, which is the reason for the delay.

Richard and or Hossein, Can you ask Sandia when they will be able to provide their input on this write up?

Thank you,
Michelle

From: Helton, Donald
Sent: Monday, March 11, 2013 8:56 AM
To: Flanagan, Michelle
Cc: Coyne, Kevin
Subject: FW: Nitriding writeup

Michelle,

What is the status on the writeup about cladding phenomena relative to Level 1 PRA success criteria, SFP accident analysis, etc. Have you gotten most of the input you asked for from everyone back in January?

Hope all is well,
Don

From: Helton, Donald
Sent: Friday, February 01, 2013 12:23 PM
To: Flanagan, Michelle; Esmaili, Hossein
Subject: Nitriding writeup

Michelle - Here is my crack at nitriding.

Hossein – Feel free to edit

Nitriding refers to the formation of ZrN when zirconium cladding oxidizes at high-temperatures in an air environment (e.g., > 800C). In general, high-temperature zirconium has a greater affinity for the

C-48

oxygen constituent of air, and thus the reaction between air and high-temperature zirconium is more affected by these reaction kinetics. Nevertheless, ZrN does form (and particularly as the surrounding media becomes depleted of oxygen due to the stronger oxidation reaction) and ZrN may play a role in the behavior of the oxide layer during a phase known as breakaway (wherein the columnar structure of the oxide layer begins to develop radial cracks along the columnar grain boundaries. Nitriding as an additional heat source during autocatalytic zirconium oxidation transients is only important in oxygen-starved situations, based on insights obtained from recent experiments (QUENCH-16 and OECD-SFP). Following reflood, re-oxidation takes place which releases the nitrogen stored in nitrides. NRC, Sandia National Labs, and the Paul Sherrer Institut have ongoing discussions about the possibility of a MELCOR code/model development activity to address this phenomena, which is presently not modeled by the code.

As discussed above, nitriding is only relevant when nuclear fuel is undergoing a severe accident in an air environment, and only in situations where the conditions may become oxygen-starved. As such, this phenomena is not expected to affect prediction of reactor accidents, with the possible exception of accidents during shutdown operations when the vessel head has been removed. Similarly, the phenomena would not be of relevance to SFP boiloff or partial draindown accidents, but may be relevant for SFP complete draindown accidents where the surrounding building is predominantly intact (relates to the need to attain oxygen-starved conditions).

Even for the above-identified accident types where nitriding may be important, the existing modeling omission remains reasonable. In these cases, nitriding affects the degree of uncertainty in the prediction of severe accident-regime temperature ramp rates and peak temperatures following the onset of autocatalytic zirconium oxidation (i.e., temperatures at and above typical Level 1 PRA core damage surrogates and emergency operating procedure to severe accident management guideline transition thresholds). The potential affect of nitriding on the uncertainties in this regime is likely commensurate with other uncertainties present in this regime. Thus, while more work would be beneficial, it would not be expected to invalidate the current modeling approach.

Don Helton
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

From:	Helton, Donald
To:	Coyne, Kevin
Subject:	FW: Nitriding writeup
Date:	Tuesday, March 12, 2013 10:22:00 AM

FYI - FSTB is still actively working the cladding phenomena item, and is getting a peer review of sorts from SNL...

-----Original Message-----
From: Flanagan, Michelle
Sent: Monday, March 11, 2013 4:56 PM
To: Helton, Donald
Cc: Esmaili, Hossein; Lee, Richard
Subject: RE: Nitriding writeup

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Cc: Coyne, Kevin
Subject: FW: Nitriding writeup

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Sent: Friday, February 01, 2013 12:23 PM
To: Flanagan, Michelle; Esmaili, Hossein
Subject: Nitriding writeup

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Hossein – Feel free to edit

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E-49

the behavior of the oxide layer during a phase known as breakaway (wherein the columnar structure of the oxide layer begins to develop radial cracks along the columnar grain boundaries. Nitriding as an additional heat source during autocatalytic zirconium oxidation transients is only important in oxygen-starved situations, based on insights obtained from recent experiments (QUENCH-16 and OECD-SFP). Following reflood, re-oxidation takes place which releases the nitrogen stored in nitrides. NRC, Sandia National Labs, and the Paul Sherrer Institut have ongoing discussions about the possibility of a MELCOR code/model development activity to address this phenomena, which is presently not modeled by the code.

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Ph: 301 251-7594

From: Helton, Donald
To: Wood, Jeffery
Cc: Coyne, Kevin; Kuritzky, Alan
Subject: Castle Paper - SFP input
Date: Monday, March 11, 2013 2:06:00 PM
Attachments: Castle Meeting - 2013.docx
ACRS - Dec 2012 - SFP L12.pptx

Jeff,

Attached is a starting point for the SFP component of the Castle Meeting paper. Its 2 pages as is, focuses exclusively on the Vogtle PRA plans, and doesn't say anything that couldn't be combed out of the publicly-available TAAP.

I've included the presentation that I was going to give the ACRS back in December, but it doesn't really follow this particular paper, per se, so I'm not sure if it is useful.

Don

Don Helton
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C/50

From: Helton, Donald
To: Jones, Steve
Subject: FW: Interesting graphic of 1F3 SFP debris
Date: Friday, March 08, 2013 8:42:00 AM
Attachments: image001.png

FYI

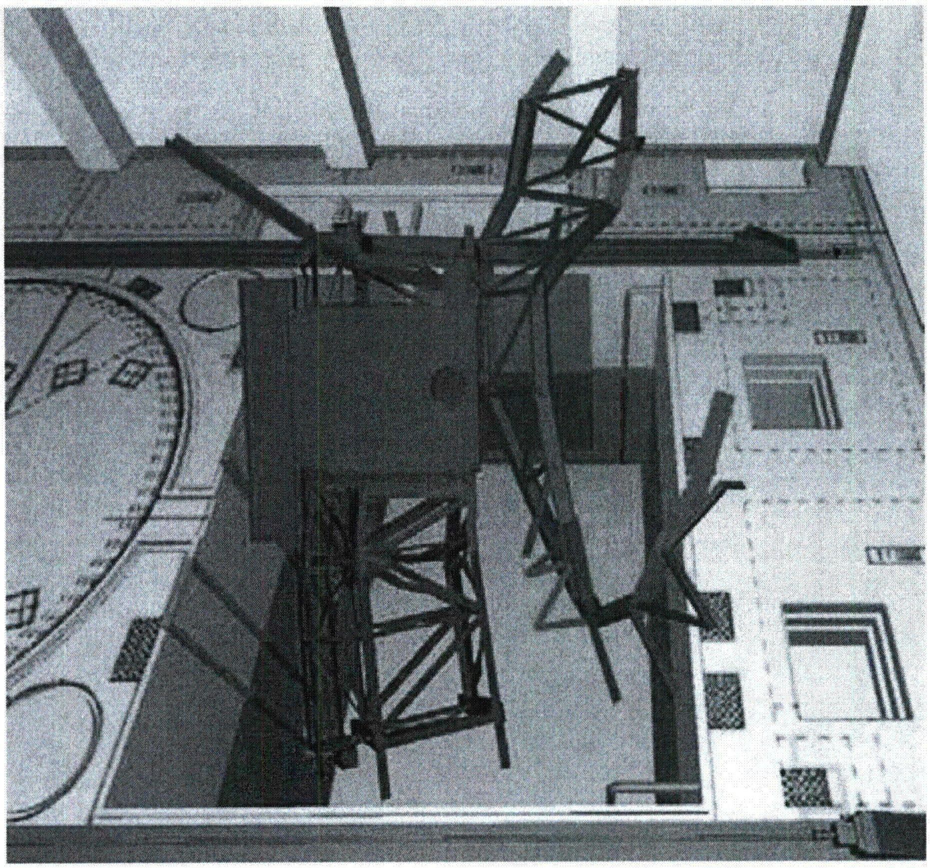
From: Helton, Donald
Sent: Friday, March 08, 2013 8:42 AM
To: Esmaili, Hossein; Pires, Jose
Subject: FW: Interesting graphic of 1F3 SFP debris

FYI – PDF Pages 160 – 174 have some new (at least to me) photos of the 1F SFP debris…

From: Marksberry, Don
Sent: Friday, March 08, 2013 6:12 AM
To: Helton, Donald
Subject: Interesting graphic of 1F3 SFP debris

See pdf page 172

http://www.tepco.co.jp/nu/fukushima-np/roadmap/images/d130307_01-j.pdf



E-51

From: Helton, Donald
To: Algama, Don; Wagner, Brian; Nosek, Andrew; Chang, James
Cc: Esmaili, Hossein; Compton, Keith; Pires, Jose; Murphy, Andrew
Subject: Another perspective
Date: Tuesday, March 05, 2013 2:57:00 PM

All,

If interested, EPRI TR-1025295 (2012 SAM TBR Update) talks about EPRI's views on BDBA SFP accidents in the context of accident management. The SFP-related material is sprinkled throughout, and in Appendix EE. However, there is a fairly succinct writeup on PDF pages 72-74 (of Volume 1) that touches upon things like accessibility and recriticality. I'm not suggesting that we should allow an industry study to influence our views any more than a UCS study, but it is a good example of how a different group of accident analysts view some of the same issues.

The report is downloadable from EPRI's website by typing in "1025295" in to the search field. I would have simply attached it, but the file size is quite large and it wouldn't let me extract the aforementioned section.

Don

Don Helton
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Ph: 301 251-7594

E-52

From:	Helton, Donald
To:	Esmaili, Hossein; Pires, Jose
Subject:	FYI - 2-pager on 1F1 SFP sloshing
Date:	Tuesday, February 26, 2013 8:48:00 AM
Attachments:	U1-sloshing-sim-handouts_130218_02-e.pdf

FYI

Don Helton
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Postal address: US NRC / MS CSB4-C7M / Washington, DC 20555
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E-53

From: Helton, Donald
To: Corson, James
Subject: tentative OCP definitions, etc.
Date: Friday, February 22, 2013 2:08:00 PM
Attachments: x - DOC - SFP Level 1 and 2 on 2-22-13.docx
initial time screening table.docx

The tables we were looking at are on page 9 of the 1st attachment and in the 2nd attachment; all subject to change...

Don Helton
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Ph: 301 251-7594

C-54

Thanks for considering this input.

Steve

From: Algama, Don
Sent: Wednesday, February 06, 2013 9:38 AM
To: Jones, Steve
Subject: RE: SFPSS: IOWG Document Review and Comment

Steve:

Thanks for your time and effort in helping us with this report.

-Don A.

From: Jones, Steve
Sent: Wednesday, February 06, 2013 9:37 AM
To: Algama, Don
Cc: Casto, Greg
Subject: RE: SFPSS: IOWG Document Review and Comment

Don,

My higher level comments are attached. I apologize for overshooting the due date, but I have few comments to add. I believe the report was well prepared overall.

Thanks,

Steve

Steven R. Jones
Sr. Reactor Systems Engineer
NRR/DSS/SBPB
301-415-2712

From: Algama, Don
Sent: Tuesday, February 05, 2013 8:51 AM
To: Powell, Eric; Tegeler, Bret; Jones, Steve; Ennis, Rick; Mitman, Jeffrey; Witt, Kevin; OPA Resource; Cahill, Christopher; Bernhard, Rudolph; Kozak, Laura; Runyan, Michael; Ziedonis, Adam
Cc: Esmaili, Hossein; Zabel, Joseph; Burnell, Scott; McIntyre, David
Subject: RE: SFPSS: IOWG Document Review and Comment

Dear IOWG:

Please provide your comments by COB today. Thanks for your time in advance.

Thanks,

C/55

Don A.

From: Algama, Don
Sent: Monday, January 28, 2013 5:25 PM
To: Powell, Eric; Tegeler, Bret; Jones, Steve; Wood, Kent; Bowman, Eric; Ennis, Rick; Mitman, Jeffrey; Witt, Kevin; Schrader, Eric; Sullivan, Randy; OPA Resource; Cahill, Christopher; Bernhard, Rudolph; Kozak, Laura; Runyan, Michael; Ziedonis, Adam; QTE Resource
Cc: Lee, Richard; Esmaili, Hossein; Zabel, Joseph; Burnell, Scott; McIntyre, David; Replogle, George; Casto, Greg
Subject: SFPSS: IOWG Document Review and Comment

Dear Inter-Office Working Group (IWOG):

Purpose of Email:
Provide a draft copy of the SFPSS Report to the IOWG for their Review and Comment.

Actions Requested:

1. Please review the SFPSS document (Second attachment).

2. Please provide comments in the EXCEL spreadsheet that is attached (First attachment). If for some reason a comment cannot be included in the EXCEL file, please contact me. Example of what is expected is already included in the EXCEL file.

3. Please provide comments by:
- IOWG and OPA: 02.05.13
- Tech Editing: 02.08.13

4. OPA: Please be aware that the following individuals in your organization are already familiar with this report. They are: a. Scott Burnell and b. David McIntyre

5. QTE: I would appreciate a level 3 review if possible. There is an expectation that this report will become a RES NUREG.

6. Regions 2, 3 and 4: Please provide your comments to Christopher Cahill. He will be compiling comments for the Regions.

7. Please do not share the SFPSS report as it is in draft form, and is not public.

8. Understand the expected process for concurrence of this report. Please see the below section on document finalization.

Outcome of Action:
Have IOWG Team members provide their insight and knowledge to improve the SFPSS report.

Background:
SFPSS Team technical review has ended. We are now in the Inter-Office Review and Comment phase of the report completion. Please see the third attached document for more details.

SFPSS Document Finalization Plan:
Step 1 (ACRS SC Meeting):

- Review and comment by SFPSS Team. Email sent and received from Team by PM for group consensus

- Review and comment by Inter-Office Working Group (IWOG). Email sent and received from IOWG Team by PM for group consensus.

- Review and comment by RES and other office BCs. Email sent by FSCB BC requesting BC review, send comments to SFPSS PM for incorporation.

- Concurrence memo for NRR, NSIR, NRO and NMSS Div. Dir. Unlike the prior steps this will be a memo from the RES/DSA Div. Dir. to the other Div. Dir. for concurrence.

- Report sent to ACRS sub-committee for their review via email and later to the full committee.

Step 2 (Document to NRR and Commission):

- Once ACRS letter has been received and comments incorporated, the final draft SFPSS will be sent for office-level concurrence. The Step 1 reviewers will receive a copy of the final draft by email from the PM.

- Since the Div. Dir. have already concurred on the SFPSS report in Step 1. A red-line strike out version will also be provided comparing the draft ACRS version and the final version of this document that will include comments from the ACRS. This will be provided to aid the office-level concurrence process. Once offices' comments have been incorporated and all concurrences received, the SFPSS report will be finalized.

- The SFPSS will be sent to the Commission via either a RES info SECY or included as an attachment to the NRR SECY on the Tier 3 SFP Transfer item. If the report is sent to the Commission by RES, the info SECY will be included with the final draft report when it is sent for office concurrence – so offices will concur on the SECY and SFPSS attachment at the same time. If the report will be part of NRR's SECY, the offices will concur on the report only, and the final report (with all offices concurrences) will be provided to NRR.

Thank you for your time and patience,
Don A.
(301.251.7940)

From: Helton, Donald
To: Wagner, Brian
Subject: RE: FYI
Date: Thursday, January 31, 2013 8:46:00 AM

Oh, but I meant to mention, that cask mislead is the first of these reports I've seen in over a year that had anything to do with dry cask storage...and very few have anything to do with the SFP. Most are reactor or emergency response-related...

From: Wagner, Brian
Sent: Wednesday, January 30, 2013 4:57 PM
To: Helton, Donald
Subject: RE: FYI

How does one subscribe to the 50.72 RSS feeds? I can't seem to find it on our website.

Brian

From: Helton, Donald
Sent: Monday, January 28, 2013 4:08 PM
To: Gonzalez, Felix; Barto, Andrew
Cc: Wagner, Brian
Subject: FYI

Happened upon this b/c I subscribe to the 50.72 RSS feeds. Thought you might find this interesting. A bit of mis-load operating experience for PB3... The actual mis-loads (if that is the correct term for putting an assembly in a few months too early) took place in 2001, but were only recently discovered/reported...

Independent Spent Fuel Storage Installation	Event Number: 48698
Rep Org: PEACH BOTTOM Licensee: EXELON GENERATION COMPANY, LLC Region: 1 City: PHILADELPHIA State: PA County: YORK & LANCASTER License #: GL Agreement: Y Docket: 72-29 NRC Notified By: BARRY LEVINS HQ OPS Officer: STEVE SANDIN	Notification Date: 01/25/2013 Notification Time: 09:43 [ET] Event Date: 01/24/2013 Event Time: 11:00 [EST] Last Update Date: 01/25/2013
Emergency Class: NON EMERGENCY 10 CFR Section: OTHER UNSPEC REQMNT	Person (Organization): MARC FERDAS (R1DO) ERIC THOMAS (NRR)

Event Text

NON-COMPLIANCE WITH STORAGE CASK TECHNICAL SPECIFICATION LIMITS

"This report is being submitted pursuant to Transnuclear (TN)-68 Technical Specification (TS) Section 2.2, which requires reporting of non-compliances with the Functional and Operational limits of TS Section 2.1.1.

"A recent review of historical ISFSI [Independent Spent Fuel Storage Installation] fuel characterization data found that in the ISFSI 2001 campaign, a total of four Unit 3 fuel assemblies were loaded into four dry cask storage casks (i.e., one assembly per cask) having been cooled for 9.8 years, with a decay heat value of 0.201 kW each, which is well below the

5-26

0.312 kW limit (TN-68 TS 2.1.1.). Therefore, it is not expected that there were any actual thermal related concerns with the fuel or the associated cask components. However, this was contrary to the Functional and Operational limits of TS Section 2.1.1 , Table 2.1.1-1, which requires the assemblies to have been cooled for 10 years. The decay heat of the assemblies has continued to decrease since their initial loading in 2001 and all assemblies currently meet the TS 2.2.1 limits. The fuel assemblies are in a safe condition as required by TS 2.2.1.

"These casks were loaded under TN-68 Certificate of Compliance (C of C) Amendment 0 (Certificate 1027). This notification is required pursuant to TN-68 TS Section 2.2.2. This issue has been entered into the Corrective Action Program.

"The NRC Resident Inspector has been informed."

Don Helton
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Ph: 301 251-7594

From: Helton, Donald
To: Esmaili, Hossein
Subject: RE: SFPSS Sensitivity Analyses
Date: Thursday, February 07, 2013 7:03:22 PM

K. I will look at Steve's email more closely tomorrow morning.

Are you in tomorrow (Friday?) I need to talk to you at some point about SFPSS coverage...

From: Esmaili, Hossein
Sent: Thursday, February 07, 2013 4:43 PM
To: Helton, Donald
Subject: RE: SFPSS Sensitivity Analyses

I am struggling with this a little. Not sure what to do about the leakage – maybe talk to Jose. It could change the steam concentration inside the RB and affect the rate of oxidation.

From: Helton, Donald
Sent: Thursday, February 07, 2013 3:43 PM
To: Esmaili, Hossein
Subject: FW: SFPSS Sensitivity Analyses

FYI – I haven't read it yet to have formed any opinion...

From: Jones, Steve
Sent: Thursday, February 07, 2013 3:26 PM
To: Algama, Don; Helton, Donald
Cc: Casto, Greg
Subject: SFPSS Sensitivity Analyses

Don and Don,

After revisiting the sensitivity analyses provided in Chapter 9, I recommend one more analysis. I am concerned that the hydrogen combustion modeling may be unrealistic because changes in reactor building leakage have not been evaluated.

Section 5.3.1.3 of the PBAPS FSAR states that the insulated metal siding above the refueling floor is installed with sealed joints. While I understand the assumption that the siding remains in place after a large earthquake, I do not understand using nominal (low) reactor building leakage. Page 92 of the SFPSS states:
A single control volume models the refueling bay. An open hatch in the southeast quadrant connects (via a flowpath) the refueling room to a boundary condition volume representing the flow connection to the lower sections of the building. The nominal reactor building leakage is modeled at the center elevation of the refueling bay, and the leakage flow from elevations in the simplified model from the lower regions was tuned to match the leakage flow rate of a detailed reactor building model.

The sealant used between siding panels could credibly separate during the seismic event, particularly near the corners of the building. Figure 82 of the SFPSS indicates that the hydrogen generation occurs over just a 2 hour period when water level is near the baseplate and steam generation is low (i.e., the reactor building is not pressurized). Increased building leakage under these conditions could prevent hydrogen concentrations reaching values supporting combustion. Increased leakage may also enhance the effect of air cooling by reducing building temperature at this stage of the event.

I suggest an additional sensitivity analyses to investigate the effect of changes in reactor building leakage. Separation of the sealant between siding panels could significantly increase leakage and alter the progression of the event in the spent fuel pool. Also, this sensitivity would help assess the effect of hydrogen mitigation vent panels considered for deployment in Japan.

E-57

From: Helton, Donald
To: Casto, Greg
Subject: RE: Emailing: Incoming - 08-10-12 Lochbaum (2).pdf, initial reactions.pdf
Date: Wednesday, January 09, 2013 4:04:00 PM

Good info. Thanks.

-----Original Message-----
From: Casto, Greg
Sent: Wednesday, January 09, 2013 3:42 PM
To: Helton, Donald; Jones, Steve; Esmaili, Hossein
Subject: RE: Emailing: Incoming - 08-10-12 Lochbaum (2).pdf, initial reactions.pdf

Just FYI, but in case it comes up:

I did some homework on the Lochbaum letter and assumptions on General Emergencies for SFP emergency classifications. He mis-sighted, and used the security based EALs (which are true for terrorist attack on an SFP being a GE). However, currently (NEI 99-01, Rev 5), and for Ginna, specific non-security events in an SFP only get you to an ALERT (and this is currently fuel uncovery for ALERT), and it takes offsite dose to drive you to a GE. That is pretty consistent among all plants (based on about a 20% sample and IAW our discussion with NEI). The new (NEI 99-01 Rev 6) EALs will escalate the non-security EALs to align with pool level, generally speaking with: any loss of level = NOUE, significant loss of level (level 2) = ALERT, TOF/level 3 = SAE, and TOF/level 3 > 60 min. = GE.

Also, and contrary to Lochbaum's letter, an ISFSI escalation above NOUE would also proceed similarly for both a security and non-security event (based on dose offsite for GE). I have data to support, if it comes up. Tx greg

-----Original Message-----
From: Helton, Donald
Sent: Wednesday, January 09, 2013 11:26 AM
To: Casto, Greg; Jones, Steve; Esmaili, Hossein
Subject: Emailing: Incoming - 08-10-12 Lochbaum (2).pdf, initial reactions.pdf

August letter from UCS, and my initial reactions...as discussed during our prep meeting for the Chairman briefing...

The message is ready to be sent with the following file or link attachments:

Incoming - 08-10-12 Lochbaum (2).pdf
initial reactions.pdf

Note: To protect against computer viruses, e-mail programs may prevent sending or receiving certain types of file attachments. Check your e-mail security settings to determine how attachments are handled.

E58

From:	Helton, Donald
To:	Nosek, Andrew
Subject:	FW: Official report of The Fukushima Nuclear Accident Independent Investigation Commission (just released)
Date:	Friday, July 06, 2012 4:24:00 PM
Attachments:	NAIIC report lo res2.pdf

FYI – Pg. 19 estimates the amount of land in Japan contaminated above 500 mrem to be 700 square miles.

From: Nicholson, Thomas
Sent: Friday, July 06, 2012 3:43 PM
To: Correia, Richard; Coe, Doug
Cc: Ott, William; Uhle, Jennifer; Noggle, James; Cady, Ralph; Philip, Jacob; Raione, Richard; Helton, Donald; Marksberry, Don; Lee, Richard; Fuhrmann, Mark; Kanney, Joseph; Tiruneh, Nebiyu; McGowan, Anna
Subject: FW: Official report of The Fukushima Nuclear Accident Independent Investigation Commission (just released)

Richard and Doug:

I just received from Fernando Ferrante, NRR, the following executive summary report authored by the Fukushima Nuclear Accident Independent Investigation Commission of the National Diet of Japan.

Thanks .... Tom

From: Ferrante, Fernando
Sent: Friday, July 06, 2012 12:43 PM
To: Bensi, Michelle; Caverly, Jill; Cook, Christopher; Kanney, Joseph; Miller, Ed; Nicholson, Thomas; Patterson, Malcolm; Pohida, Marie; Sancaktar, Selim; Thompson, Jenise; Uribe, Juan; Pohida, Marie
Subject: Official report of The Fukushima Nuclear Accident Independent Investigation Commission (just released)

All,

This report is worth reading.

Thanks,

Fernando

E59

From: Helton, Donald
To: Taylor, Robert
Cc: Norton, Charles
Bcc: Marksberry, Don
Subject: 1f4 sfp
Date: Monday, July 09, 2012 2:52:00 PM

Rob,

Below is some input that Chuck Norton and I put together at the end of May to answer a question from the US embassy in Japan. It may be helpful for your purpose.

I've cc'd Chuck just in case he is aware of any more recent response formulation on this front (precluding the possibility of a zirc fire for 1F4 sfp).

Don

The short answer is, a release of radioactive material from the Unit 4 pool is unlikely, and becomes less and less likely as the decay heat of the fuel continues to decrease. We will never be in a position to say that there is no risk.

To elaborate, there are a number of scenarios that are worth covering individually when positing what might happen to an SFP with older fuel (here we are using the term "old" to describe fuel that was discharged from the reactor for a long time, in this case > 17 months). These are:

Scenario 1 - Total loss of fuel pool cooling and injection (e.g., loss of power along with site abandonment):

In this scenario the closed loop cooling system is lost and the ability to make up water is lost with no other damage assumed.

The time for the water in the fuel pool to become saturated and boil down to a level where there is a possibility of fuel damage leading to a release (resulting from extended exposure and heatup of the fuel due to inadequate cooling) is on the order of weeks for a pool with the relatively small (and old) inventory of 1F4. In other words, there would be weeks available for actions to restore cooling or injection. Relatively modest flow rates (10s of gallons per minute) would be sufficient to prevent damage to the fuel during this time.

Scenario 2 - Very large seismic event that leads to a complete leakage of all water from the SFP.

In this scenario, for 1F4 (where the total inventory is low, the fuel has been discharged for > 17 months, and the racks are not that tightly spaced), there is a high likelihood that the fuel would be passively cooled by natural circulation air flow. This situation would lead to no radioactive release, though it could complicate recovery actions because the dose rates in the reactor building would be very high from neutron and gamma "shine."

Scenario 3 - Very large seismic event causing a leak in the SFP liner above the bottom of the fuel.

E60

In this scenario there is the possibility of fuel damage leading to a release due to insufficient water inventory to provide adequate cooling and the water level being such that air flow under the racks is blocked (thus resulting in greatly reduced air cooling. That being said, this would require a tear of the pool's liner at specific elevations, along with unsuccessful deployment of mitigation strategies (e.g., additional water injection). The heatup of the fuel would likely be very slow owing to the low decay heat of the (older) fuel. It is possible that the fuel physically could not heat up enough to undergo a radioactive release for fuel this old, in this condition, but we don't have sufficient analysis to state that dispositively.

Scenario 4 - Very large seismic event that drains the pool and causes direct mechanical damage to the fuel and racks

In this scenario there is a possibility of losing the air coolable geometry. There could also be a release directly from the damaged fuel rods, known as a gap release. An event of this magnitude is even less likely than those described above.

Don Helton
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Ph: 301 251-7594

From: Helton, Donald
To: Gibson, Kathy; Wagner, Katie; Lee, Richard
Cc: Correia, Richard; Scott, Michael; Weaver, Doug; Coyne, Kevin; Nosek, Andrew; Murphy, Andrew; Pires, Jose; Esmaili, Hossein
Subject: RE: SRM on spent fuel
Date: Sunday, July 15, 2012 3:00:00 PM
Attachments: perspective on SFPSS results.pptx

Kathy,

The attached does what you request. Other SFPSS team members can handle the Commissioner TA brief as well as I can, though I'll support if I'm available. Note that I will be at a doctor's appointment tomorrow (Monday) morning.

For the remaining 4-6 weeks before I depart on paternity leave I will be focusing more and more exclusively on my other projects (Level 3 PRA, TH success criteria, ICM, LTRP Level 2, ANS Level 2 Standard).

Don

From: Gibson, Kathy
Sent: Saturday, July 14, 2012 2:00 PM
To: Helton, Donald; Wagner, Katie; Lee, Richard
Cc: Correia, Richard; Scott, Michael; Weaver, Doug
Subject: Fw: SRM on spent fuel

Don,
Brian wants a meeting with the TAs Monday to get clarity on both sides about what we've done and what the Comm thinks they want. This may come up quickly on Monday so I suggest you pull the slides out of the package that show the historical timeline and the pie charts that show we reviewed past studies to conclude that the seismic event was a good place to start. Then add one slide that summarizes the section of the report where you compare to the previous studies on seismic impacts on casks. And then add the one table from the slide package you recently sent me that compares the SFPSS results to the results from past studies (and tell them some version of this will be in the report). I think if we walk them through this it will give them a good perspective on the comparisons we have and can do. Then hopefully they can give us some perspective about what more the Comm wants. Brian feels that even though we covered some of this in the Commissioner briefings, it was apparent from his periodic with Comm A that it didn't sink in. They are focusing solely on the "high consequences" - AJs results charts. So we have a real risk communication challenge here. I think they are looking for the comparisons to provide some perspective.

An open issue is any previous studies or plans for a new study on risk of fuel and cask movements. Hopefully SFST can shed light on that.

If you have better ideas or other information that you think should be covered in the meeting, feel free to adjust to what I proposed.

Thanks,
Kathy

From: Gibson, Kathy
To: Merzke, Daniel; Chen, Yen-Ju

E-61

Cc: Scott, Michael; Weaver, Doug
Sent: Fri Jul 13 16:50:38 2012
Subject: SRM on spent fuel

Yen/Dan,

As a result of a conversation Brian had with Comm. Apostolakis at his periodic today, Brian would like to request a meeting with the TAs on Monday to discuss with them the SRM, explain what the staff has done already regarding SFP studies and comparisons, and get their insights on what and why the Commission means by:

"The Office of Nuclear Regulatory Research should conduct a comparative assessment of SFPSS results against previous studies of safety consequences associated with ~~expedited~~ **[AMM change]** loading, transfer, and long-term dry storage. These previous studies should be updated as necessary to conduct the comparative assessment. "

We would ideally like to have this meeting before the SRM is issued.

Sorry for the late notice, but Brian was in WF all morning and I had meetings this afternoon so we were just now able to connect.

Please let me know what is within the realm of possibility.

Thanks,
Kathy

PS I am scheduled to be in training Monday and Tuesday but I will be checking email and will attend whatever meetings can be arranged.

Kathy Halvey Gibson
Director
Division of Systems Analysis

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(301) 251-7499 Work
(240) 688-6556 Cell

US Nuclear Regulatory Commission
Office of Nuclear Regulatory Research
Protecting People and the Environment

From:	Helton, Donald
To:	Algama, Don
Cc:	Esmaili, Hossein; Wagner, Katie; Wagner, Brian; Nosek, Andrew; Gonzalez, Felix
Subject:	FW: Potential for Re-Criticality
Date:	Wednesday, August 01, 2012 10:37:00 AM
Attachments:	V6229 LetterReport Task 5 draft Rev01.pdf

Don A. – Thanks. The only comments I have are:

- Will this be made publicly available, or can we get buy-in from ORNL that it can be made publicly available by inclusion in another document? For both SFPSS and the Vogtle Level 3 PRA, I think we would want to point to this in a public document.
- For the lead-in to Section 3, ORNL should consider referencing NUREG-1738 (Section 3.6 and Appendix 3), which conclude that, "the staff believes that qualitative risk insights demonstrate conclusively that **SFP** criticality poses no meaningful risk to the public.
- Consider sending the document to Kent Wood (NRR) for his review.
- When the document is finalized, consider sending it to Bob Beall (NRR) as an FYI, relative to Japan Lessons Learned Recommendation 8 (which involves integration and potential update of the EOPs/SAMGs/EDMGs, which in at least the PWROG case is prompting them to consider adding SFP guidance to the SAMGs).

Brian / AJ / Felix – FYI, see attached. Even though this is in the context of accident management as opposed to accident analysis, it's timely relative to our discussion of yesterday. Please do not distribute further since this is a draft. The way I've tentatively treated this issue in the reactor Level 2 and SFP TAPs is to say that inadvertent criticality isn't explicitly considered, but rather, any specific simulations that lead to combinations of conditions where inadvertent criticality would be more likely to occur would be highlighted for potential future analysis.

From: Algama, Don
Sent: Tuesday, July 31, 2012 1:45 PM
To: Helton, Donald; Esmaili, Hossein; Wagner, Katie
Subject: Potential for Re-Criticality

Team:

This report discusses the issues associated with potential re-criticality in the reactor and the sfp during severe accident progression. This report is a white paper designed to obtain high-level expert opinion on the subject. There were no calculations performed.

I would appreciate any comments by the end of the week if possible (it is short).

Thanks,
Don

E62

From: Helton, Donald
To: Voglewede, John
Cc: Velazquez-Lozada, Alexander; Zigh, Ghani; Esmaili, Hossein; Santiago, Patricia
Subject: RE: Spent Fuel Pool Program
Date: Tuesday, August 14, 2012 11:37:00 AM

John –

I provide some suggestions in red below. I'm clearly aware of the experimental work, but not nearly as knowledgeable about it as Ghani et al. I'm making comments from the broader perspective of SFP safety/security, and it's possible that the statements I'm reacting to make sense in the narrower context even if they (arguably) don't make sense in the broader context.

Bottom line – just use whatever is useful and makes sense...

Don

From: Voglewede, John
Sent: Tuesday, August 14, 2012 11:02 AM
To: Helton, Donald
Cc: Velazquez-Lozada, Alexander; Zigh, Ghani; Esmaili, Hossein; Santiago, Patricia
Subject: RE: Spent Fuel Pool Program

Don,

Thank you for the comments. Your response was one of the most constructive (and instructive) set of remarks that I have received on anything in a long time.

In developing this presentation package, I borrowed heavily from the work of others. Most of your comments apply to a Sandia presentation entitled "Investigations of Zirconium Fires during Spent Fuel Pool LOCA's" by Sam Durbin and Eric Lindgren. The presentation was made available as ML120380359 and has been made public for six months (an important point for me).

I have annotated your message below with additional remarks.

John

From: Helton, Donald
Sent: Tuesday, August 14, 2012 8:24 AM
To: Voglewede, John
Cc: Velazquez-Lozada, Alexander; Zigh, Ghani; Esmaili, Hossein
Subject: RE: Spent Fuel Pool Program

John,

The slides you circulated yesterday do not appear to have any OUO information in them from the perspective of the SFP Security Assessments. While that is the aspect you were hoping I'd focus on, I do have some technical/editorial comments below, for your

E63

consideration.

Don

- Slide 3 – "cask" rather than "cast"
  Okay
- Slide 3 - "Air ingress during late stages of core melt-down" – that seems like a stretch to me, but if someone has convinced themselves of this, okay
  Would you be okay if I simply deleted this line? – That would be fine with me; others (in DSA) may feel differently. My concern is simply that anything that looks like the SFP experimental program won't be remotely prototypic for an in-vessel shutdown accident (in terms of geometry, decay heat, flow patterns, etc.). Whether some of the separate effects information (mixed air/steam environment oxidation kinetics) is transferrable is better answered by the phenomenologist / MELCOR developers. It just struck me as a pretty abstract claim, but it could be that someone's done the thinking to back it up.
- Slide 3 – There is no safety (Chapter 15) analysis that requires an SFP whole pool source term analysis – I think it would be better to say "...risk and consequence analyses"
  Okay
- Slide 6 – "O_2" rather than "O^2"
  Okay
- Slide 10 – "Low-density racking least vulnerable" – While generally true, I think you are using the term low-density racking out-of-context (this term is used to refer to a type of rack design with a much larger cell-to-cell pitch than is currently used at essentially any US plant). Also, the term vulnerable is a little inflammatory in this setting. Suggest: "Alternative configurations that spread out recently-discharged fuel can utilize empty cells or "cold" assemblies to slow down heatup during accident conditions"
  Okay. Your suggested replacement sentence is good.
- Slide 11 – "Lowest powered assembly in study potentially more vulnerable" This makes it sound like for a given accident, 10 year old fuel could be more vulnerable than 10 day old fuel in the same SFP, which wasn't the point. The point was that, unlike the complete LOCA case, the highest-decay power fuel isn't always the most limiting. 100 day old fuel might ignite at a higher water level than it would have when it was 30 days old, because of less steam cooling. I think the point you are really trying to make is that the accident progression can be fundamentally different for partial LOCAs, so I suggest saying "Fuel heatup characteristics/phenomena differ from complete LOCAs," and replace the existing sub-bullet with one that says: "Mixed air/steam oxidation environment"
  Okay
- No suggested change to the final set of bullets, but be careful. I think I understand the point you are trying to make, but there are compensating factors here. The air (complete LOCA) oxidation is more energetic than steam, and the leak location has to be in a specific place to get you in this situation. What is actually just as likely to get you in this situation is a small, complete LOCA where the fuel is heating up as it is slowly uncovered.
  I don't understand this completely. I would think oxygen starvation would impact the kinetics. Would you consider

MELCOR calculations performed for uniform different pool loading patterns
Fuel heatup characteristics/phenomena differ from complete LOCA
Mixed air/steam oxidation environment
No robust natural circulation flow pattern
Benefits of steam cooling until fuel becomes significantly uncovered (e.g., fuel mid-height)
Greater likelihood for producing hydrogen
~~Air oxidation more energetic than steam~~
~~Scenario dependent~~Could address behavior of both:
~~Rapid complete LOCA~~
Slowly developing complete LOCAs (i.e., fuel heatup before water level clears rack baseplate)
Partial LOCAs

From: Voglewede, John
Sent: Monday, August 13, 2012 4:12 PM
To: Zigh, Ghani
Cc: Velazquez-Lozada, Alexander; Santiago, Patricia; Helton, Donald; Lee, Richard; Gibson, Kathy; Scott, Michael
Subject: Spent Fuel Pool Program

Ghani,

Once again, I have been asked to brief the OECD/NEA Working Group on Fuel Safety on the subject of the OECD/NEA Spent Fuel Pool Project at Sandia.

Attached is a draft presentation, which is similar to the one I presented previously (and you reviewed). Please review this one as well. I expect to tell the Working Group that your next meeting is in Albuquerque in October, where details will be discussed.

I tried to limit the number of slides to a dozen. Please note that there are two slides about the future: pool loading configuration recommendations and further experimental work on partial LOCA. Both slides cite the (public version) of the Wagner and Gauntt 2006 report.

Also note that I did not discuss recent MELCOR versus experimental differences, which are now a matter of public record.

John

From:	Helton, Donald
To:	Easton, Earl; Witt, Kevin; Jones, Steve
Cc:	Wagner, Katie
Subject:	FW: German SFP Work
Date:	Friday, September 21, 2012 10:14:00 AM
Attachments:	2.02 MCAP 2012 GRS Activities 2012 09 14.pdf 2.05 Modeling BWR SFP Accidents Loeffler GmbH.pdf

FYI – Recent presentations from GRS and Areva-GmbH (Germany) about SFP modeling. We're also having a telecom with the Swiss regulator in November (under an existing cooperative agreement) regarding SFP PRA. I'm sending you this in the context of the SRM that requested the staff consider what folks in other countries are doing about expedited fuel movement. If someone can let me know who has the action on that SRM item I'll keep that person in mind.

The attached aren't all that helpful with respect to the big picture, but they do show that other countries are doing similar work as SFPSS. I also found it interesting that the German PWR studied has their SFP inside containment (I'm not very familiar with European SFP characteristics that differ from US designs). As we've previously discussed, I expect that many countries will have fundamentally different situations with respect to accrued SFP inventories due to reprocessing. There are no surprises in the attached (in my opinion) in terms of large differences in insights between their work and ours, for the instances where the work is comparable.

On an unrelated note, we're starting to find out a little more about Vogtle's SFP situation as we embark on the Level 3 PRA project. Interestingly, they are very asymmetric in their spent fuel management practices, with a preference for storing fuel (even from Unit 1) in the Unit 2 pool. Despite the near-symmetry of the reactors, the SFPs are very asymmetric in rack density, poison material, and loading. They are planning to do their first ever cask load next year.

Don

From: Helton, Donald
Sent: Friday, September 21, 2012 9:45 AM
To: Wagner, Katie
Cc: Esmaili, Hossein
Subject: German SFP Work

Katie,

I lost the thread on the draft SRM from the meeting between the Commission and ACRS, which at one point directed the staff to look at what other countries are doing with respect to expedited fuel transfer. Do you know if that SRM was ever issued with that language, and if so, who the action was assigned to?

I mention it because the attached GRS MCAP presentation would be of interest. It describes their activities for SFP modeling using MELCOR. There was another German presentation by Areva that also focused on SFP modeling. An interesting tidbit from the attached is that there PWR SFP (at least for the plant they are studying) is inside

E64

From: Helton, Donald
To: Esmaili, Hossein; Wagner, Katie; Pires, Jose; Nosek, Andrew; Wagner, Brian; Chang, James; Nosek, Andrew
Subject: FW: Debris in 1F3 SFP
Date: Tuesday, October 02, 2012 9:48:00 AM

Just an FYI in case somebody mentions something about it...see below.

This incident is also covered in the IAEA status that some of you received from Don M. this morning...

From: Marksberry, Don
Sent: Tuesday, October 02, 2012 8:09 AM
To: Helton, Donald
Subject: Debris in 1F3 SFP

At around 11:07 AM on September 22, during debris removal from the upper part of Unit 3 Reactor Building, a steel beam (Approx. 300mmX200mmX7m, Approx. 470kg) which had been on the side of the spent fuel pool (SFP) slipped and fell into the pool when a worker was trying to grab it using an oil-pressure fork attached to the head of the crane. At around 11:45 AM on the same day, it was confirmed that there was no problem with the operation status of the spent fuel pool alternative cooling system and the skimmer surge water level. No significant change was found in the monitoring post data, the atmosphere dose rate around the spent fuel pool and the spent fuel water level. No injury was reported due to this matter. From 9:55 AM on September 24, we started investigating the condition (and location) of the steel beam which fell into the pool and the condition inside the pool using a remote control underwater camera. The investigation was finished at 1:55 PM on the same day. From around 7:00 AM on September 25, we started investigating the condition (and location) of the steel beam which fell into the pool and the condition inside the pool using a remote control underwater camera. The investigation was finished at 11:10 AM on the same day. From 7:05 AM on September 26, we started investigating the condition (and location) of the steel beam which fell into the pool and the condition inside the pool using a remote control underwater camera. The investigation was finished at 10:08 AM on the same day. As a result of investigation, the steel beam found on the upper part of the fuel storage rack located in the southeast side of the spent fuel pool was assumed to be the one which fell into the pool based on its length and shape. The steel beam is currently on the debris in the pool and no problem was found with the fuel assembly, fuel storage rack and the pool liner due to this incident according to the information obtained during the investigation. We will continue to investigate the cause of the incident and discuss recurrence prevention measures.

Sep 26, 2012
Investigation of the Inside of Unit 3 Spent Fuel Pool Using an Underwater Camera at Fukushima Daiichi Nuclear Power Station (September 26) (PDF 28.9KB)

Sep 25, 2012
Investigation of the Inside of Unit 3 Spent Fuel Pool Using an Underwater Camera at Fukushima Daiichi Nuclear Power Station (PDF 27.9KB)

Sep 24, 2012
A Steel Beam Fell into the Spent Fuel Pool in Unit 3 Reactor Building at Fukushima Daiichi Nuclear Power Station (PDF 82.9KB)

E65

From: Helton, Donald
To: Correia, Richard
Cc: Coe, Doug; Coyne, Kevin
Subject: RE: FYI Release of formerly restricted product: GAO-12-797, Spent Nuclear Fuel: Accumulating Quantities at Commercial Reactors Present Storage and Other Challenges, 361313
Date: Thursday, October 04, 2012 1:41:00 PM
Attachments: RE Action GAO-12-797 Spent Nuclear Fuel Accumulating Quantities at Commercial Reactors Present Storage and Other Challenges.msg

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Rich,

We (RES – primarily me on DSA's behalf) provided input to GAO throughout the study, including the information gathering, factual review, and agency comment stages. Staff affiliated with SFPSS/ Tier 3 AR#5 are familiar with the study. GAO basically agrees that it's a complex issue with lots of moving parts. In the end, there only hard conclusion was that the agency needed to do a better job of tracking classified studies, which stemmed from frustration on GAO's part about NRC's ability to produce some classified studies referenced in the information gathering interviews. The difficulties stemmed from a combination of mis-coordination, mis-communication, and some genuine information control issues. Now that the report is on the streets, NMSS/DSFST has a ticket to deal with this finding, and they are appropriately coordinating with RES/DE (as the custodians of the classified room). The attached email thread demonstrates this coordination.

Hope that helps,
Don

From: Correia, Richard
Sent: Thursday, October 04, 2012 11:56 AM
To: Helton, Donald
Cc: Coe, Doug; Coyne, Kevin
Subject: FW: FYI Release of formerly restricted product: GAO-12-797, Spent Nuclear Fuel: Accumulating Quantities at Commercial Reactors Present Storage and Other Challenges, 361313

Don,

Are you familiar with this GAO report on SF? My bottom line read of it looks like GAO wants access to classified studies.

Rich

Richard Correia, PE
Director, Division of Risk Analysis
Office of Nuclear Regulatory Research
US NRC

richard.correia@nrc.gov

From: Uhle, Jennifer
Sent: Monday, October 01, 2012 11:03 PM
To: Correia, Richard; Sheron, Brian; Case, Michael; Richards, Stuart; Gibson, Kathy; Scott, Michael; Grancorvitz, Teresa

E66.

Subject: Re: FYI Release of formerly restricted product: GAO-12-797, Spent Nuclear Fuel: Accumulating Quantities at Commercial Reactors Present Storage and Other Challenges, 361313

Thx rich. Do you know who is going to review this to see if anything pertinent to sfpss is included? J

From: Correia, Richard
To: Sheron, Brian; Uhle, Jennifer; Case, Michael; Richards, Stuart; Gibson, Kathy; Scott, Michael; Grancorvitz, Teresa
Sent: Mon Oct 01 12:51:25 2012
Subject: FW: FYI Release of formerly restricted product: GAO-12-797, Spent Nuclear Fuel: Accumulating Quantities at Commercial Reactors Present Storage and Other Challenges, 361313

FYI...not sure if you have seen this report.

Richard Correia, PE
Director, Division of Risk Analysis
Office of Nuclear Regulatory Research
US NRC

richard.correia@nrc.gov

From: Arildsen, Jesse
Sent: Friday, September 14, 2012 4:14 PM
To: Helton, Donald; White, Bernard; Brochman, Phil; Wastler, Sandra; Westreich, Barry; Gibson, Kathy; Correia, Richard; Bowman, Eric; Easton, Earl; Jones, Steve; Jordan, Natreon; Weaver, Doug; Benney, Brian; Rubenstone, James; Munday, Joel; Carrion, Robert; Caniano, Roy; Everett, Vincent; Brookhart, Lee; Lorson, Raymond; Boland, Anne; Lipa, Christine; Young, Mitzi; Forsyth, Daniel; Poole, Brooke; StAmour, Norman; Ruland, William; Lui, Christiana
Cc: Chen, Yen-Ju
Subject: FYI Release of formerly restricted product: GAO-12-797, Spent Nuclear Fuel: Accumulating Quantities at Commercial Reactors Present Storage and Other Challenges, 361313

FYI

Please see the link below.

GAO-12-797

Spent Nuclear Fuel: Accumulating Quantities at Commercial Reactors Present Storage and Other Challenges

http://www.gao.gov/prerelease/7MLY

Best Regards,
Jesse

From: Helton, Donald
To: Gonzalez, Felix; Wagner, Brian; Esmaili, Hossein; Pires, Jose; Cooper, Susan; Jones, Steve; Wood, Kent; Wagner, Katie; Witt, Kevin; Algama, Don; Compton, Keith
Subject: SFP Level 1/2 Slides for Vogtle Level 3 TAG/ACRS Meetings
Date: Monday, October 22, 2012 5:05:00 PM
Attachments: ACRS - Dec 2012 - SFP L12.pptx

All,

Please find attached a brief set of slides on the SFP Level 1/2 PRA portions of the Vogtle Site Level 3 PRA project. These slides will be folded in to a larger project presentation to a Technical Advisory Group and ACRS Subcommittee, and a higher-level set of slides will also be used for an upcoming public meeting.

Please take a minute (it's only 6 slides) to look at the attached. Please let me know if you have any concerns by COB Thursday (10/25).

Hope all is well,
Don

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E67

From: Helton, Donald
To: Kuritzky, Alan
Cc: Wagner, Brian
Subject: VL3 Slides - SFP and Level 2 - TAG & ACRS
Date: Friday, October 26, 2012 11:01:00 AM
Attachments: ACRS - Dec 2012 - Level 2.pptx
ACRS - Dec 2012 - SFP L12.pptx

Alan,

Per your request, please find attached 2 sets of viewgraphs, one for reactor Level 2 PRA and one for SFP Level 1 / 2 PRA. These slides are for your use in preparing the overall project presentations to the ACRS and TAG, and can also be used for the public meeting. If you'd like me to condense the slides for the public meeting, I'd be happy to do so. These slides incorporate all comments received by relevant staff.

I do not believe that these slides contain any information that shouldn't be made publicly available. The closest is probably the mention that Unit 2's SFP has more fuel than Unit 1, which can't be deduced unless you know that they move Unit 1 fuel to the Unit 2 SFP, but may very well be contained in public submittals like the license renewal application. That said, sending a courtesy copy to the licensee and NRR may not be a bad idea, if feasible.

Best,
Don

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E 68

From: Helton, Donald
To: Wagner, Brian; Nosek, Andrew
Subject: past SFP NUREGs
Date: Monday, November 05, 2012 12:52:00 PM

I tried to quickly confirm my supposition that the high release fractions from past SFP studies were from the fuel rather than to the environment. In a nutshell, I couldn't quickly confirm that to be the case for either 1353 or 1738. It looks like they do not consider the hydrogen deflagration (as I recalled) but that they may have sometime, or always, assumed no building holdup based on a disregard for the phenomena...

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E69

From: Helton, Donald
To: Sullivan, Randy
Cc: Schaperow, Jason
Bcc: Esmaili, Hossein; Nosek, Andrew
Subject: RE: EPRI TBR Reports and SFPs
Date: Thursday, November 08, 2012 12:40:00 PM

Responses in red.

-----Original Message-----
From: Sullivan, Randy
Sent: Thursday, November 08, 2012 8:01 AM
To: Helton, Donald
Cc: Schaperow, Jason
Subject: RE: EPRI TBR Reports and SFPs

Thanks, interesting

Fuel rack channel melt precedes clad damage? I did not know that, seems odd

>> This refers to local melting of some of the constituent materials in the racks. Industry's slides cited melting temperatures such as 660 C for the Aluminum component of Boral, and lower temperatures for the onset of damage for Boraflex. Clad rupture is generally predicted to occur at 900C in MELCOR (following localized rod ballooning that is anticipated in the 700 – 850C range). Consideration of eutectics, along with the large axial variation in the accident temperature profile, makes the above comparison too simplistic, which is why I caution against using this lower metric.

I would like to discuss with NEI or INPO or some industry group, as to who is this guy? So if the water level is low, what happens? They go home?

>> Bob Henry is not affiliated with either NEI or INPO, and if I interpret your tone correctly, you are right to think that the regulatory answer you would receive from either organization might be different. He, of course, also made the statements in the context in which he was speaking; that being general guidance to industry for development of accident management programs. His presentation was on behalf of EPRI. As for who he is, he's an individual who has worked with the MAAP computer code development and domestic/international accident management program development since their inception. He was also a peer reviewer for SOARCA owing to his recognized expertise. I'm just offering it as one of many perspectives. I'm sure if he were questioned directly, he'd offer a lot of other considerations as well, some favorable to your position and some not.

Yes access would be restricted, and difficult, SCBA is used at NPPs... and so is occasional high dose rate work

>> Absolutely, though under controlled conditions.

We are talking about running a hose up a flight of stairs and attaching it to a pre-staged rig, much more complicated work is performed.

>> Again, I agree, but point out the difference in controlled versus uncontrolled conditions.

E70

I don't think that doing eddy current testing in a steam generator inlet plenum during an outage can be equated to the situation under discussion.

Randolph Sullivan, CHP

-----Original Message-----
From: Helton, Donald
Sent: Thursday, November 08, 2012 7:17 AM
To: Chang, James; Mitman, Jeffrey; Esmaili, Hossein; Nosek, Andrew; Bowman, Eric; Jones, Steve; Wood, Kent; Sullivan, Randy
Cc: Wagner, Katie; Pires, Jose; Murphy, Andrew; Wagner, Brian
Subject: RE: EPRI TBR Reports and SFPs

Not yet. They will be available as part of the meeting summary. You may be able to get them directly sooner from the NRR PM, Bob Beall.

From: Chang, James
Sent: Wednesday, November 07, 2012 7:11 PM
To: Helton, Donald; Mitman, Jeffrey; Esmaili, Hossein; Nosek, Andrew; Bowman, Eric; Jones, Steve; Wood, Kent; Sullivan, Randy
Cc: Wagner, Katie; Pires, Jose; Murphy, Andrew; Wagner, Brian
Subject: RE: EPRI TBR Reports and SFPs

Don,
Thansk for the information. Do you know if the industry's slides are available?

James

From: Helton, Donald
Sent: Wednesday, November 07, 2012 3:50 PM
To: Mitman, Jeffrey; Chang, James; Esmaili, Hossein; Nosek, Andrew; Bowman, Eric; Jones, Steve; Wood, Kent; Sullivan, Randy
Cc: Wagner, Katie; Pires, Jose; Murphy, Andrew; Wagner, Brian
Subject: EPRI TBR Reports and SFPs

All,

There was a public meeting today on Recommendation 8 (Integration of EOPs, SAMGs, ..) which had a couple of interesting points. During the meeting, industry presented slides on their efforts to update 2 documents from 1992 known as the Technical Basis Reports (TBRs) for accident management. The revised versions, which relate to an industry initiative to update the generic Owners Group SAMGs, will be available for download from the EPRI website in "a few days."

The updated TBRs have a new appendix related to SFP accident management that did not exist before. There are two points that I want to bring to your attention. Quoted items are from the slides at today's meeting, and for context, the slides I quote are those of Bob Henry (Fauske and Associates), a widely acknowledged reactor accident management technical expert.

1. ""Personnel Access Limits:

a. Boiling of the SFP water will likely generate harsh conditions...that would challenge personnel safety. Access may require a special suit and a breathing apparatus.

b. ...Access should be limited when the level is significantly reduced and prohibited when the water level is below one-half of the nominal value." [I confirmed with Bob that I was correctly interpreting this latter value to be a level that corresponded to a SFP water level several feet (e.g., 5) above the top of the racks.]

You may recall that we've equated a water level roughly 2 feet above the racks as the point where significant concern begins. The above statement seems to be in conflict, in some respects, with the licensee commitments under 50.54(hh)(2), though I'm not the expert on that. Nevertheless, the TBRs are not a regulatory document, so the licensee commitments trump it.

2. In developing timing estimates for operator actions, they elected to define fuel failure as the time at which the fuel reaches the melting temperature of the racks' constituent absorber material, with the thought that this poses a coolable geometry/structural challenge to the fuel. I confirmed that they did not make this choice based on criticality concerns. Such a time would precede gap release. I personally don't believe that this is a suitable (or necessary) surrogate for our purposes, as these temperature would be reached first at the top of the racks where they provide no structural .purpose and where they pose less of a concern for blocking flow. I speculate they made this assumption to simplify the analytical calculations they were using.

I just wanted to bring these items to your attention since they offer yet another perspective on some of the issues we've been debating.

Best,
Don

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From: Helton, Donald
To: Murphy, Andrew
Subject: Swiss slides
Date: Tuesday, November 13, 2012 2:32:00 PM
Attachments: NRC-ENSI-Discussion-Nov 2012.pdf

As requested.

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E71

From: Helton, Donald
To: Hogan, Rosemary
Subject: FW: Response to Congress Regarding GAO-12-797, "Spent Nuclear Fuel: Accummulating Quantities at Commercial Reactors Present Storage and Other Challenges"
Date: Wednesday, November 14, 2012 11:33:00 AM
Attachments: 11-13-12 Lieberman GAO-12-797 ltr.pdf

FYI

This is the first activity on this front that's come across my inbox since early October...

From: Arildsen, Jesse
Sent: Tuesday, November 13, 2012 5:12 PM
To: Weber, Michael
Cc: Johnson, Michael; Wiggins, Jim; Haney, Catherine; Sheron, Brian; Stapleton, Bernard; Chen, Yen-Ju; StAmour, Norman; Lui, Christiana; Helton, Donald; White, Bernard; Rubenstone, James; Carrion, Robert; Skeen, David; Lorson, Raymond; Ferdas, Marc; Spitzberg, Blair; Pederson, Cynthia; Boland, Anne; Lipa, Christine; Learn, Matthew; Poole, Brooke; Forsyth, Daniel; Dapas, Marc; Rivers, Joseph
Subject: Response to Congress Regarding GAO-12-797, "Spent Nuclear Fuel: Accummulating Quantities at Commercial Reactors Present Storage and Other Challenges"

Michael,
The Chairman's letter to Congress regarding agency actions to address GAO's recommendation in GAO-12-797 is attached.
Best Regards,
Jesse

E72

From:	Helton, Donald
To:	Pires, Jose
Cc:	Wagner, Brian; Sancaktar, Selim; Stutzke, Martin; Kuritzky, Alan; Murphy, Andrew
Subject:	Seismic Hazard Calculations
Date:	Friday, November 16, 2012 11:37:00 AM
Attachments:	GSI Info wrt Vogtle Seismic Hazard - info as of June 2011.xlsx

Jose,

You asked me for the calculations that I had done a year or so ago on the Peach Bottom seismic hazard, in reference to generating similar information for Vogtle. I went ahead and updated my spreadsheet to do the same calculations for Vogtle (I was curious), and it is attached for your reference. A few notes:

- This is based on information Marty provided me over a year ago, so there very well may be newer information.
- The attached calculations should be verified before being used for anything. I'm the least qualified person on this email to be generating such estimates.
- The Vogtle PGA curve looks odd compared to the other sites I plotted (see 4th tab). Not sure why this is.
- The actual bin estimates are on the 5th tab.
- If the attached manipulations are correct and up to date, then it suggests to me that we should specify that Vogtle is a (relative to other CEUS plants) low-seismicity site for very large seismic events, but not so for events that are a few multiples of the SSE.

You raised a very good question yesterday about whether it is worth building an FEA model for the SFP solely for estimating the seismic fragility, if the large seismic events are going to result in extremely low (truncation level) frequencies. In the SFP study we will need to be philosophically consistent with the reactor external hazards screening analysis, but in my opinion, there should be a significant offset (e.g., an order of magnitude) between the reactor and SFP cutoff (since the SFP has the potential for larger releases from that sort of event). Depending on what discussion this email provokes, I may set up a meeting to discuss this further.

Hope this is helpful,
Don

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E 73

From: Helton, Donald
To: Scott, Michael; Gibson, Kathy; Coe, Doug; Correia, Richard
Cc: Wagner, Katie; Coyne, Kevin; Wagner, Brian; Demoss, Gary
Subject: RE: SFPSS
Date: Thursday, November 29, 2012 9:07:00 AM

>> Mike - In response...I'll use >> b/c I don't know if colors show up on Blackberrys...

Don't know whether you have seen this.

>> Yes, we have. NRR/DSS was assigned the lead for responding to the May 7, 2012 David Lochbaum letter because (despite the title), the letter has very little to do with SFPSS and mostly to do with SFP design-basis regulation. NRR's response was signed out by the Chairman on 9/5/12 (ML12164A825). Note that David Lochbaum also submitted a letter on August 10, 2012 that focused on EP aspects of SFP postulated accidents, which I assume was assigned to NSIR (though I've not seen any email traffic about it since mid-August).

Point is raised that SFPSS is incomplete because it does not address draindown risks of normal SFP operations. Easy answer is – of course it does not – that was not its intent.

>> SFPSS did not look at draindown accidents because it focused solely on the SFP BDBA scenario that past studies (which did consider draindown events) indicate is most problematic, that being large seismic.

Lochbaum is asserting that the Agency's regulatory framework is not adequate to capture/control such risks, and by implication that we do not adequately understand them.

>> As discussed in the agency response, David Lochbaum mis-characterized a few things in his letter, namely an issue related to SFP level requirements in the Technical Specifications. The agency had thoroughly investigated the issue of inadvertent SFP draindowns as of the mid-90s. I'll leave it to NRR to defend why the level of activity on that front since the mid-90s is adequate.

Which leads me to the question as to how adequately risks of "normal" SFP operation under the existing regulatory framework (considered by Lochbaum to be inadequate) are captured in the planned SFP risk evaluation. Theoretically, pool operation miscues could lead to draindowns.

>> They absolutely can, and there is operating experience to demonstrate that. That said, NRC/AEOD conducted a study in the mid-90s (NUREG-1275, Volume 12) that included a limited-scope SFP PRA by INL to investigate the importance of these issues. NRR then undertook the "SFP Action Plan" in the mid-to-late 90s to address issues that were identified (e.g., the need for manual isolation valves for specific lines in specific plants, etc.) Note that this activity was, in part, motivated by concerns that David Lochbaum raised in the early 90s.

Will these risks be evaluated as part of the Tier 3 item on accelerating spent fuel removal from SFPs? Or do we have a document somewhere that puts that risk so far in the weeds that it need not be considered? Your thoughts?

E74

>> It will be Brian Wagner/Gary's call as to what extent this is considered in Task 2.2 of the SFPSS SRM. For perspective, NUREG-1738 (2001) estimates the frequency of fuel uncovery from inadvertent draindowns to be a factor of 700 less than the frequency of fuel uncovery from a large seismic event (as estimated using the LLNL curves, which most closely approximate the current seismic hazard estimates).

From: Scott, Michael
Sent: Thursday, November 29, 2012 8:40 AM
To: Helton, Donald; Gibson, Kathy; Coe, Doug; Correia, Richard
Subject: SFPSS

Don't know whether you have seen this. Point is raised that SFPSS is incomplete because it does not address draindown risks of normal SFP operations. Easy answer is – of course it does not – that was not its intent.

Lochbaum is asserting that the Agency's regulatory framework is not adequate to capture/control such risks, and by implication that we do not adequately understand them.

Which leads me to the question as to how adequately risks of "normal" SFP operation under the existing regulatory framework (considered by Lochbaum to be inadequate) are captured in the planned SFP risk evaluation. Theoretically, pool operation miscues could lead to draindowns. Will these risks be evaluated as part of the Tier 3 item on accelerating spent fuel removal from SFPs? Or do we have a document somewhere that puts that risk so far in the weeds that it need not be considered? Your thoughts?

From: Helton, Donald
To: Fuller, Edward; Compton, Keith
Subject: FW: L3PRA
Date: Friday, November 30, 2012 12:50:00 PM

FYI - My current perspective is provided below.

-----Original Message-----
From: Helton, Donald
Sent: Friday, November 30, 2012 12:47 PM
To: Sullivan, Randy
Cc: Jones, Joe A
Subject: RE: L3PRA

> Will we have a SFP Zr fire? [For Vogtle]

We don't know, and we won't for a while. There is certainly that potential. I've taken a quick look at the 2008 USGS seismic hazard data for Vogtle and it isn't evident that the very large seismic will screen out, though it might. The other wild card for seismic is the soil/structure interaction and soil liquefaction aspect, since Vogtle is a 'deep soil' site. For non-seismic, cask drops are less of a concern than they would otherwise be because they can only drain the pool down to a certain level (connected but separate cask pit). We simply won't have a good guess until the Level 1 SFP PRA is done, and won't know for sure until the SFP Level 2 PRA is done, neither of which will happen in the next few months.

> Will we have a seismic event that will cause release? maybe they are the same?

For the reactor, almost certainly (akin to the seismically-induced SBOs for both Surry and Peach Bottom). For the SFP, see above.

-----Original Message-----
From: Sullivan, Randy
Sent: Friday, November 30, 2012 12:32 PM
To: Helton, Donald
Cc: Jones, Joe A
Subject: RE: L3PRA

Ok

Well we will have to adapt

Thanks

Basically I need to know:

Will we have a SFP Zr fire?

Will we have a seismic event that will cause release? maybe they are the same?

thanks

Randolph Sullivan, CHP

-----Original Message-----
From: Helton, Donald
Sent: Friday, November 30, 2012 11:06 AM
To: Sullivan, Randy
Cc: Jones, Joe A

E75

Subject: RE: L3PRA

Randy,

I'm the SFP guy, in addition to the reactor Level 2 guy. I'm not sure that I know anymore now then I did when we talked a couple of weeks ago, but always happy to brainstorm.

Unfortunately, I am on travel from Wednesday morning to Friday afternoon next week. I might be able to call in to a meeting while in transit on late Wednesday...might even be to my hotel by 4:30 or 5...

I'd recommend that we include Keith Compton and Ed Fuller in on as many of these types of discussions as possible. They will have strong input to the release categorization when the time comes.

I'm in and out of the office today (sick kid at home), but will be checking email periodically. Let me know how you'd like to proceed.

Best,
Don

From: Sullivan, Randy
Sent: Friday, November 30, 2012 9:20 AM
To: Helton, Donald
Cc: Jones, Joe A
Subject: L3PRA

Don

I was at SNL this week, working on a few things, and we discussed support I need for the L3PRA. As we discussed I can reduce the request if I know what the likely release bins are...

Joe Jones will be here next week and I wonder if we could meet and who is the SFP L3PRA guy again?

And I have training at the PDC which limits the potential meeting times as does the ACRS meeting. Is there any chance of meeting late on Wednesday? Like 430 or 5?

Or we will punt and do it over the phone in a week or two.

Thanks

Randolph Sullivan, CHP

From: Helton, Donald
To: Wagner, Brian
Subject: FW: Seabrook SFP
Date: Friday, November 30, 2012 2:15:00 PM
Attachments: SFP PRA Results.doc

FYI

From: Wood, Jeffery
Sent: Friday, November 30, 2012 2:08 PM
To: Helton, Donald
Subject: Seabrook SFP

Don,

When we went to Seabrook back in 2007, they generously (and informally) gave us a bulk dump of a bunch of their PRA documentation. I had never looked through the SFP folder until now, but it looks like there are a handful of documents and spreadsheets. There is even a folder of CAFTA files if you were interested in digging into those. I wouldn't get too excited though. It doesn't look like there's a ton of detail there. Still it's probably worth taking a look.

Here is a short summary document. If you want to see the other stuff, then come by with a thumb drive.

-Jeff

E76

From:	Helton, Donald
To:	Coyne, Kevin
Subject:	FW: Region III Questions on SFP Study
Date:	Sunday, December 09, 2012 6:48:04 PM

Either Hossein and Jose are being kind, or I should work on Saturday nights more often. The responses below are now ready to go to Julio. Do you want me to send them since you'd be cutting and pasting from a Blackberry?

From: Esmaili, Hossein
Sent: Sunday, December 09, 2012 5:46 PM
To: Helton, Donald; Pires, Jose
Cc: Coyne, Kevin; Wagner, Katie; Algama, Don
Subject: RE: Region III Questions on SFP Study

Don- it looks good. I have nothing to add.

Hossein

From: Helton, Donald
Sent: Saturday, December 08, 2012 10:10 PM
To: Esmaili, Hossein; Pires, Jose
Cc: Coyne, Kevin; Wagner, Katie; Algama, Don
Subject: RE: Region III Questions on SFP Study

Hossein / Jose:

Kevin Coyne gave a presentation this past week in Region 3 that included slides on SFPSS. He got 3 questions that he (quite reasonably) was not able to answer. Can you look at my draft responses below and edit at will (Hossein - #1 & #3 / Jose - #2).

Thanks,
Don

From: Coyne, Kevin
Sent: Friday, December 07, 2012 4:57 PM
To: Helton, Donald
Subject: Region III Questions on SFP Study

Don –

The presentation went well – just a handful of questions I couldn't answer...

1. Does the initiative to add noble chemicals to the fuel cladding (e.g., platinum) affect fission product releases?

The fission product releases associated with a major SFP (or reactor) accident come from the gap region, and the fuel matrix itself. While there are some fission products embedded in the cladding, this is not the major contributor. Thus, changes to the cladding composition being made to reduce fuel failures during normal operation (such as the program described in http://mydocs.epri.com/docs/Portfolio/PDF/2013-Roadmaps-NUC/FRP_04_R2-BWR-Corrosion-and-Crud-Control.pdf), don't have a direct effect on the fission product release models used in MELCOR. They do, at least theoretically, have the potential to affect the course of a severe accident if they change the cladding's response to heatup (either via changes in the cladding's mechanical properties or changes in the rate of the cladding's oxidation kinetics). In general, the various forms of cladding materials used in US reactors have relatively similar mechanical properties and oxidation kinetics (because they only differ

E77

from each other in the elements found in trace amounts). This would suggest that they type of doping mentioned in the cited article wouldn't have a major impact, though we've certainly not taken a hard enough look to say this dispositively.

2. Would existing SFP degradation (e.g., many plants have small existing leaks) affect the study results significantly?

Assuming that the degradation referred to is a leak stemming from a single (or a few) places where a weld has degraded, this could affect the results. These locations would be more likely to "fail" if they are in regions of high stress/strain. However, if they are localized to begin with, then there contribution to the overall leak rate for the "small" or "moderate" leak situations could be relatively minor. Similarly, the degradation could shift a "no leak" situation to one where minor additional leakage (from the degraded weld) could be expected. Even so, the change in results from the degradation may very well be subsumed in the uncertainty acknowledged by using the 3 leak-size approach.

3. Is the standby gas treatment system credited (or would it help)? This one I'm assuming it's not credited due to the loss of AC power, but if available, it would help. Do we have any guesstimate as to how much it would help (a lot, not much, hard to tell...)?

SGTS is not credited due to the lack of AC power. Whether it would help is a complex question, best answered by 'it depends'. For example, good ventilation (which SGTS would provide) can be beneficial early in a complete draindown accident for preventing a zirconium fire – by facilitating natural circulation of air once the pool is empty). If a zirconium fire does occur, SGTS could have a positive or negative effect on hydrogen combustion (positive in that it could prevent stratification; negative in that it could serve to condense steam and de-inert the air space). SGTS can also have a positive or negative effect on fission product releases offsite, depending on whether the plant in question has Iodine filters and how the system is designed.

No rush, but if there is any insights to pass on, I'll get them back to Julio Lara...

Thanks!

Kevin

From: Helton, Donald
To: Chang, James; Murphy, Andrew; Pires, Jose
Cc: Wagner, Katie; Mitman, Jeffrey; Zoulis, Antonios; Cahill, Christopher
Subject: RE: SFPSS
Date: Thursday, December 13, 2012 2:31:00 PM

James,

In an attempt to save Jose a few minutes since I know he's swamped...the short answer to your question is yes, we do expect it to survive a 0.5-1g event. This has to do with seismic margin above the design-basis conditions. Section 4.2 (Other Damage States) of the SFPSS report, subsection entitled "Damage to the Reactor Building and Other Relevant SSCs," provides more discussion. The salient paragraph is pasted below.

Don

According to the fragility analysis for the NUREG-1150 seismic PRA (Lambright et al., 1990), the median fragility for the reactor building is about 1.6g. The response of the reactor building structure is expected to be more sensitive to the horizontal ground motions than to the vertical ground motions. Natural frequencies of vibration for horizontal modes of vibration of the reactor building are about 7 Hz (i.e., frequencies at which the spectral accelerations of the ground motion for the scenario considered are less than those for the ground motions with the same PGA considered in earlier evaluations of the median fragility). On these bases, failure or severe damage to the reactor building would not be expected for the seismic scenario considered.

From: Chang, James
Sent: Thursday, December 13, 2012 2:09 PM
To: Murphy, Andrew; Pires, Jose
Cc: Wagner, Katie; Helton, Donald; Mitman, Jeffrey; Zoulis, Antonios; Cahill, Christopher
Subject: SFPSS

Andy and Jose,

A quick question regarding the magnitude and effects of the earthquake magnitude in SFPSS. The earthquake in SFPSS is 0.5 – 1.0 g. The Peach Bottom's final safety analysis report states that for the reactor building (a category I structure):

"A dynamic analysis of the reactor building was conducted for the design earthquake with 0.05g horizontal ground acceleration and for the maximum credible earthquake with 0.12g horizontal ground acceleration."

From the number it seems that we are talking a earthquake 10 time different in magnitude between the Category I design and the SFPSS hypothetical earthquake. Is the Category I structure likely to survive the 0.5 – 1.0 g earthquake?

james

Yung Hsien J. Chang, Ph.D.

E78

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From: Helton, Donald
To: Scott, Harold
Cc: Esmaili, Hossein
Subject: RE: Fuel Cladding / Oxidation FAQs
Date: Wednesday, December 19, 2012 10:59:00 AM

Harold,

Dana Powers brought up the decrepitation issue about 8 years ago in the context of the SFP security assessments. Charlie never felt that it was worth engaging on. I believe the thought had to do with high burnup spent fuel that has seen high termperatures during a postulated accident, and whether it will behave differently (e.g., break up) due to material property changes. The term is mentioned by the ACRS as something the staff has looked in to, in NUREG-1635, Volume 4 (PDF page 73 of the link provided below). I'm not familiar with anymore of the history of this...

http://pbadupws.nrc.gov/docs/ML0117/ML011710324.pdf

Regarding breakaway oxidation in air, I am less concerned about this because it has received attention in the past. Through further interpretation of the Nathessan (sp?) ANL experiments, as well as use of the BWR Zirc Fire experiment results, we have attempted to model pre-breakaway, post-breakaway, and lifetime correlation transition. One can certainly challenge the uncertainty in such modeling, but one can't say that we haven't paid it due attention...

Don

From: Scott, Harold
Sent: Wednesday, December 19, 2012 10:14 AM
To: Helton, Donald
Subject: RE: Fuel Cladding / Oxidation FAQs

decrepitation

Breaking up of mineral substances when exposed to heat
By this you mean when the core degrades to debris?

Another topic might be breakaway cladding oxidation in air

From: Lee, Richard
Sent: Wednesday, December 19, 2012 9:30 AM
To: Voglewede, John; Raynaud, Patrick; Flanagan, Michelle; Scott, Harold
Subject: FW: Fuel Cladding / Oxidation FAQs

Let's get together sometime after the Holidays to discuss this.

From: Helton, Donald
Sent: Wednesday, December 19, 2012 9:13 AM
To: Lee, Richard
Cc: Corson, James; Esmaili, Hossein; Coyne, Kevin

E79

Subject: Fuel Cladding / Oxidation FAQs

Richard,

I'd like FSCB assistance in developing short (e.g., 1 paragraph) responses (each) to a few phenomenological issues that have arisen in the context of some of our collaborative work. The questions relate to the following topics:

- The use of noble metal chemical additives during NPP operation and the subsequent effects on cladding behavior (more background is provided in the attached email)
- Fuel fragmentation for high burnup fuel during clad rupture
- Fuel thermal conductivity degradation (e.g., IN-2011-21)
- Nitriding
- Hydriding
- Decrepitation
- Radiolysis (though not a cladding issue, I raise this because the updated EPRI SAMG basis document states that it should be considered)

The basic questions that arise regarding each are:

- To what extent are they considered by MELCOR (or more accurately to what extent were they covered by the cladding experiments that the MELCOR oxidation/fuel failure models were based on);
- To what extent should we be more or less concerned about the impacts of the phenomena during low-pressure air oxidation conditions (SFP and some reactor shutdown accidents);
- Why is their omission (when not considered) reasonable for PRA core damage surrogate (i.e., non-50.46) activities and severe accident analysis?

The projects that I see directly benefiting from these "FAQs" are:

- SFPSS
- Level 1 PRA confirmatory success criteria using MELCOR
- Vogtle Level 3 reactor shutdown analysis
- Vogtle SFP analysis

This is something that we can take some time to develop, but I think that by this Spring (ACRS meetings on SFPSS and Vogtle SFP; Byron NUREG) roll around, we'll want to have this in our hip pocket. Please let me know how you'd like to proceed.

Best,
Don

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US Risk Studies Related to Spent Fuel Storage or Transportation

Year	Study Title:	Scope	Lead entity	Synopsis
2007	A Pilot Probabilistic Risk Assessment Of a Dry Cask Storage System At a Nuclear Power Plant (NUREG-1864) [ML071340012]	Dry Cask Storage	NRC	The results of this analysis indicate that the risk is solely from latent cancer fatalities, and no prompt fatalities are expected. The risk is dominated by accident sequences occurring in three stages of the handling phase. These involve the drop of the transfer cask through the equipment hatch (Stage 18) and drops of the MPC into the storage overpack (Stages 20 and 21). The aggregated risk values are quite low. The estimated aggregate risk is an individual probability of a latent cancer fatality of 1.8E-12 during the first year of service, and 3.2E-14 per year during subsequent years of storage. Note that when insufficient information was available, "conservative bounding assumptions or estimates" were used. Other limitations include (i) no consideration of uncertainty and conservative assumptions about the translation of failure modes to hole sizes, amongst others.
2004	Probabilistic Risk Assessement (PRA) of Bolted Storage Casks: Updated Quantification and Analysis Report (EPRI-1009691)	Dry Cask Storage	EPRI	This report describes radiological risks and consequences to individuals from a bolted cask containing spent fuel from a PWR while the cask is on-site. The risk to the public from cask deisgn is extremely low, with no calculated early fatalities and a first year risk of latent cancer fatality of 5.6E-13 per year per cask. Subsequent year risk to the general public is even lower, again, with no early fatalities and a cancer risk of 1.7E-13 per cask per year. This report presents an update to the original study. The update was performed to remove selected conservatisms from the original analysis. The methods employed in the updated report are the same as those used in the original study. The seismic event evaluated in this study used a seismic hazard representative of a location in the Northeastern United States.
2001	Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants (NUREG-1738) [ML010430066]	Spent Fuel Pools/ Decommissioning	NRC	The results of the study indicate that the risk at SFPs is low and well within the Commission's Quantitative Health Objectives (QHOs). The risk is low because of the very low likelihood of a zirconium fire even though the consequences from a zirconium fire could be serious. The staff found that the event sequences important to risk at decommissioning plants are limited to large earthquakes and cask drop events. Some important conservatisms associated with this study when applied outside of its context include: (i) the use of assumed and often bounding configurations, (ii) simplified treatment of the thermal-hydraulic response, (iii) conservative assumptions regarding structural response, and (iv) EP response representative of a decommissioned site, amongst others. [Note that a supporting CFD study is documented in NUREG-1726.]
2000	Reexamination of Spent Fuel Shipment Risk Estimates (NUREG/CR-6672) [ML003698324]	Spent Fuel Transport	NRC	The results of this study and the previous studies demonstrate that the risks associated with the shipment of spent fuel by truck or rail are very small. Overall, the results of this study confirm the validity of the NUREG-0170 estimates of spent fuel incident-free population doses. The results also show that the NUREG-0170 estimates of spent fuel accident population dose risks were very conservative, as was believed to be true when NUREG-0170 was published

E80

Year	Study Title:	Scope	Lead entity	Synopsis
1997	Followup Activities on the Spent Fuel Pool Action Plan [ML003706412]	Spent Fuel Pools	NRC	The staff performed probabilistic screening analyses and found that, in most cases, event frequencies for sequences associated with these design issues were sufficiently low that further analyses were not warranted. In one instance where the probabilistic screening criteria was met, the staff performed a deterministic evaluation of the issue using plant-specific information and found that safety enhancements were not warranted.
1996	Loss of Spent Fuel Pool Cooling PRA: Model and Results (INEL-96/0334)	Spent Fuel Pools	NRC	The results of this study show that for a representative two-unit boiling water reactor, the annual probability of spent fuel pool boiling is $5x10^{-5}$ and the annual probability of flooding associated with loss of spent fuel pool cooling scenarios is $1x10^{-3}$. Qualitative arguments are provided to show that the likelihood of core damage due to spent fuel pool boiling accidents is low for most U.S. commercial nuclear power plants. It is also shown that, depending on the design characteristics of a given plant, the likelihood of either: a) core damage due to spent fuel pool-associated flooding, or b) spent fuel damage due to pool dryout, may not be negligible.
1989	Value/Impact Analyses of Accident Preventive and Mitigative Options for Spent Fuel Pools (NUREG/CR-5281) [ML071690022]	Spent Fuel Pools	NRC	Options studied included limited low-density reracking of spent fuel, installation of water sprays above the spent fuel pool, and the installation of redundant cooling and/or makeup systems. The results of these studies indicated that the measures were in general not likely to be cost effective. The reason for this is due to both the low likelihood of a spent fuel pool accident that could result in a significant radiological release and the high cost of proposed modifications. These insights are largely contingent upon compliance with guidelines developed for licensees to assure the safe handling of heavy loads in the vicinity of spent fuel pools thus reducing the likelihood of the structural failure of the pool and rapid loss of water inventory due to a cask drop event.
1989	Regulatory Analysis for the Resolution of Generic Issue 82, "Beyond Design Basis Accidents in Spent Fuel Pools" (NUREG-1353) [ML082330232]	Spent Fuel Pools	NRC	This was the first comprehensive evaluation of expanding requirements for fuel pool cooling. The analysis concluded that if the decay heat level is high enough to heat the fuel rod cladding to about 900C the oxidation becomes self-sustaining, resulting in a Zircaloy cladding fire.The conditional probability of a Zircaloy cladding fire given a complete loss of water was found to be 1.0 for PWRs and 0.25 for BWRs in high density configurations. The conditional probability of a Zircaloy cladding fire given a complete loss of water in low density storage racks is estimated to be at least a factor of five less than for the high density configurations. Although these studies conclude that most of the spent fuel pool risk is derived from beyond design basis earthquakes, this risk is no greater than the risk from core damage accidents due to seismic events beyond the safe-shutdown earthquake. Therefore, reducing the risk from spent fuel pools due to events beyond the safe-shutdown earthquake would still leave at least a comparable risk due to core damage accidents. Therefore, Alternative 1 - "No Action" is justified.

Year	Study Title:	Scope	Lead entity	Synopsis
1977	Final Environmental Statement on the Transportation of Radioactive Material by Air and Other Modes (NUREG-0170) [ML022590355]	Radioactive Material (Including Spent Fuel) Transport	NRC	This study examines the risk associated with transportation of radioactive material (including spent fuel). One conclusion of the study, amongst many, was that, "the environmental impacts of normal transportation of radioactive material and the risk attendant to accidents involving radioactive material shipments are sufficiently small to allow continued shipments by all modes." The study recognizes the large uncertainties associated with portions of the assessment, and committed to "continuing to study other aspects of transportation, such as the accident resistance of pacakges and the physical/chemical form of the radioactive contents, to maintain the present high level of safety..."
Note 1: This is not a comprehensive list of domestic studies, particularly with respect to transportation risk studies. However, it is believed to capture significant contemporary studies of broad relevance to NRC.				
Note 2: The "Comments" field has been populated by extracting text directly from the subject reports. Editorial comments (additional study limitations) have been added only for NUREG-1738 and NUREG-1864.				

Other Studies and Activities of Interest

Year	Study Title:	Scope	Lead entity	Synopsis
2010	Documentation of Evolution of Security Requirements with Respect to Mitigation Measures for Large Fires and Explosions. [ML092990438]	Spent Fuel Pools	NRC	To respond to a Commission's request, the staff performed a search to identify existing classified and unclassified documents which describe the evolution of the implementation of security requirements with respect to mitigation measures taken since September 11, 2001. Identified documents were cataloged in an Excel database called the "Consolidated Timeline Documents." The staff used the information contained in this database to prepare a summary of the information, which is contained in the subject document. The report is responsive to the Commission's request for a document describing the evolution of the implementation of security requirements with respect to mitigation measures for large fires or explosions starting with the advisories and through Phases 1, 2 and 3.
2009	Generic Environmental Impact Statement for License Renewal of Nuclear Plants Appendix E.3.7 (NUREG-1437, Revision 1, Draft Report for Comment) [available on the external NRC website]	Spent Fuel Pools	NRC	This assessment re-evaulated SFP environmental considerations related to spent fuel pools by considering information developed since the original License Renewal GEIS was issued in 1996. The update concluded that the environmental impacts from accidents at SFPs (as quantified in NUREG-1738) can be comparable to those from reactor accidents at full power (as estimated in NUREG-1150). Subsequent analyses performed, and mitigative measures employed, since 2001 have further lowered the risk of this class of accidents. In addition, even the conservative estimates from NUREG-1738 are much less than the impacts from full power reactor accidents as estimated in the 1996 GEIS. Therefore, the environmental impacts stated in the 1996 GEIS bound the impact from SFP accidents.
2008	Denial of Petitions for Rulemaking for PRM-51-10 and PRM-51-12 [ML080170587 (SECY); ML081710446 (SRM)]	Spent Fuel Pools	NRC	These documents describe the NRC's denial of two petitions for rulemaking (PRM), one filed by the Attorney General of the Commonwealth of Massachusetts and the other filed by the Attorney General for the State of California, presenting nearly identical issues and requests for rulemaking concerning the environmental impacts of high-density storage of spent nuclear fuel in SFPs. This action was included as an example of the numerous activities that have in litigation/contention space related to the topic of spent fuel storage.
2008	Assessment of Differential Risk in Spent Fuel Storage; High Density and Low Density Pool Storage (DSA Project Plan) [ML081790081]	Spent Fuel Pools	NRC	This document is a preliminary project plan put together by DSA in 2008, describing an approach for evaluating differential risk between the use of high density and low density SFP storage configurations.
2008	White Paper on Spent Fuel Pool Source Term Guidance [ML080780048]	Spent Fuel Pools	NRC	The paper provides a high-level comparison between the results of the RES SFP security assessments and the modeling employed in the Radiologic Assessment System for Consequence AnaLysis (RASCAL) computer code. Specifically, the paper investigates how newer, state-of-the-art MELCOR and CFD analysis informs the timing and magnitude of source terms for postulated spent fuel pool accidents, within the context of incident response.

Year	Study Title:	Scope	Lead entity	Synopsis
2002 - 2008	RES SFP Security Assessments [e.g., ML062550218]	Spent Fuel Pools	NRC	Following the events of 9/11/01, RES conducted security assessments for spent fuel pools as part of the agency's overall response. These studies considered aircraft impact and and other sabotage events. They included extensive deterministic modeling of relevant accident progression scenarios, and formulated the technical basis for later agency activities under ICM B.5.b. and the 'Phase 2' site-speicific assessments. This project also included confirmatory BWR hydraulic and ignition testing at Sandia; ongoing PWR testing is being performed as part of an OECD project.
2002 - 2007	NMSS Dry Cask Storage and Transportation Security Assessments [e.g., ML060340452]	Dry Storage and Transportation	NRC	Following the events of 9/11/01, NMSS conducted security assessments for dry cask storage and transportation as part of the agency's overall response. The staff evaluated appropriate representative spent fuel storage casks, spent fuel transportation packages, and radioactive material (non-spent fuel) transportation packages that were certified at the commencement of the evaluation. These studies considered aircraft impact and other sabotage events. The work (along with other non-reactor/SFP assessments is covered in SECY-06-0045.
2006	Safety and Security of Commercial Spent Nuclear Fuel Storage	Spent Fuel Storage Safety and Security	National Academies	This study documents a Congressionally mandated study by the National Academies of the safety and security of spent nuclear fuel storage. The study was sponsored by the NRC, and the National Academies Committee was briefed on numerous occasions by NRC staff regarding past and ongoing studies related to this topic. The study resulted in a classified and publicly available report. The publicly available report documented numerous findings and recommendations, many of which have been addressed as part of the agency's continued activities in this area (e.g., the "Phase 2" site-specific assessments). The agency's initial response to the study can be found at ML050280428.
2003	NRC Review of "Reducing the Hazards from Stored Spent Power-Reactor Fuel in the United States" [ML031210075]	Spent Fuel Pools and Dry Cask Storage	NRC	This paper concludes that the authors' assessment of possible spent fuel pool accidents stemming from potential terrorist attacks does not address such events in a realistic manner. In many cases, the authors rely on studies that made overly conservative assumptions or were based on simplified and very conservative models. The NRC does not believe that the fundamental recommendation of this paper, namely that all spent fuel more than five years old be placed in dry casks through a crash 10-year program costing many billions of dollars, is at all justified. Spent fuel stored, in both wet and dry storage configurations, is safe and measures are in place to adequately protect the public.

Year	Study Title:	Scope	Lead entity	Synopsis
2003	Reducing the Hazards from Stored Spent Power-Reactor Fuel in the United States [ML031130327]	Spent Fuel Pools and Dry Cask Storage	Robert Alvarez et al.	From the report: It has been known for more than two decades that, in case of a loss of water in the pool, convective air cooling would be relatively ineffective in such a "dense-packed" pool. Spent fuel recently discharged from a reactor could heat up relatively rapidly to temperatures at which the zircaloy fuel cladding could catch fire and the fuel's volatile fission products, including 30-year half-life Cs-137, would be released. To reduce both the consequences and probability of a spent-fuel-pool fire, it is proposed that all spent fuel be transferred from wet to dry storage within five years of discharge. The cost of on-site dry-cask storage...is estimated at $3.5-7 billion dollars...The removal of the older fuel would reduce the average inventory of Cs-137 in the pools by about a factor of four, bringing it down to about twice that in a reactor core.
2002	Analysis of Spent Fuel Heatup Following Loss of Water in a Spent Fuel Pool: A Users' Manual for the Computer Code SHARP (NUREG-CR-6441) [ML021050336]	Spent Fuel Pools	NRC	An analysis of spent fuel heatup, using representative design parameters and fuel loading assumptions was performed. Sensitivity calculations were also performed to study the effect of fuel burnup, building ventilation rate, baseplate hole size, partial filling of the racks, and the amount of available space to the edge of the pool. The spent fuel heatup was found to be strongly affected by the total decay heat production in the pool, the availability of open spaces for air flows, and the building ventilation rate. [Note that SFP analyses performed by RES after this time did not rely on this computer code. Rather, they relied on the use of MELCOR and 2 CFD codes, along with confirmatory experiments.]
2002	Dry Cask Storage Probabilistic Risk Assessment Scoping Study (EPRI-1003011)	Dry Cask Storage	EPRI	The report describes a dry cask storage PRA approach via appropriate supporting elements and investigates how the elements are best analyzed and integrated to provide PRA results and insights. This report does not document the development and results of a completed dry cask storage PRA; rather, it assesses applicable methodologies for developing such a risk assessment.
1997	Operating Experience Feedback Report: Assessment of Spent Fuel Cooling (NUREG-1275, Vol. 12) [ML010670175]	Spent Fuel Pools	NRC	The overall conclusions were that the typical plant may need improvements in SFP instrumentation, operator procedures and training, and configuration control. [Note that this is the conclusion stated in the report, and has not been placed in the regulatory context of balance-of-plant activities since the issuance of that report.] The staff determined that loss of SFP coolant inventory greater than 1 foot occurred at a rate of about 1 event per 100 reactor years. Loss of SFP cooling with a temperature increase greater than 20 F occurred at a rate of approximately 3 events per 1000 reactor years. The primary cause of these events was human error. The staff determined that utilities' efforts to reduce outage duration have resulted in full core off-loads occurring earlier in outages. This increased fuel pool heat load reduces the time available to recover from a loss of SFP-cooling event early in the outage.

Year	Study Title:	Scope	Lead entity	Synopsis
1977	Spent Fuel Heatup Following Loss of Water During Storage (NUREG/CR-0649)	Spent Fuel Pools	NRC	An analysis of spent fuel heatup following a hypothetical accident involving drainage of the storage pool is presented. Computations...have been performed to assess the effect of decay time, fuel element design, storage rack design, packing density, room ventilation, drainage level, and other variables on the heatup characteristics of the spent fuel and to predict the conditions under which clad failure will occur...It has been found that the likelihood of clad failure due to rupture or melting following a complete drainage is extremely dependent on the storage configuration and the spent fuel decay period, and that the minimum prerequisite decay time to preclude clad failures may vary from less than 10 days for some storage configurations to several years for others. The potential for reducing this critical decay time either by making reasonable design modifications or by providing effective emergency countermeasures has been found to be significant. - Note that this study considers both low-density racking and mitigative accessibility, which may be of particular interest.
Note: This list contains references to sensitive documents, but is not believed to contain sensitive information by itself.				

Draft 11/21/12
with Jennifer Uhle markup

Tier 3, Additional Recommendation 5 (AR5) objective (from SECY-11-0137, SECY-12-0095, and December 15, 2011 SRM) is stated as follows:

> The purpose of this program plan is to evaluate whether there would be a substantial increase in the overall protection of public health and safety from expedited transfer of spent fuel into dry cask storage. This plan will be based on insights and lessons learned from the events at Fukushima Dai-ichi and provide a sound technical basis for considering whether additional regulatory action may be justified. While the staff has previously concluded that public health and safety is adequately protected, the staff has determined that it should confirm, using insights from Fukushima, that both SFPs and dry cask storage continue to provide adequate protection, and assess whether any significant safety benefits (or detriments) would occur from expedited transfer of spent fuel to dry casks.
>
> For the purposes of this program plan, expedited transfer is defined as the more expeditious transfer of spent fuel into dry cask storage over a period of a few years, and could mean the removal of all spent fuel with more than five years decay time to dry cask storage within a five to ten year timeframe.

Not clear. what fuel is what

The staff provided the Commission the program plan for Tier 3 AR5 in SECY 12-0095 on July 13, 2012. In this SECY, the staff proposed a five (5) step plan concluding in a recommendation to the Commission regarding the need for regulatory action to require expedited transfer of spent fuel from spent fuel pools to dry cask storage. The staff informed the Commission that they would be using the current regulatory framework in developing a recommendation and would provide additional information for a detailed schedule with milestones.

Subsequent to the July 13 SECY, the staff developed a strategy to provide a three phase stepped approach to resolving this issue, with a Commission paper to be provided at the conclusion of each phase. The ultimate objective ~~for~~ of this approach is to allow the staff to provide information to the Commission in a timely manner, while allowing for major policy decisions to be ~~played out~~ evaluated and integrated in a later phase. The later considerations would include Tier 1 Order implementation elements (currently being developed and implemented), additional economic consequence considerations and any resultant changes in the regulatory framework. The phased approach would allow the staff to provide the Commission with shorter-term information in a concise package while the longer term items could be completed in parallel. All internal stakeholders have been involved with this effort at the staff and management level including RES, NRR, NSIR, NMSS, NRO, and JLD counterparts.

The phased approach to resolving this issue is described below:

Phase 1 – Measure whether a substantial increase in public health and safety would be achieved by requiring a change from high to low density SFP loading (Goal to complete by 10/31/13)

- Phase 1 will determine the maximum potential benefit per reactor year of expedited fuel

E8/

this is included in SFPSS

the SFPSS has the risk from fuel movement. It sounds like this needs to be added

we are comparing the probability informed consequences between two loadings.

transfer by comparing the safety benefits of low density fuel storage relative to high density fuel storage. The comparison will be informed by the initiating frequency and consequences from the Spent Fuel Pool Scoping Study (SFPSS) Interim Report for a representative boiling water reactor (BWR) as an indicator of any major changes in our understanding of safe storage of spent fuel in a SFP. It will also be informed by a preliminary evaluation of relative risks involved in SFP storage, spent fuel transfer, and dry storage. The staff will use past regulatory analyses for spent fuel consequences (i.e., NUREG-1353 and NUREG-1738) to extend the applicability to the operating reactor fleet. ~~The determination whether expedited spent fuel movement achieves the threshold of relative substantial safety enhancement in accordance with 10 CFR 50.109, Backfitting, will determine the basis for conducting further inquiry into this matter.~~ In addition, the phase 1 analysis will include the following within the bases for the recommendation(s) provided:

Comment [MLS1]: I continue to believe that 50.109 cannot be invoked based on preliminary and incomplete view of the situation. This will not be the point to invoke 50.109 to require licensee action. That belongs in Phase 2. If the plan is to simply use the criteria in 50.109 as a guideline/thought process for whether to move to Phase 2, we should state that clearly.

- Sensitivity analyses to address the safety significance of effective mitigation strategy implementation,
- Human reliability study information as directed by SRM M120607C, Commission Meeting with ACRS, dated July 16, 2012, which directed the staff to look at the capacity to implement effective SFP cooling mitigating strategies from 10 CFR 50.54 (hh) or Order EA-12-49 ("Mitigation Strategies for Beyond Design Basis External Events"),
- Comparative consideration of the performance of SFP during real incidents, which was directed by SRM M120607C, Commission Meeting with ACRS, dated July 16, 2012,
- Presentation of integrated results of the phase 1 draft analyses to the ACRS for comment.

how is this done?

The recommendation to the Commission for phase 1 will be made via a notation vote SECY in October 2013. It will describe the results of the preliminary regulatory analysis to determine whether there is potential for a substantial increase in the overall protection of public health and safety for the low density SFP configuration using the current regulatory framework, such that proceeding to Phase 2, as described below, is justified. ~~(To be clear, phase 1 only considers the resultant low density SFP configuration, while phase 2 will determine the risk and costs associated with fuel transfer from the SFP to dry storage.)~~ This analysis may also show the need for inclusion of additional mitigating strategies for high density SFP configurations. Within the phase 1 recommendation to the Commission, the staff will develop a single document to provide a complete executive summary of the staff's analysis and bases for recommendation(s) to the Commission. This document will include the information from the SFPSS and explanative additional information such that an integrated basis for the staff's conclusions is clearly presented. The staff ~~does not plan to provide the SFPSS to the ACRS nor does the staff intend~~ also plans to provide the SFPSS to the Commission in 2013 consistent with SRM direction and subsequently to publish the SFPSS as a NUREG. ~~The staff intends to provide a notation vote SECY to the Commission by October, 2013. The SECY would include a recommendation as to~~

Comment [MLS2]: Question of whether SFPSS will be provided to ACRS in spring 2013 to support its review needs further discussion.

NRR wants to ask Comm to allow us to not provide SFPSS to them separately

~~whether, based on the preliminary evaluations, there is a sufficient potential for a substantial increase in the overall protection of public health and safety, and that the direct and indirect costs of implementation are justified in view of this substantial increase in protection to proceed tosuch that proceeding to~~ Phase 2, described below~~, is justified.~~

Phase 2 – Detailed analysis of costs and benefits for expedited loading to dry storage (Goal to complete by 6/30/15)

Phase 2 will evaluate the effects of expedited transfer of spent fuel from the SFP to dry storage to achieve low density SFP storage on the net benefit results determined in Phase 1. This regulatory analysis will include the evaluation of fuel loading risks (i.e., risk of cask drops resulting from increased frequency of cask movement), occupational radiological exposure resulting from cask movement activities[1], cost and benefit analysis for transfer to dry storage and burial, public risk considerations such as security assessments of dry storage protection, regulatory considerations for repackaging dry storage casks for transportation, and current/future permanent storage proposals. In addition, insights from implementation of Order EA-12-049 will be included. The staff will also include information to the Commission resulting from the following:

- Comparative assessment of SFPSS results against previous studies of safety consequences associated with loading, transfer, and long-term dry storage as directed by SRM M120607C, Commission Meeting with ACRS, dated July 16, 2012, (Note, this will require a request by the staff to extend the date for this action)
- Comparison of practices for spent fuel transfer from pools to dry cask storage in other countries versus the United States, as directed by the Commission in SRM M120807B, Japan Lessons Learned Commission Meeting, dated August 24, 2012 (Note, this will require a request by the staff to extend the date for this action)

The recommendation to the Commission for phase 2 will complete the regulatory analysis for expedited transfer of spent fuel using the current regulatory framework. The staff intends to provide a notation vote SECY to the Commission by June, 2015.

way too late

Should the Commission elect to have the staff continue the evaluation based on recommendations for changes in determination of economic consequences and/or direction for new regulatory analysis bases, then the staff can continue the Tier 3 program plan with the inclusion of phase 3.

Phase 3 – Consideration of other factors (e.g., criticality, mitigating strategies, solar storms, economic consequences, new regulatory framework) (Goal to complete by 7/31/17)

Phase 3 will include additional issues that have been raised by staff and other stakeholders and could have an impact on the analysis that was conducted in Phase 1 and Phase 2 to require the

[1] Industry estimates of occupational exposures from expedited cask movement are contained in EPRI 2012 Technical Report "Impacts Associated with Transfer of Spent Nuclear Fuel from Spent Fuel Storage Pools After Five Years of Cooling," Revision 1.

the

what is safety risk?

expedited transfer of spent fuel. The resulting recommendation(s) will consider the change in net benefits determined in phases 1 and 2 resulting from any change in the consideration of economic consequences and/or resulting from a new regulatory framework, as well as affects to overall ~~safety~~ risk associated with the following considerations:

- Multi-unit risk to SFP loading and configuration strategies
- Inadvertent criticality in SFP from boron absorbing material dedgration/loss from a seismic event, boiling, or other SFP water loss scenario
- Long term loss of power events not addressed by Order EA-12-049 (as informed by pending solar storm petition conclusions)

The recommendation(s) to the Commission for phase 3 analysis could modify earlier phase 1 and 2 recommendations using new regulatory analysis considerations, or in the case where the current regulatory framework is unaltered, expand ~~the Tier~~ 3 recommendation(s) scope to include a broad range of conditions for the Commissions direction. The staff intends to provide a notation vote SECY to the Commission by July 2017.

dates are way too long

DPO - Miela

FEVS

7

SFPSS-SRM Status Report for DSA Management Briefing (Dec. 6, 2012)

Item	Status	Schedule Concern

Task 1.1	• Draft plan submitted: Oct. 5th
	  o Review the information already available at the time of the SFP scoping study and information obtained since the completion of the interim draft report for that study. The emphasis of this additional review will be on structural details and seismic loads for a single SFP (SFP for Fukushima Daiichi Unit 4) for which more information is available.
	    ▪ Product: a revised and expanded version of the observations and comparisons in Section 9 of the interim draft report.
	  o Revised Completion Date: Dec. 20th
	• No additional info. from Japan on SFPs so far, however more info. is available now than in June 2012
		No

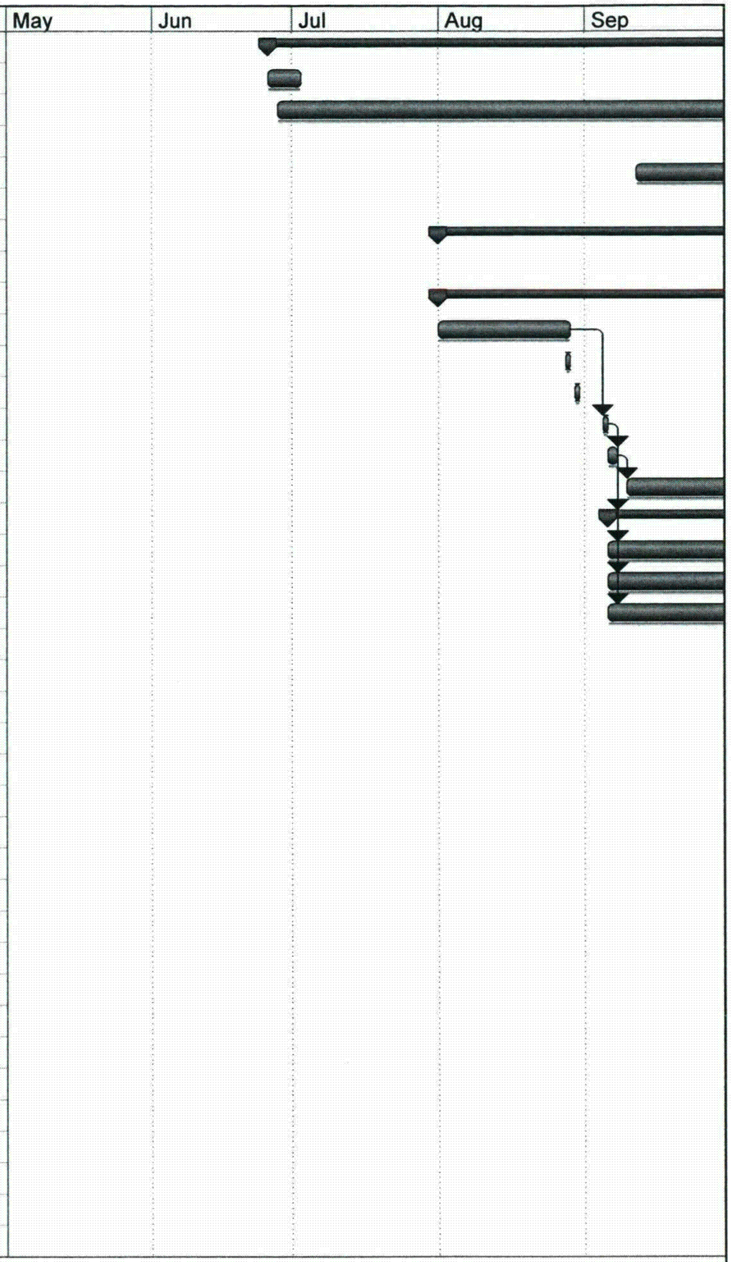
Task 2.1	• Current plan: Use NUREG-1864 (Dry Cask Pilot PRA) source terms to generate output metrics consistent with those used in SFPSS (e.g., weighted and un-weighted societal dose, land contamination, and displaced individuals) and with site characteristics consistent with the SFPSS. This requires developing modified SFPSS input files, executing code, and writing a summary. • Product: Written summary for inclusion into SFPSS appendix. • Updated code arrived on Nov. 12, model setup is in progress. • AJ, Keith, Drew, and Earl met on Nov. 29th to discuss how to incorporate and best report the results for the comparison. Result: RES to provide NMSS with analysis for review	Not yet

E82

Item	Status	Schedule Concern
Task 2.2	• Draft plan developed (draft plan submitted on Nov. 26th), G. DeMoss has ok'ed plan o 3 phases defined: current situation, expedited movement to achieve low-density configuration, and static low-density configuration o Analysis type: qualitative gap analysis that is meant to provide risk insights o Write-up intended to highlight relatively new information Note: B. Wagner met with F. Schofer and A. Nosek on Dec. 3rd to discuss integration with the regulatory analysis.	Not yet
HRA	• Comment Resolution Continues: updated draft version of HRA report sent to other-office stakeholders on Nov. 20th o G. Casto sent a second round of comments on Dec. 4th o R. Sullivan sent a second round of comments on Dec. 4th	Not yet
Final SFPSS runs (w/updated evacuation models)	• Updated code arrived on Nov. 12, all base case runs are finished (including those with extended evacuations). • Next step is analysis and incorporation into the report. • Preliminary insights: The new runs lowered health consequences by up to 3.5%. Of this 3.5%, up to 0.5% may have been from the extended evacuations, and the rest of it is from other code fixes. The extended evacuations may have in some cases lowered the emergency phase consequences by up to 10-25%.	Not yet
Sensitivity Studies	See separate chart	

Sensitivity Study Name	MELCOR Status	MACCS2 Status	Analysis/Write-up Status	Overall % Completion	Schedule Concern
Multi-unit source term release	100% complete, info. given to AJ	Discontinued due to limited available time.	In progress	70%	No
Time truncation	100% complete, info. given to AJ	Runs complete.	In progress	70%	No
Land contamination	N/A	Runs complete.	In progress	70%	No
Molten core-concrete interaction (MCCI)	100% complete, info. given to AJ	Runs subject to cancellation, dependent on available time.	In progress	70%	Under assessment, may only include MELCOR results.
1x8 configuration	100% complete	N/A	Analysis complete, write-up not started	90%	No
Hydrogen combustion	100% complete, info. given to AJ	Discontinued; judged by technical leads to provide limited benefit compared to other priorities.	MELCOR results and analysis are documented in the report.	100%	No

ID		Task Name	Duration	Start	Finish	Predecessors
1		**Phase 1: SFPSS July 2012 Scope**	**89 days?**	**Tue 6/26/12**	**Wed 10/31/12**	
2		RES SFPSS (original scope)	5 days	Tue 6/26/12	Mon 7/2/12	
3		RES SFPSS HRA w/50.54 measures compl	66 days?	Thu 6/28/12	Mon 10/1/12	
4		HRA added to draft report	15 days	Tue 10/2/12	Tue 10/23/12	3
5		Consider Fuku. Empirical evidence	35 days	Wed 9/12/12	Wed 10/31/12	
6						
7		**Phase 2: SFPSS SRM-related Scope**	**293 days?**	**Wed 8/1/12**	**Tue 10/1/13**	
8						
9		**Phase 2.1: Comparative Assessment**	**161 days?**	**Wed 8/1/12**	**Tue 3/26/13**	
10		NMSS to determine need for add'l rese	20 days?	Wed 8/1/12	Tue 8/28/12	
11		DD Briefing	1 day	Tue 8/28/12	Tue 8/28/12	
12		RES FO Briefing	1 day	Thu 8/30/12	Thu 8/30/12	
13		RES/NMSS align on SRM team/scope	1 day	Wed 9/5/12	Wed 9/5/12	10
14		Draft SRM project memo	2 days	Thu 9/6/12	Fri 9/7/12	13
15		SRM project memo concurrence	25 days	Mon 9/10/12	Mon 10/15/12	14
16		**SRM project work/documentation**	**78 days**	**Thu 9/6/12**	**Mon 12/31/12**	**13**
17		Task 1	72 days	Thu 9/6/12	Thu 12/20/12	13
18		Task 2.1	78 days	Thu 9/6/12	Mon 12/31/12	13
19		Task 2.2	78 days	Thu 9/6/12	Mon 12/31/12	13
20		SRM-related products added to draft re	5 days	Thu 1/3/13	Wed 1/9/13	16FS+1 day
21		Final editing of draft report w/Task 2.2 a	3 days	Thu 1/10/13	Mon 1/14/13	20
22		ACRS Pre-brief NRR/NMSS/RES	1 day	Mon 3/18/13	Mon 3/18/13	16
23		ACRS Pre-brief RES FO	1 day	Wed 3/20/13	Wed 3/20/13	16
24		Pre-brief JLD Steering Committee	1 day	Tue 3/26/13	Tue 3/26/13	16
25						
26		**Phase 2.2: ACRS Review**	**94 days**	**Mon 3/25/13**	**Mon 8/5/13**	**21**
27		Draft report to ACRS	0 days	Mon 3/25/13	Mon 3/25/13	
28		ACRS Subcommittee Meeting	1 day	Thu 4/25/13	Thu 4/25/13	
29		ACRS Full Committee Meeting	1 day	Fri 5/10/13	Fri 5/10/13	
30		ACRS Letter Issuance (3 wks)	15 days	Mon 5/13/13	Mon 6/3/13	29
31		Address ACRS Letter Comments in Re	44 days	Tue 6/4/13	Mon 8/5/13	29,30
32						
33		**Phase 2.3/2.4: Info. Paper/Report Concur**	**72 days**	**Mon 1/7/13**	**Thu 4/18/13**	**16**
34		Team Review	7 days	Mon 1/7/13	Tue 1/15/13	16FS+3 days
35		Team Comment Resolution/Concurrenc	7 days	Wed 1/16/13	Fri 1/25/13	34
36		Other Office WG review?	7 days	Mon 1/28/13	Tue 2/5/13	35
37		Other Office WG comment resolution	7 days	Wed 2/6/13	Thu 2/14/13	36
38		Branch Chief review	7 days	Fri 2/15/13	Tue 2/26/13	37
39		B.C. comment resolution/Concurrence	7 days	Wed 2/27/13	Thu 3/7/13	38



Project: DRAFT SFPSS-SRM Activitie
Date: Tue 12/11/12

Task	Milestone	External Tasks
Split	Summary	External Milestone
Progress	Project Summary	Deadline

E83

ID		Task Name	Duration	Start	Finish	Predecessors	May	Jun	Jul	Aug	Sep
40		Division Director review	10 ~~20~~ days	Fri 3/8/13	Thu 4/4/13	39					
41		Div. Dir. Comment resolution/Concurrei	5 ~~10~~ days	Fri 4/5/13	Thu 4/18/13	40					
42						27 follows 41.					
43		**Phase 2.3/2.4 Info. Paper/report - office le**	**190 days?**	**Wed 1/2/13**	**Tue 10/1/13**	**18**					
44		Start Draft Info. (SECY?) paper (pre-Ta	20 days	Wed 1/2/13	Wed 1/30/13	18					
45		Finish Draft Info. (SECY?) Paper (post-	10 days	Thu 1/31/13	Wed 2/13/13	44,19					
46		Initial EDO Alignment Brief?	1 day?	Thu 2/14/13	Thu 2/14/13	45					
47		Other Office Director review	20 days	Thu 6/6/13	Wed 7/3/13	41,29					
48		Other O.D. comment resolution	10 days	Fri 7/5/13	Thu 7/18/13	47					
49		OGC Review	20 days	Thu 7/25/13	Wed 8/21/13	47					
50		OGC comment resolution?	7 days	Thu 8/22/13	Fri 8/30/13	49					
51		CFO Review	7 days	Fri 7/19/13	Mon 7/29/13	48					
52		CFO comment resolution	5 days	Tue 7/30/13	Mon 8/5/13	51					
53		RES FO Review	7 days	Tue 9/3/13	Wed 9/11/13	52,50					
54		RES FO comment resolution	5 days	Thu 9/12/13	Wed 9/18/13	53					
55		"Final-check" EDO Alignment Briefing	1 day?	Thu 9/12/13	Thu 9/12/13	53					
56		SECY paper to OEDO	2 days?	Thu 9/19/13	Fri 9/20/13	55,53,54					
57		SECY paper to SECY	7 days	Mon 9/23/13	Tue 10/1/13	56					
58											
59											
60		**Finalize Communication Plan w/results**	**137 days?**	**Mon 10/1/12**	**Fri 4/19/13**						
61		Revise Draft to include available results	59 days	Mon 10/1/12	Thu 12/27/12						
62		Revise Draft to include available Task 2	20 days	Mon 1/7/13	Mon 2/4/13	19					
63		Team Review/concurrence	5 days	Tue 2/5/13	Mon 2/11/13	62					
64		Team Comment Resolution	7 days	Tue 2/12/13	Thu 2/21/13	63					
65		Other Office WG review?	5 days	Fri 2/22/13	Thu 2/28/13	64					
66		Other Office WG comment resolution	7 days	Fri 3/1/13	Mon 3/11/13	65					
67		FSCB Branch Chief review	5 days	Tue 3/12/13	Mon 3/18/13	66					
68		B.C. comment resolution	3 days	Tue 3/19/13	Thu 3/21/13	67					
69		Division Director review	5 days	Fri 3/22/13	Thu 3/28/13	68					
70		Div. Dir. Comment resolution	5 days	Fri 3/29/13	Thu 4/4/13	69					
71		RES OD/OPA/OCA review	5 days	Fri 4/5/13	Thu 4/11/13	70					
72		RES OD/OPA/OCA Comment resolutio	5 days	Fri 4/12/13	Thu 4/18/13	71					
73		Post Plan on EDO Website	1 day?	Fri 4/19/13	Fri 4/19/13	72					

Project: DRAFT SFPSS-SRM Activitie
Date: Tue 12/11/12

Task		Milestone		External Tasks	
Split		Summary		External Milestone	
Progress		Project Summary		Deadline	

Compton, Keith

To: Burnette, Danielle
Cc: Santiago, Patricia (Patricia.Santiago@nrc.gov)
Subject: RE: Emails for FOIA-PA-2013-00240 - transmittal method and review procedures for electronic files?

Danielle,

I have reviewed my sent emails in response to FOIA-PA-2013-00240, looking for emails containing data, assumptions, computer programs, quantitative or qualitative results, or analyses related to spent fuel accident risks. I have identified only one email that I consider to be responsive at this stage. I don't believe that there is any need to withhold any portion of it. I am sending a printout over. I have a substantially larger number of sent emails (about 80-90, not including attachments) that might be considered potentially relevant, but that are all essentially early drafts; editorial comments on early drafts; or suggested revisions of early drafts, of either the draft Spent Fuel Pool Study or the draft Waste Confidence EIS (that I was reviewing for consistency with the draft SFPS). My understanding is that such correspondence is akin to "previous versions" of these two reports and are therefore subsumed by the final reports (to the extent that they do not contain substantive information that did not show up in the final report). However, I am retaining them in case the scope is expanded (or in case my understanding of the scope is wrong).

I will begin going through my received emails and other files, and will transmit anything that I consider to be responsive, and will archive anything that might need to be held for further review in the event of an expansion of the scope. Please let me know if you have any questions about my review or if I need to do anything else. Thanks!

Keith Compton

From: Burnette, Danielle
Sent: Thursday, July 25, 2013 10:12 AM
To: Compton, Keith
Cc: Algama, Don
Subject: RE: Emails for FOIA-PA-2013-00240 - transmittal method and review procedures for electronic files?

Hi Keith,

If you can print them out and send them to me interoffice mail "5F53"

Thank you
Danielle

From: Compton, Keith
Sent: Tuesday, July 23, 2013 11:42 AM
To: Burnette, Danielle
Cc: Algama, Don
Subject: Emails for FOIA-PA-2013-00240 - transmittal method and review procedures for electronic files?

Danielle,

I have gotten organized enough to start identifying the documents that I have that I believe may be responsive to FOIA-PA-2013-00240 (the FOIA on SFP accidents) and would like to start transmitting documents. I have saved the relevant sent emails as pdf documents, which appears to save any email attachments as

E84

attachments to the pdf. I'd like to know how you would like the emails – as electronic versions emailed to you, or as printouts. If as electronic versions, what are our responsibilities for reviewing any electronic metadata or attachments to ensure that they do not contain any non-releasable information? Thanks –

Keith Compton

From: Compton, Keith
To: Gonzalez, Felix
Subject: RE: Release Fractions
Date: Friday, March 15, 2013 6:47:00 PM

Felix,

I just checked NUREG-2125 (Spent Fuel Transportation Risk Assessment), issued May 2012. Table 5-10 (Parameters for Determining Release Functions for the Accidents that Would Result in Release of Radioactive Material), on p. 123, provides rod-to-cask release fractions for particulates ranging from 2.4E-6 to 4.8E-6. These are further discussed in Appendix E, Table E-17, where the technical basis for these numbers is identified as: "From the release fraction in Hanson et al. Particles— (2008), Table 4.10). "
The full citation for Hanson et al 2008 is
"Hanson, B.D., Daniel, R.D., Casella, A.M., Wittman, R.S., Wu, W., MacFarlan, P.J., and Shimskey, R.W. "Fuel-in-Air FY07 Summary Report," Revision 1, PNNL-17275, Pacific Northwest National Laboratory, Richland, WA, 2008."
I have a pdf version of that report if you would like it. I could be wrong, but I think that was also the basis for the release fractions used in the DOE Yucca Mountain Preclosure Safety Analysis.

I also checked NUREG/CR-6672 (Reexamination of Spent Fuel Shipment Risk Estimates), issued in Feb 2000. Release fractions are discussed in Section 7. In particular, it is stated (p. 7-34) that "use of values of 4 x 10-7 and 3 x 10-5 respectively for FRC for release of particles during non-impact and impact accidents seems appropriate." The actual values used appear to be in Tables 7.31, where the highest particulate release fraction that I found was 2.4E-5 (admittedly a quick scan).

In short, both of these references have much lower values for the rod-to-cask release fraction of 1.2E-3 that seems to be used in NUREG-1864. I am wondering whether there is some confusion regarding which release fraction Gordon is referring to – there is the rod-to-cask release fraction and the cask-to-environment release fraction, and I am wondering if he is thinking of the cask to environment release fraction.

It sounds like you are doing the same thing that I did, with regard to checking the transportation risk assessment. If you find results different than what I found I would appreciate it – mine was just a quick look. I may certainly have missed something, and hope that you can correct me if I am misreading anything. Would appreciate hearing anything that you find out. Thanks, and have a great weekend - KLC

From: Gonzalez, Felix
Sent: Friday, March 15, 2013 9:42 AM
To: Compton, Keith
Cc: Wagner, Brian
Subject: Release Fractions

Keith:

I spoke with Gordon and he told me that the Release Fractions that were used in the Transportation Risk Assessment where the same as in the NUREG-1864. I am reading the Transportation Risk Assessment to corroborate to see if there are any differences. I

E85

know that the these release fractions might be extremist but if the case is that both use the same release fraction it might be our only option as I do not think we have the resources to explore or come up with other release fractions.

I am hoping that getting into more detail into the fuel failure analysis but we will see about it.

I will try to schedule another meeting and include guys from NMSS in the discussion probably in the next two or three weeks as I will be out next week.

Thanks:

FELIX E. GONZALEZ
PROJECT MANAGER
RISK & RELIABILITY ENGINEER
US Nuclear Regulatory Commission
RES/DRA/Fire Research Branch
Phone: 301-251-7596
Office: CSB 04C11

TN:
71
Aiken → [illegible] water

Ahn, Tae

From: Einziger, Robert
Sent: Monday, May 21, 2012 9:59 AM
To: Raynaud, Patrick; Ahn, Tae; Clifford, Paul; Flanagan, Michelle; Gordon, Matthew; Landry, Ralph; Panicker, Mathew; Patel, Amrit; Proffitt, Andrew; Scott, Harold; VanWert, Christopher; Voglewede, John; Wu, Shih-Liang
Cc: Lee, Richard; Mendiola, Anthony; Donoghue, Joseph; Guttmann, Jack; Pstrak, David
Subject: RE: TAG-Fuels Meeting Minutes

Pat,

Below in red are some comments on the recent TAG meeting

Spent Fuel

a) Cladding forces due to alpha decay and helium build-up: During long periods of extended storage, helium builds up in the fuel-cladding gap as a result of alpha decay, and results in increased pressurization of the fuel rod. This phenomenon may cause cladding failure over long storage times. A recent paper by Ahn and Rondinella describes and discusses this phenomenon, which can drive delayed hydride cracking (DHC), hydride reorientation, and ultimately cause cladding failure.

Note that this paper has not been published. I have recommended to management that in its present form the paper is speculative and premature and should not be published.

b) Public comments on "Technical Information Needs for Extended Storage": NMSS is currently collecting public comments on the Extended Storage and Transportation (EST) Gap Analysis report. The NRC staff has already been given a chance to comment prior to the public comment period.

c) Priority for hydride reorientation understanding: Harold Scott reported that the EST Gap Analysis says that hydride reorientation is well understood, and that he does not agree 100% since the Penn State work is still investigating this phenomenon, and work is underway. He asked if this implies that Penn State should no longer be funded to investigate this issue. The general consensus was that more information on the subject was welcome, but may not impact the way EST is treated in the regulations.

   The gap report never said that the phenomena hydride reorientation was well understood. It said that NRC had enough information from its testing program and other programs to identify an issue with respect to the ductility changes I the fuel. The information at hand was sufficient for NMSS to determine that there was a regulatory concern that the the applicants needed to address hence the need for further NRC research was a low priority.

d) Schedule and milestones on High Burnup Fuel Consequence Analysis: CNWRA produced a study on the source term associated with high burnup fuel, and the report is under review by the staff. Katie Wagner of RES/DSA/FSCB is the project manager. A comment was made that it appeared that this work was forced onto both the NRC staff and the CNWRA. Harold Scott suggested that the staff (particularly the members of TAG-F) should take a close look at this report

There was a item in the user need to pursue this task IF ACTIVATED BY NMSS. This task was activated by RES of it's own accord and without the approval of NMSS. In addition the scope was ill defined. As I understand it, doing the work at CNWRA was forced upon RES by management. This was opposed by NMSS as they felt that CNWRA did not have he capabilities to do this work.

Fuel Structural Integrity

Ex6

a) Plan for publication of high burnup fuel cladding bend test results from ORNL regarding NMSS-2009-002 User Need: RES and NMSS meet quarterly to discuss the status of User Needs. In the last meeting, it came to RES attention that the NMSS Op Plan has a milestone to publish the results from the ORNL research program on high burnup spent fuel mechanical properties (stiffness of a fueled segment) and fatigue behavior (cycles to failure for different stress levels: S-N curve). Michelle Flanagan summarized the ORNL work as follows. At this point in time, ORNL has mostly finished developing the new test capabilities that are required for this work. 4-point bend testing will be used. For now, the apparatus is outside the hot cell, but it will be moved in-cell late in the summer, after full benchmarking and testing of the apparatus. Irradiated fuel rod testing should take place in the fall and winter. There was some discussion about what NMSS should/should not include in their milestones, since they ultimately do not have direct control over the progress of the ORNL research project.

As far as I was aware the only NMSS milestone should have been the date the project was completed. This was based on the information presented at the previous NMSS/RES quarterly meeting

AREVA fuel performance meeting: a large number of NRC staff are planning to attend this meeting: NRR – A. Proffitt, P. Clifford, A. Mendiola, M. Panicker, J. Kaiser, K. Heller; NRO – S. Lu, F. Forsaty, J. Donoghue, C. VanWert, M. Hayes; RES – J. Voglewede. NMSS would like to obtain all presentation materials from the meeting after the fact. Harold Scott asked if it is possible to ask the licensee ahead of time if we want them to discuss certain topics during these meetings. Such a thing is unlikely according to NRR and NRO mainly because the licensee often has some customers in the room, so they wish to be in control of the meeting as much as possible.

I intend to attend this meeting in Lynchburg. I am looking for the agenda to determine the extent of my participation

Upcoming travel and meetings: this topic was not discussed for lack of time, but trips are listed below:
a) Halden Program Group Meeting, Lyon, France, May 29-June 1, 2012 (M. Flanagan, P. Clifford)
b) ANS Annual Meeting, Chicago, IL, June 24-28, 2012 (M. Panicker, RE Einziger (will chair two sessions on EST), others from NMSS and RES to participate in the EST sessions.)
c) Top Fuel Meeting, Manchester, UK, September 2-6, 2012 (P. Raynaud, C. VanWert, P. Clifford, F. Forsaty)
Note: if NMSS sends a traveler, additional justification and paperwork will be needed. Currently no one from NMSS intends to attend.

Items for next TAG meeting

1- Past NEI and ESCP meetings in St Pete May 6th
2- Past IAEA SPAR-III meeting in Charlotte, May 14th
3- Current relicensing issues concerning fuel
4- New user needs items

From: Raynaud, Patrick
Sent: Wednesday, May 16, 2012 11:50 AM
To: Ahn, Tae; Clifford, Paul; Einziger, Robert; Flanagan, Michelle; Gordon, Matthew; Landry, Ralph; Panicker, Mathew; Patel, Amrit; Proffitt, Andrew; Scott, Harold; VanWert, Christopher; Voglewede, John; Wu, Shih-Liang
Cc: Lee, Richard; Mendiola, Anthony; Donoghue, Joseph; Guttmann, Jack; Pstrak, David
Subject: TAG-Fuels Meeting Minutes

Dear TAG-Fuels,

Attached are the minutes from our last meeting. Please let me know if there any gross inaccuracies, and I will make corrections.

Thanks,
Patrick

Patrick A.C. Raynaud, PhD
Reactor Systems Engineer (Fuels)
U.S. Nuclear Regulatory Commission
RES/DSA/FSCB
Mailstop: CSB-3A07M
Washington, DC 20555
Tel: (+1) 301-251-7542
patrick.raynaud@nrc.gov

82

From:	Campbell, Debbie
To:	Ahn, Tae; Alejano, Consuelo; Almoguera, Ramon; Alonso, Jose Manuel; Alsaed, Halim; Asano, Ryoji; Askarieh, Mehdi; Auzoux, Quentin; Bader, Sven; Baker, Steven; Ballinger, Ron; Barnabas, Istvan; Bateman, Mark; Behravesh, Mohamad; Bellamy, Steve; Bennett, John; Bernard, Felix; Bevilacqua, Arturo; Billone, Michael; Birk, Sandra; Bonano, Evaristo Jose; Bracey, William "Bill"; Brookmire, Tom; Brown, James; Bunt, Randy; Buschmann, Nancy; Butt, Darryl; Cairns, Martin; Cannell, Gary; Carlsen, Brett; Carter, Joe; Caseres, Leonardo; Channell, Clay; Cheng, Shih-Chung; Choi, Jongwon; Chung, Sunghwan; Coaster, Don; Codee, Hans; Cole, Kent; Conde, Jose-Manuel; Connell, James; Couplet, Damien; Cummings, Kris; Danker, William; Danner, Thomas; Darby, Sam; Davis, Demitrius; Dawson, Chris; Deboi, Kristi; Delannay, Michel; Di Gasbarro, Fernanda; Dobson, Alan; Duncan, Andrew; Dunn, Darrell; Dyck, Peter; Easton, Earl; Edwards, Steve; Einziger, Robert; Elwood, Randy; Engelvaart, Marco; England, Jeffery; Erhard, Anton; Farnum, Cathy Ottinger; Fernandez, Rene; Floyd, Mike; Francia, Lorenzo; Gago, Jose; Garamszeghy, Miklos (Mike); Geiser, Heinz; Gonzalez, Hipolito; Gonzalez, Rafael; Gordon, Matthew; Grahn, Per H.; Grant, Glenn; Graves, Herman; Greer, Bruce; Grizzi, Robert; Guimaraes, Maria; Gustems, Brian; Guttmann, Jack; Haddad, Roberto; Hanson, Brady; Heck, Matthias; Herrera Nevarro, Jose-Antonio; Hinojosa, Luis; Hodgson, Zara; Hollinger, Gary; Honjin, Masao; Hopf, Jim; Howard, Robert; Huang, Yuhao; Hueggenberg, Roland; Ishiko, Daiichi; Issard, Herve; Jacobs, Christian; James, Richard; Johnson, Lawrence; Jorgensen, Vern; Jubin, Bob; Jung, Dae-Il; Jung, Hundal "Andy"; Kadak, Andrew; Kapoor, Ashok; Katayama, Jiro; Kato, Masami; Kessler, John; King, Christine; Kitamura, Takafumi; Kokajko, Lawrence; Kook, Donghak; Kowalewski, Ron; Kuba, Stanislav; Kumano, Yumiko; Kumano, Yumiko; Kunerth, Dennis; Kuo, Roang-Ching
Cc:	Campbell, Debbie; Kessler, John
Subject:	2013 International High-Level Radioactive Waste Management Conference calls for papers (due 8 October 2012 - but can be extended a bit)
Date:	Friday, October 05, 2012 10:11:12 AM
Attachments:	Calls for Papers.pdf
Importance:	High

Sent from the desk of John Kessler, manager, Used Fuel and HLW Management Program…

Dear Extended Storage Collaboration Program members:

You may wish to submit abstracts to the upcoming International High-Level Radioactive Waste Management Conference to be held in Albuquerque NM 28 April to 2 May 2013. While abstracts are due next week for this conference, I suspect the conference organizers will allow an extension for late abstracts.

2013 International High-Level Radioactive Waste Management Conference

Attached is the call for papers. I understand that 15 abstracts in the long-term storage area have already been submitted. I am planning to submit an abstract describing the ESCP program and progress and one of the proposed full-scale, high burnup extended storage demonstration programs. I was also thinking of the following issues about which some of you may wish to submit an abstract:

- "International Atomic Energy Agency Programs Related to Long-Term Used Fuel Storage" (authors: Bevilacqua, Danker)
- ENRESA or one of the Spanish organizations could submit an abstract on the Spanish centralized storage facility
- Gap analyses by any of the groups that have performed one
- Experimental or field study results (we are submitting one on the inspection of the SS dry storage canister at Calvert Cliffs)

E87

- Description of national programs, laws, regulations related to extended storage
- Specific degradation mechanism(s). This could be produced by some of the ESCP subcommittees.

Let me know if you think you would want to submit an abstract on the above topics – or others. The level of effort should be fairly small as ANS only wants 250 to 500 words. If you are interested, but don't have the time to submit and abstract, let me know, and I can try to help put one together for you. I can also help you upload it onto the ANS web site if you wish.

Thanks.

John

John Kessler
Manager, Used Fuel and HLW Management Program
1300 West WT Harris Blvd.
Charlotte NC 28262

Work: +1-704-595-2737
Fax: +1-704-595-2860

CALL FOR PAPERS - Abstract Deadline: September 28, 2012

2013 International High-Level Radioactive Waste Management Conference

April 28 - May 2, 2013 · Albuquerque, New Mexico, USA · Albuquerque Marriott

"Integrating Storage, Transportation, and Disposal"

CONFERENCE PURPOSE:

The conference is a forum for the discussion of the scientific, technical, social and regulatory aspects of the "back end" of the nuclear fuel cycle. These issues include waste generation, transportation, storage, treatment, disposal, and associated aspects (such as facility remediation, regulation, and stakeholder involvement). The conference is an opportunity for an exchange of information on current topics of interest among the international participants in nuclear-waste activities. The conference will appeal to an international audience as an opportunity to share information across programmatic, disciplinary, and international boundaries. Intended participants and audiences include personnel working on all aspects of irradiated fuel and HLW management such as: geologic waste-disposal systems, interim storage systems, spent nuclear fuel reprocessing systems, transportation systems, facility remediation systems, the governmental and private organizations using these systems, regulators, and those involved in scientific and societal issues related to policy questions for these systems. Conference participants are encouraged to focus their submissions (either as oral or poster presentations) on the theme of this conference.

SPONSOR:

Sponsor: American Nuclear Society.

Cooperation is expected from numerous professional and technical societies, national laboratories, federal agencies, and commercial organizations throughout the world.

TECHNICAL PROGRAM CHAIRS:

Kevin McMahon, *Sandia National Laboratories*
Barry Butterfield, *HDR Inc.*

PAPER ACCEPTANCE CRITERIA:

Papers are expected to contain descriptions of work that is new, significant, and relevant to the conference purposes. Both abstracts and full papers will be reviewed prior to acceptance. Submissions should contain new data and investigations in scientific or program areas that are of general interest, address problems of interdisciplinary significance, or include in-depth discussions of scientific and technical issues related to public-policy questions.

Criteria for selection include originality of work, relevance of topic, validity of method, clarity and conciseness of communication, and adherence to the scientific method (if appropriate). Compliance with content and length guidelines (following) are also part of the acceptance requirements. Both abstracts and full papers must be submitted electronically to www.ans.org/meetings/ihlrwm. Papers may be submitted for oral or poster presentation; papers may be designated for submission to a refereed journal. All submissions must be in English.

ELECTRONIC SUBMISSIONS:

To submit a paper electronically, please refer to the detailed instructions available on the Internet at: www.ans.org/meetings/ihlrwm.

INSTRUCTIONS TO AUTHORS
FORMAT OF ABSTRACT FOR REVIEW:

1. Abstracts must be submitted electronically in ASCII text, HTML, Word, WordPerfect, and/or PDF (Adobe Acrobat) format.
2. Use SI units (with English units following in parenthesis, if desired). Exceptions are made for ev and barns.
3. List references numerically at the end of the abstract, and use numbers in the text, enclosed within brackets.
4. If using the ASCII text of HTML format, please include tables or figures in GIF or JPEG format. Also, please upload your original source document for use in the printed program, if available.

PLEASE NOTE:

- The title of your abstract will be used as the title of your presentation in the preliminary program.
- Authors of accepted papers will be expected to register for the conference. There are no funds available in the conference budget to support travel fees or complimentary conference registration.

ABSTRACT LENGTH:

1. Title Maximum - 10 words.
2. Text Minimum - 250 words.
3. Text Maximum - 500 words.
4. Figures and Tables - One figure and/or table maximum.

CONTENT:

The contents of the abstract must include the objectives of the study/investigation and the methodology used. It should also briefly describe the main findings and their potential applications. Sufficient information should be included for an independent reviewer to determine its suitability for the conference.

DEADLINE:

Your abstract must be submitted electronically no later than September 28, 2012, in order to ensure that it is included in the review process.

AUTHOR'S ORGANIZATIONAL APPROVAL:

- All internal reviews and organization approvals must be completed prior to submittal of the final paper.
- It is the responsibility of the author to protect proprietary information.

CALL FOR PAPERS - Abstract Deadline: September 28, 2012

2013 International High-Level Radioactive Waste Management Conference

PAPER PREPARATION FOR PUBLICATION IN CONFERENCE PROCEEDINGS

IMPORTANT INFORMATION:

- *Accepted papers will be included in the CD-ROM Proceedings that will be distributed at the beginning of the conference.*
- After the full paper review is completed by the Technical Program Committee, authors of accepted papers will receive information for preparation of final papers in camera-ready format via email.
- Authors of accepted papers will be allowed 8 pages for publication at no charge. Authors who exceed the 8 page limit will be billed a per-page charge of $150.
- All type and illustrations should appear within designated margins—dimensions are 7 in. (178 mm) by 9 in. (229 mm). We recommend 10-point type with 12 points of leading (spacing between lines). Use Times Roman typeface or an equivalent.
- Indent each paragraph 1/4 inch (use tab; do not use the space bar to indent). Single-space your text in two-column format. Your equations, figures, and tables do not need to comply with the two-column format. In other words, equations, figures, and tables may span the columns.
- Changes to accepted papers must be limited to revisions or changes requested by the Technical Program Committee.

IMPORTANT DATES:

· Abstract Deadline:	September 28, 2012
· Notification to Authors:	October 15, 2012
· Full Draft Papers Due for Review:	December 14, 2012
· Notification to Authors:	January 14, 2013
· Final Paper for CD-ROM Publication*:	February 22, 2013
· Early Registration Deadline:	April 5, 2013
· Conference:	April 28 - May 2, 2013

*Full paper revisions must be submitted electronically to www.ans.org/meetings/ihlrwm. Full papers will be reviewed.

The ANS Scientific Publications Department may be contacted at the address and phone number below:

AMERICAN NUCLEAR SOCIETY
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SUBJECT CATEGORIES FOR ABSTRACTS:

1. Total Repository System (Generic and Site-Specific)
· Site Selection Criteria
· Post-Closure Safety Assessment
· Interface Between Subsystems
· Modeling Near Field and Far Field Interactions
· Sensitivity Analyses
· Uncertainty Management and Confidence Building
· Performance Demonstration, Confirmation and Safety Research
· Definition, Preparation, and Documentation of a Safety Case
· Safety Case and Regulatory Reviews
· Alternate Lines of Arguments
· Data and Information Systems

2. Natural Systems for Disposal (Generic and Site-Specific)
· Site Characterization Techniques
· In-Situ Measurement of Properties and Their Scaling
· Hydrologic, Chemical, Thermal, and Mechanical Processes
· Seismic, Volcanic, and Tectonic Processes
· Climate, Environmental, and Natural Hydrogeologic Process Modeling
· Natural Analog Studies
· Studies in Underground Research Facilities
· Geo-scientific Data Synthesis

3. Engineered Systems for Disposal
· Surface and Underground Facilities
· Waste Handling, Storage, and Emplacement at Disposal Facility
· Engineered Barriers (e.g., Waste Package, Backfill, etc.) Design and Performance
· Near-Field Environment Modeling
· Thermal Load Management
· Pre-Closure Operational Issues (safety, QA/QC, constructability)

4. Biosphere
· Defining Generic and Site-Specific Biosphere Characteristics
· Estimating Impact on Environment
· Pathway Analysis and Dose Modeling
· Exposure Scenarios

5. Regulatory Topics
· International, National and Sub-national Regulations, Requirements and Guidance
· Prescriptive versus Risk-Informed Regulations
· Time Scales, Safety Measures, and Confidence Measures
· Safety Margins and Statement of Confidence
· Licensing and Hearing Processes and Procedures
· Quality Assurance, Quality Control, and Inspections

6. Institutional Topics (With Emphasis on Lessons Learned)
· Roles and Relationships of Sub-national Regulatory and Oversight Authorities
· International Successes with High-Level Waste Management
· Stakeholder Confidence Building/ Techniques of Public Involvement
· Risk Perception, Public Communications, and Media Coverage
· Institutional Issues in Site Selection
· Site Selection Strategies for Storage and Disposal Facilities
· National Programs and Policies
· Retrievability and Reversibility
· Alternative Institutional Structures

7. Storage of Used Nuclear Fuel and High Level Waste
· Long-term (>60 years) Storage
· Dry and Wet Storage
· High Burn-up and Mixed Oxide Spent Nuclear Fuel
· Integrated Safety Analysis
· Developing Consent Based Approaches for Siting
· Site Specific vs. Regional vs. Centralized Storage
· Waste Management Systems Analysis
· Options for Direct Disposal of Storage Canisters

8. Advanced Fuel Cycles: Impacts on Waste Management
· Fuel Cycle Modeling
· Radionuclide Inventories and Waste Forms
· Waste Management Impacts from Reprocessing
· Fuel Cycle and Waste Management System Optimization

9. High-level Radioactive Waste Transportation
· Cask Integrity Analysis and Testing
· Transportation Risk (Rail, Road, and Marine)
· High Burn-up and Mixed Oxide Spent Nuclear Fuel Transportation

10. Security, Safeguards, and Non-Proliferation
· Implementing Non-Proliferation and Security Measures
· Transportation Safety and Safeguards
· Multinational Cooperation in Waste Management

11. Emerging Issues in Waste Management
· Large Volume Cleanup Waste
· Damaged Fuel and Core Waste
· Specialized Repositories

83

Einziger, Robert

From: Voglewede, John
Sent: Thursday, October 18, 2012 2:35 PM
To: Einziger, Robert
Subject: RE: Rod Internal Gas Pressure

I thought you could have written more.

What I thought you might say was:

- Yes, that is exactly what I am looking for.
- Yes, but too bad it's MOX
- Yes, but NMSS no longer has high interest in this lower priority issue.
- Yes, I was surprised that helium absorption continues to such high burnup.
- No, I am more interested in post-irradiation fuel swelling and its impact on cladding stresses.

From: Einziger, Robert
Sent: Thursday, October 18, 2012 1:52 PM
To: Voglewede, John
Subject: RE: Rod Internal Gas Pressure

Thanks. Yes

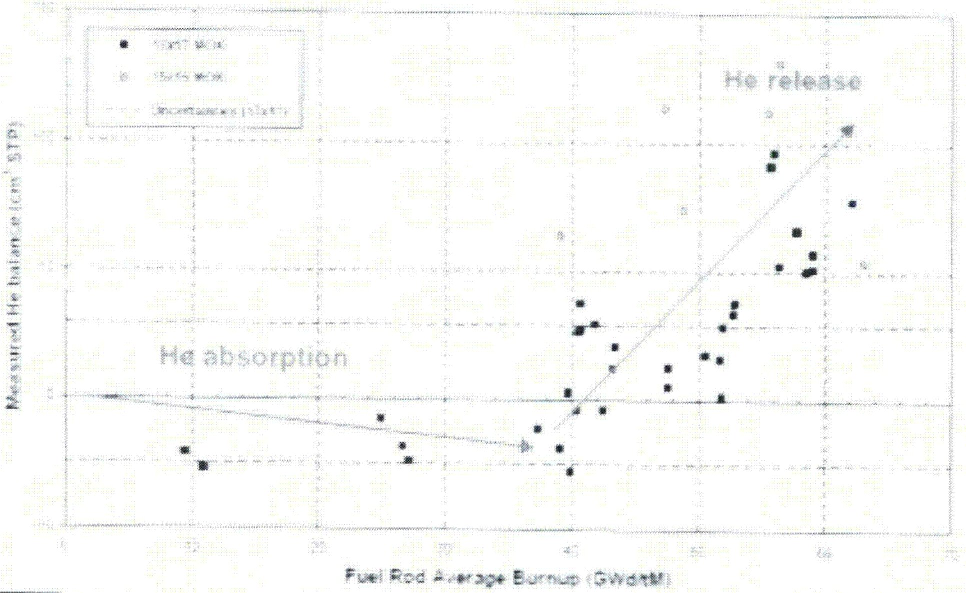
From: Voglewede, John
Sent: Thursday, October 18, 2012 1:49 PM
To: Einziger, Robert
Cc: Lee, Richard
Subject: Rod Internal Gas Pressure

Bob,

On Tuesday, I attended a proprietary meeting with AREVA on MOX fuel, in which the following figure was shown:

E88

- **Helium production**
  - α decay of some of the actinides (~ 65%)
  - (n, α) reaction on ^{16}O (~ 25%)
  - Ternary fissions (~ 10%)
- **Helium balance** (final – initial He volume)



The mention of helium production due to decay of actinides made me think of your interest in sources of stress on the cladding after irradiation.

We are able to predict noble gas inventory in the rod and how it changes for long decay times. Estimates of release to gap can also be made. In my mind, this is the major source of any <u>change</u> in the load on the cladding after discharge. Are you interested in such information?

John

84

From: Campbell, Debbie
To: Ahn, Tae; Alejano, Consuelo; Almoquera, Ramon; Alonso, Jose Manuel; Alsaed, Halim; Asano, Ryoji; Askarieh, Mehdi; Auzoux, Quentin; Bader, Sven; Baker, Steven; Ballinger, Ron; Barnabas, Istvan; Bateman, Mark; Behravesh, Mohamad; Bellamy, Steve; Bennett, John; Bernard, Felix; Bevilacqua, Arturo; Billone, Michael; Birk, Sandra; Black, Bradley; Bonano, Evaristo Jose; Bracey, William "Bill"; Brookmire, Tom; Brown, James; Bryan, Charles; Bunt, Randy; Buschmann, Nancy; Butt, Darryl; Cairns, Martin; Cannell, Gary; Carlsen, Brett; Carter, Joe; Caseres, Leonardo; Channell, Clay; Cheng, Shih-Chung; Choi, Jongwon; Chung, Sunghwan; Coaster, Don; Codee, Hans; Cole, Kent; Conde, Jose-Manuel; Connell, James; Couplet, Damien; Cowan, Pamela B.; Cummings, Kris; Danker, William; Danner, Thomas; Darby, Sam; Davis, Demitrius; Dawson, Chris; Deboi, Kristi; Delannay, Michel; DePaula, Sara; Di Gasbarro, Fernanda; Dobson, Alan; Doering, Thomas; Duncan, Andrew; Dunn, Darrell; DuPont, Mark; Dyck, Peter; Easton, Earl; Edwards, Steve; Einziger, Robert; Elwood, Randy; Engelvaart, Marco; England, Jeffery; Enos, David; Erhard, Anton; Farnum, Cathy Ottinger; Fernandez, Rene; Ferry, Sarah; Floyd, Mike; Francia, Lorenzo; Friant, Carl "Lee"; Gago, Jose; Garamszeghy, Miklos (Mike); Garg, Kris; Geiser, Heinz; Gonzalez, Hipolito; Gonzalez, Rafael; Gordon, Matthew; Grahn, Per H.; Grant, Glenn; Graves, Herman; Greer, Bruce; Grizzi, Robert; Guimaraes, Maria; Gustems, Brian; Guttmann, Jack; Haddad, Roberto; Hanson, Brady; He, Xihua; Heck, Matthias; Herrera Nevarro, Jose-Antonio; Hinojosa, Luis; Hodgson, Zara; Hollinger, Gary; Honjin, Masao; Hopf, Jim; Howard, Robert; Huang, Yuhao; Hueggenberg, Roland; Ishiko, Daiichi; Issard, Herve; Jacobs, Christian; James, Richard; Johnson, Lawrence; Jorgensen, Vern; Jubin, Bob; Jung, Dae-Il; Jung, Hundal "Andy"; Kadak, Andrew; Kapoor, Ashok; Katayama, Jiro; Kato, Masami; Kessler, John; King, Christine; Kitamura, Takafumi; Kokajko, Lawrence; Kook, Donghak; Kowalewski, Ron; Kuba, Stanislav; Kumano, Yumiko; Kumano, Yumiko; Kunerth, Dennis; Kuo, Roang-Ching
Cc: Campbell, Debbie; Waldrop, Keith
Subject: 2012 ESCP Meeting
Date: Friday, October 26, 2012 1:46:34 PM
Attachments: ESCP Nov 2012 - Schedule.docx

From the desk of Keith Waldrop…

Dear ESCP Members,
We are preparing the agenda for next month's ESCP meeting in Charlotte, NC at the Hilton Executive Park.

We have decided to shorten the full ESCP meeting and provide an extended session for the Marine Environment and Hi Burnup Demo subcommittees to meet most of the day on Tuesday. We will hold the planned breakout sessions from 3:30 to 6:00 p.m. for the Steering, NDE, Marine Environment, and Hi BU Demo subcommittees to meet. The Fuels and Concrete subcommittees are not meeting and the International subcommittee will meet on Friday morning. Please note that due to space limitations at the Hilton the subcommittee meetings scheduled for 3:30-6:00 p.m. on Tuesday, 27 November, will be held at the Marriott located directly across the parking lot from the Hilton. I have included a schedule to summarize.

In preparing the agenda, I am looking for presentations on collaboration opportunities that would be useful information to the ESCP members; research you've done recently or are doing in the next year. Given that we've shortened the full ESCP meeting, we need to make the best use of this time. Please **respond by Wednesday, 7 November,** with presentation topics for the agenda.

Lastly, a reminder that **TODAY** is the deadline for the early bird registration for the US Nuclear Used Fuel Strategy Conference sponsored by Nuclear Energy Insider that will take place on Wednesday and Thursday, 28 – 29 November 2012, at the same location for those that may wish to also attend that conference. The Early Bird discount is $200. It is my understanding that Nuclear Energy Insider has set up a discount code for ESCP members "EPRI100" for $500 off. The link for the conference is: www.nuclearenergyinsider.com/used-fuel-strategy-conference/. Again, EPRI is not endorsing Nuclear Energy Insider's conference; we are merely sharing the information to make it convenient for our ESCP members if they wish to attend.

E89

Keith Waldrop, Senior Project Manager
Used Fuel – HLW
704-595-2887

ELECTRIC POWER RESEARCH INSTITUTE

SCHEDULE

November 2012

Extended Storage Collaboration Program

Hilton Executive Park Drive 5624 Westpark Drive Charlotte, NC 28217

26 – 27, 30 November 2012

DATE: MONDAY, 26 NOVEMBER 2012		
TIME	**MEETING**	**LOCATION**
7:30 a.m.- 8:30 a.m.	Continental Breakfast	Hilton - Charlotte Ballroom
8:00 a.m.- 5:00 p.m.	Full ESCP Meeting	Hilton - Charlotte Ballroom
DATE: TUESDAY, 27 NOVEMBER 2012		
TIME	**MEETING**	
7:30 a.m.- 8:30 a.m.	Continental Breakfast	Hilton - Charlotte Ballroom
8:00 a.m.- 9:00 a.m.	Full ESCP Meeting	Hilton - Charlotte Ballroom
9:00 a.m.- 3:00 p.m. (concurrent sessions)	Marine Environment Subcommittee High Burnup Demo Subcommittee	Hilton - Charlotte Ballroom
3:30 p.m.- 6:00 p.m. (concurrent sessions)	Steering Committee Marine Environment Subcommittee High Burnup Demo Subcommittee NDE Subcommittee	**Marriott** – (Emerald, Magnolia, Dogwood, Azalea; room assignments TBD)
DATE: FRIDAY, 30 NOVEMBER 2012		
TIME	**MEETING**	
7:30 a.m.- 8:30 a.m.	Continental Breakfast	Hilton – Myers Park Room
8:00 a.m.- noon	International Subcommittee	Hilton – Myers Park Room

1% cladd strain

Ahn, Tae

From: Ahn, Tae
Sent: Friday, November 02, 2012 3:53 PM
To: Raynaud, Patrick
Subject: RE: Meeting on Cladding Stress

Great! I will see you at 12:30 in EBB 2A13. My office is EBB2C32.

From: Raynaud, Patrick
Sent: Friday, November 02, 2012 3:20 PM
To: Ahn, Tae
Subject: RE: Meeting on Cladding Stress

Tae, I suggest we meet in person on Monday before the all staff meeting, to make sure we understand each other and so I can provide you better answers. I could be at EBB around 12:30, then that gives us time to meet and then go to all employees meeting at 1:30 at the Marriott.
Are you available then?
Patrick

From: Ahn, Tae
Sent: Friday, November 02, 2012 2:06 PM
To: Raynaud, Patrick
Subject: RE: Meeting on Cladding Stress

Thanks, Patrick. Sorry. Probably I did not know how to ask. I have added a couple of more below to be closer to what may be important for us.

From: Raynaud, Patrick
Sent: Friday, November 02, 2012 12:15 PM
To: Ahn, Tae
Subject: RE: Meeting on Cladding Stress

Tae,

Here are answers to your questions:

Assuming you mean pressure of the water around the fuel rods when you talk about back water pressure, the thermal expansion is NOT controlled (this means that no restriction to fuel diameter change, including both thermal and radiation expansion. The radiation expansion seems to be more severe with higher burnup) by the coolant pressure. The cladding overall deformation does depend in part on the coolant pressure, but this is not the only parameter that has an impact on cladding deformation – any quantitative information?.

I will ask about while I am at Studsvik. No problem – it will be very informative for us. We may add the pellet microcracks. I see very different pictures from two sources – one are more longitudinal cracks and another are more circumferential cracks.

I am not sure this answers your question completely, but feel free to ask more. I will be quicker to respond next time.

Patrick

From: Ahn, Tae
Sent: Thursday, November 01, 2012 11:21 AM

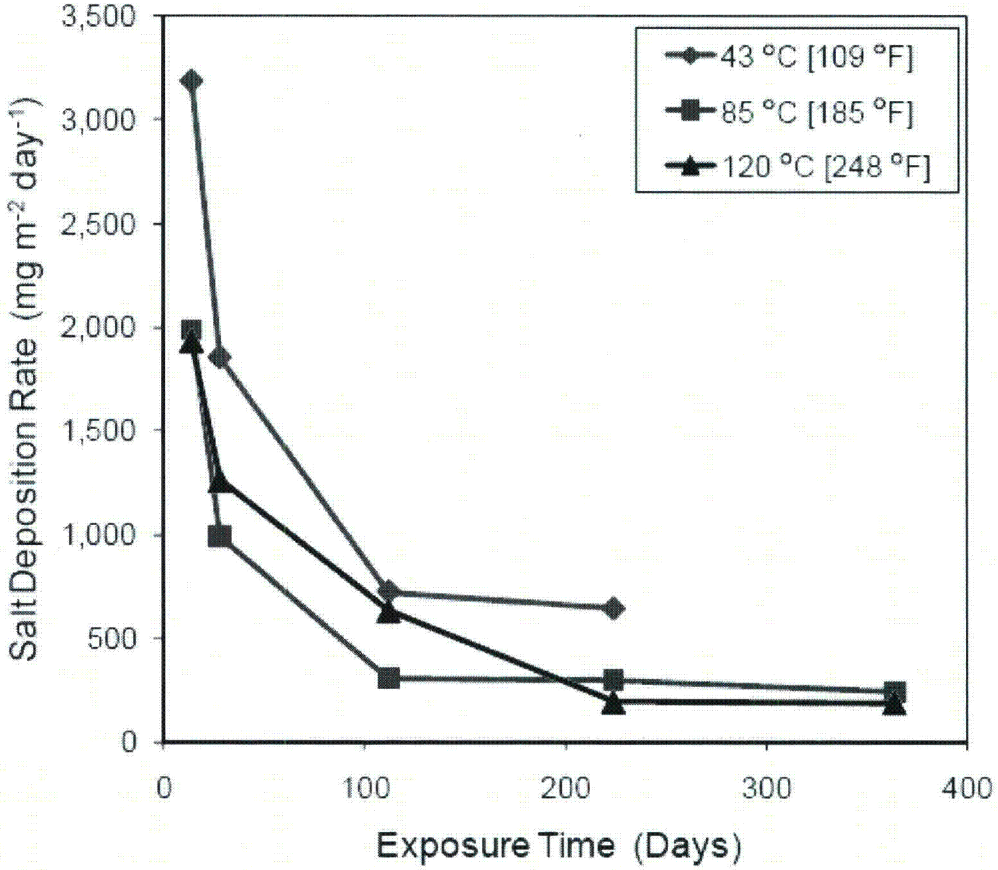
E90

To: Raynaud, Patrick
Subject: Meeting on Cladding Stress

Patrick, Sandy delayed our meeting. I will be preparing the paper this week and next week. Two questions for you:

(1) Is or isn't the thermal expansion controlled by back water pressure?
(2) In the SCIP meeting, could you meet and talk to the gentleman whom Michelle mentioned via Anna Marie regarding PIE data on cladding gap?

If you need better explanation, please give me a call. Thanks. Tae



43 °C [109 °F]
85 °C [185 °F]
120 °C [248 °F]
Salt Deposition Rate (mg m-2 day-1)
3,500
3,000
2,500
2,000
1,500
1,000
500
0
0
100
200
300
400
Exposure Time (Days)

New Comments to the Paper on Radiation-Induced Swelling of Spent Fuel Matrix

(11/16/12, T Cao)

Reply 11/22/2012, V.V. Rondinella (only concerning the alpha-decay effects and alpha-doped studies)

In my comments of October 3, I indicated that (1) the radiation-induced swelling of spent fuel is mostly not from the helium production of alpha decay. Instead, it is from the creation of uranium vacancies. (2) Part of the swelling may be recovered automatically during the high storage temperature time period. I found these comments seem having been taken care of in the current version. The following are some new comments.

Firstly, thank you very much for the stimulating comments!

It is indicated in this version that "potential thermal annealing of the swelling was considered (Rondinella, et al., 2011), which may reduce the lattice parameter with time. Under current storage conditions of time and temperature, the annealing appears to be insignificant."

A new formulation of the potential thermal annealing impact is in the text

_But Weber (1981, equation (3) on page 207) derived the following equation for the dose dependent change in lattice parameter.

$$\frac{\Delta a}{a_0} = \frac{N_d K_\alpha \Delta v_F}{3B}(1 - e^{-BD_\alpha})$$

In this equation, B is the annealing rate and D_α the alpha dose (α / cm^2). Weber (1981) indicated that the exponential form arises because at sufficiently long times the defect annealing rate becomes comparable to the defect production rate as discussed by Nellis (1977). This equation can explain the lattice parameter saturation shown in Fig. 1. This annealing process is not discussed in the paper and I think it may be important to the applicability of using the alpha-doped sample simulating radiation-induced swelling of spent fuel. This defect annealing does not depend on the high temperature but on alpha dose in long times. Here the time difference between long term storage of the spent fuel and the short time period experiment of high alpha-doped sample swelling will have different impact on the annealing process. The short time alpha-doped sample experiments do not have the defect annealing and may over estimate the swelling of the long term stored spent fuel.

The annealing processes associated with defect recombination are actually well reflected on the curve in Fig 1, which shows a very evident saturation behaviour. The saturation trend is commonly observed in this type of experiments, and affects more or less all properties studied. It is not explicitly discussed in the paper because of length issues. There is no reason to think that "more annealing" would occur in SNF compared to this alpha-doped UO2, as long as power effects do not cause temperature to achieve thermal annealing levels. Furthermore, the issue of possible "dose-rate" effects when using accelerated conditions has been specifically addressed in previous work: the conclusion was that within the specific alpha-activity levels used in the alpha-doped UO2 shown in fig 1 there are no artifacts caused by the accelerated conditions. The real issue to address to reach a conclusion on the representativeness of alpha-doped samples vs actual SNF for the swelling behaviour (for other properties we have already performed a comparison) is the different structure and morphology of the 2 materials. SNF has more sinks available for the He, and more pathways for it to be released out of the matrix. This could be a reason to limit the He-induced component of the decay swelling (see point 2) below); the lattice defect swelling component is more likely to be the same in SNF and alpha-doped.

Further remarks:

1) A saturation at 0.4 or 0.5% swelling constitutes a non negligible potential contribution to stress. It is important to verify if such swelling levels are to be expected in SNF

2) An important aspect to be taken into account is that in addition to the production of defects, which are affected by the recombination mechanisms ultimately causing the saturation, helium is generated. We know that up to a few dpa He is retained in the matrix of unirradiated UO2. He is a noble gas, with a relatively low solubility in UO2. Up to the level where it starts being released from the matrix it constitutes a "non recombining" source of additional swelling. We have seen that nm sized He bubbles form in UO2 at a certain dose level. We also think that He could well be responsible for the fact that saturation of the swelling occurs at slightly higher levels than reported for other experiments on actinide oxides. This is a feature we are currently paying specific attention to, and we hope to come up with some additional data soon.

3) Given the uncertainties on the duration of dry storage, and the evolution towards higher burnup in SNF, it is not clear up to which dpa range we should focus the investigation. If we have to go more towards ~10 dpa, for instance, we can expect additional processes to kick in. Even the defect production – recombination "equilibrium" may disappear if more significant restructuring processes start to become active

It seems that three conditions tied together are necessary for generating high stress in the spent fuel cladding: (1) the swelling due to alpha decay (lattice parameter increase), (2) not enough annealing to reduce the swelling, and (3) not enough gap between fuel matrix and cladding to accommodate the swelling. The current paper presented solid evidence from the alpha-doped sample experiments to demonstrate condition (1) above. But conditions (2) and (3) are not well established yet, especially condition (3) with very little data. My comment to use Weber's formula is trying to have more discussions on condition (2).

Concerning (3), we know that the gap is closed starting at 40-45 GWD/t. I added some sentences on this in the text. I also mentioned the additional (benign) impact of the high burnup structure which characterize the outer periphery of the fuel pellet.

I also suggest the authors to improve the writing. The following are some places I have found revisions are clearly needed.

It is not clear about the complementary measurements (on page 2), which support Figure 1: (i) the same trends as for the lattice parameter are observed by monitoring hardness and thermal diffusivity; and (ii) alpha-doped samples with 100 times. I think a bit more description is needed about how the hardness and thermal diffusivity are related to lattice parameter. The 100 times of what has to be defined. There is some additional text with references addressing the other properties.

On page 4, the three cases are not clearly defined.

On page 4, the stress in equation (1) needs to be explained or defined more specifically. You adopted Retal et al.'s (2004) thermal stress calculation to get the stress from alpha decay induced swelling. Retal et al.'s (2004) calculation is mostly for the hoop stress in the cladding as you quoted in your equation (2). It is a bit different from the stress in your equation (1). Actually, it seems to me that you did not use equation (1) at all in this study. You only need the swelling (0.3%) to get the hoop stress from Retal et al.'s (2004) calculation.

On page 6, the sentence "An observation made after storage for 20 year in Japan (Sasahara and Matsumura, 2008)." is not complete. This was addressed, too.

In the summary, I suggest you include the uncertainty discussion, especially on the lacking of gap size data.

88

Ahn, Tae

From: Ahn, Tae
Sent: Friday, November 30, 2012 10:24 AM
To: '김영석'; '최종원'; '정찬우'; '이관희'
Cc: Guttmann, Jack; Smith, Shawn; Rubenstone, James
Subject: Proposed Agenda, T. Ahn of NRC Visit KAERI and KINS, January 28 to February 1, 2013

Below is my proposed agenda. Please review and comment/modify. Thanks.

AhnT -- Ahn presentation

January 28, KAERI

Morning:

- Chloride-Induced Stress Corrosion Cracking of Austenitic Stainless Steel for Dry Storage of Spent Nuclear Fuel (AhnT)
- KAERI Tests on the same topic (KimY)

- Corrosion (general corrosion, localized corrosion and SCC) of Canister (or Container) in Nuclear Waste Management: Perspective and Model Abstraction (AhnT)
- SCC Mechanisms (KimY)

Afternoon:

- Potential Stress on Cladding Imposed by the Matrix Swelling from Alpha Decays in High Burnup Spent Nuclear Fuel (AhnT)
- DHC Theory Update and Path Forward (KimY)

- NRC Needs for Data on Irradiated Spacer Grid Material Properties (AhnT)
- KAERI Tests on the same topic (AhnS)

January 29, KAERI

Morning:

- Representations of Corrosion of Copper and Carbon Steel and Waste Form Degradation under Normal and Disruptive Conditions at Generic Disposal Site: NRC Scoping of Options and Analyzing Risk (SOAR) Model (AhnT)
- KAERI Corrosion Tests of Disposal Container (ChoiJ)
- KAERI Performance Assessment of Disposal Container and Waste Form (ChoiJ)

V ok
(2-4)

Afternoon:

- Dissolution of SIMFUEL Update (AhnT)
- SIMFUEL Fabrication for High Burnup Spent Nuclear Fuel (ChoiJ)

- Source Term Analysis in Handling Canister-Based Spent Nuclear Fuel with PCSA Tool (AhnT)
- KAERI Modeling and Code Exercise on the related topic (ChoiJ)

January 30, KINS

- U.S. Nuclear Waste Management and NRC's Integrated Research Activities (AhnT)
- KINS Status on Storage and Disposal (JeongC)
- Selected Topics Presented at KAERI

E93

January 31, Wolsong Site

- Update on Monitoring and Environmental Conditions of CANDU Storage Site (LeeK)
- Update on Wolsong Disposal Site Excavation (LeeK)
- Selected Topics Presented at KINS and KAERI

Addition: there will short visit to KAIST on February 1 and SNU on February 2, during travels.

Ahn, Tae

From: Ahn, Tae
Sent: Wednesday, December 12, 2012 11:17 AM
To: '국동학'
Cc: 최종원; '김영석'
Subject: RE: Proposed Agenda Update
Attachments: BIO_Ahn_12 2012_Modified.docx

Dear Dr. Kook:

Some comments reflecting my initial desires with respect to JSCNEC Work Plan:

1. On 1/28, the first two SCC presentations will be – (i) recent my work on uncertainty analyses associated with SCC issues in storage, including NRC/CNWRA tests results. The slides were presented at the Electrochemical Society and the paper has been submitted, and (ii) KAERI updated test data, after the last NEI meeting.
2. If we have a meeting in the afternoon, 1/28, we may discuss SCC mechanism (KimY, if he likes) and SCC damage mechanism (AhnT – it will be presented in two 2013 forums. I will go over related slides). We did some preliminary discussion on SCC mechanisms during the meeting in DC this month. Altogether about hour will do.
3. On 1/28, "Data Need for Spacer Grid," I do not work on it. I will communicate with Dr. AhnS for our NRC TAG (Technical Advisory Group)-Fuel where I am a member. The issues in reactor and spent fuel management are briefly addressed in JSCNEC Work Plan. My introduction will take only 10 minutes. I have reviewed limited Dr. Ahn's internal documents.
4. On 1/29, "Source Term Analysis in Canister" is incorrect title and may move to *Monday afternoon*. My work is written in NRC ADAMS for your access, and currently submitted to a journal. I would like to go over it. It is about risk associated with handling canistered-spent fuel during the storage or during the pre-closure period of disposal. I have focused on release fraction of radionuclides from breaches of canister and cladding and from spent fuel degradation. In addition to the analysis of material properties I used PCSA Tool for the consequence. I understand KAERI worked with CNWRA to develop a similar code based on PCSA Tool. If someone at KAERI (perhaps, Dr. Jeong Jong Tae?) did code exercises, I would like to hear about them too.

During the discussion with KAERI for this trip, I have recognized that my background experience is not well informed to pursue the JSCNEC Work Plan. Therefore, I gave a short BIO recently (which was used for IAEA previously) to Dr. Kim in DC and, thereafter, modified slightly. Attached is for your information. After we finalize the agenda, I would like to inform it to other organizations if you don't mind. Thank you. Tae M. Ahn

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From: 국동학 [mailto:syskook@kaeri.re.kr]
Sent: Wednesday, December 12, 2012 1:51 AM
To: Ahn, Tae
Cc: 최종원
Subject: Proposed Agenda Update

Dear Dr. Ahn

It is very nice to hear your visit to KAERI again and

your high interest in SF demo.

Trip to Korea

Tae M. Ahn, NMSS/SFAS

Date:

Organization: Korea Atomic Energy Research Institute (KAERI) and

Korea Institute of Nuclear Safety (KINS)

Purpose: Implement part of WORL PLAN WITH KOREA (attached)

Outcome: Exchange assessment/research results on various topics associated with storage and disposal of spent nuclear fuel and high-level waste (HLW). This trip will focus more on materials issues.

Process:

Travel to Korea – January 26 and 27, 2013 (may include January 25 if a rest day is allowed)

KAERI, Daejeon – January 28, 2013

- Dr. Young Suk (YS) Kim and his coworkers - cladding (force and hydrogen effects)

  and storage canister (cracking and corrosion)

- Dr. Jong Won Choi and his staff – container corrosion, hot cell testing and SIMFUEL testing (include Drs. Sang Bok Ahn and Young Suk Kim, and SIMFUEL expert)

KINS, Daejeon – January 29, 2013

- Dr. Chan Woo Jeong and his staff – KINS requests and potential Wolsong site visit

KAERI and KINS, Gyeongju – January 30, 2013

- KAERI with (potentially with KINS) – Wolsong site visit on corrosion and Wolsong data on excavation

Travel to U.S. – January 31 and February 1, 2013 (may include one day AL)

Need to make calls to discuss details, attaché work plan, determine date (consider other meetings and paper dues).

90

Einziger, Robert

From: Waldrop, Keith <kwaldrop@epri.com>
Sent: Thursday, January 03, 2013 7:49 AM
To: BADER Sven (AREVA); Einziger, Robert; raheel.haroon@areva.com; Martin, Zita I; Pfeifer, Holger; peter.stefanovic@constellation.com; 'Wagner, John C.'; Zigh, Ghani
Cc: Billone, Michael C.
Subject: FW: EPRI ESCP Fuels-Internals Subcommittee

Importance: High

I am forwarding this message from Mike Billone to members of the ESCP Fuels Subcommittee that Mike did not have on the initial distribution. I apologize for the delay in sending it.
Mike is now looking at having a conference call in the latter half of January (as opposed to next week as in his note below), so please respond to his request.

Thanks,
Keith Waldrop

From: Billone, Michael C. [mailto:billone@anl.gov]
Sent: Tuesday, December 18, 2012 4:05 PM
To: Bader, Samuel D.; Kris Cummings (cumminkw@westinghouse.com); Brady Hanson (brady.hanson@pnl.gov); Donghak Kook (syskook@kaeri.re.kr); Miriam Lloret (mll@enusa.es)
Cc: Campbell, Debbie; Kessler, John; Waldrop, Keith; Ken Sorenson (kbsoren@sandia.gov)
Subject: EPRI ESCP Fuels-Internals Subcommittee
Importance: High

Dear Colleagues,

At the recent ESCP meeting in Charlotte, Brady and I agreed to switch roles with me to become the chair and Brady to remain a member of the Fuels/Internals Subcommittee.
Also, John Kessler agreed to provide a list of publicly available EPRI documents on dry-cask storage and transport, as well as arranging a conference phone call in early January 2013 to allow us to ask questions regarding data trends determined by EPRI and its partners. Although such data are proprietary, EPRI may be able to share information and experience with regard to test condition ranges for past work, test condition ranges for future work, and data trends.

We have had at least 2 meetings and 1 conference phone call.
Early on we highlighted some issues that we thought we could make progress on:

1. More realistic fuel drying and storage temperatures for casks with high-burnup fuel
2. Ranges of measured and/or best estimate fission-gas release and end-of-life plenum pressures in high-burnup fuel rods
3. Cladding types and drying-storage conditions that may lead to radial-hydride-induced embrittlement
4. Other degradation mechanisms for fuels and internals
5. Arrangement of fuel assemblies in casks containing high-burnup fuel

However, we were not successful in obtaining new data, especially data from utilities, beyond the data Argonne continues to generate and share in the area of embrittlement of cladding from high-burnup fuel rods.
The January 2013 conference phone call will help jump-start our efforts.

If your schedules permit, I suggest we hold this conference phone call with EPRI during the week of Jan. 7-11, 2013.

Please let me know
if you are still interested in being a member of the Fuels/Internals Subcommittee,
if you would like to participate in the phone call, and

E94

if you will be providing me with a list of issues you would like to discuss with EPRI.

I will prepare my list and send it to you by this Friday.
It would be most productive if you would send me your list before Jan. 4th so I can compile them into one document.

I look forward to hearing from you and talking to you after the holidays,

Mike
p.s. Keith, as I may not have a complete list of Subcommittee members, please forward this email to the few members I may have omitted.

91

Einziger, Robert

From: Billone, Michael C. <billone@anl.gov>
Sent: Thursday, January 10, 2013 3:34 PM
To: Einziger, Robert
Subject: RE: ESCP Fuels-Internals Subcommittee

Bob,

Albert has an agenda.
He presents and interprets data in an attempt to show there are no unresolved or relevant issues regarding high-burnup fuel.

It took me several tries to figure out what he did in Slide 15.
For non-IFBA rods, he gives 3.89 MPa as the average EOL pressure.
Yet, most of the high-burnup data points are higher than this average (about 4.4 MPa)
Clearly his average and 2σ values come from averaging in the lower pressures at 30-35 GWD/TU

Also, recall that no data are shown for IFBA rods, which are clad in ZIRLO.
He lists the average of IFBA rods close to 6 MPa.
Most likely, he averaged in low-burnup data points with high-burnup data points.
For high-burnup IFBA rods, the average is likely to be above 6 MPa.

For our NRC tests, we used 6.5 MPa to get a 400°C hoop stress of 110 MPa
For our recent DOE tests, we used 4.55 MPa and 5.12 MPa to get 400°C hoop stresses of 82 and 92 MPa, respectively.
The difference in hoop stress between 400°C and 350°C is about 9%.
High-burnup ZIRLO at 82 MPa peak stress has a DBTT < 20°C.
We do not yet have the results for ZIRLO at 92-MPa, but I bet that the DBTT is >50°C.

Westinghouse's OFA/IFBA design is very popular and the cladding is ZIRLO.
I still think there is a radial-hydride embrittlement problem for high-burnup ZIRLO.

Also, there is still gas in the fuel column even after high burnup.
I vaguely remember that fresh rods had a 1 to 1 ratio of gas in the fuel column to gas in the plenum.
Dished fuel pellets do not sinter in such a way to eliminate the dish volume.
So, maybe 10 to 20% of the gas is still in the fuel column at temperatures higher than the plenum temperature.

Mike

From: Einziger, Robert [mailto:Robert.Einziger@nrc.gov]
Sent: Thursday, January 10, 2013 12:17 PM
To: Billone, Michael C.
Cc: Einziger, Robert
Subject: RE: ESCP Fuels-Internals Subcommittee

Mike,

1- I will reserve the noon slot on the 22 for an hour
2- Almost all the EPRI documents are opinion pieces not data compendiums.
3- Albert makes a few good points with his slides 15 and 16

Slide 15 indicates that the expected pressures in the rods, when corrected for the maximum storage temperature, and the thickness of the cladding is below the 110 MPa stress where the DBTT have been determined. There is no data to indicate how low the stress need to be before there is no longer a hydride reorientation effect. We do know that the

E95

hydride reorientation will take place at much lower stresses than 110 MPa, but will enough occur to significantly affect the DBTT. While the NRC/EPRI work was done to try to get results that would bound the complete rod population, there is nothing preventing people from measuring the DBTT for lower maximum stresses that bound the part of the population that they want to store and transport. We can easily put a condition in the Certificate mandating the stress be below whatever level the applicant feels comfortable with and can defend.

The second point that Albert makes that is worthwhile is on slide 16. The majority of the gas in high burnup rods, where the gap is limited or non-existent, is in the plenum. The pressure in the rod, hence the stress will be governed by the temperature in the plenum region. When evaluating the effects of hydride reorientation the solubility of the hydrogen should be determined at the location the cladding is being evaluated but the stress should be determined by the plenum temperature. Since there is both a radial and axial temperature distribution in a cask loading, very little of the actual cladding may be susceptible to hydride reorientation effects. I have been long advocating someone do this analysis. Of course this type of analysis requires a reasonably accurate temperature code better than the applicants are currently using.

REE

From: Billone, Michael C. [mailto:billone@anl.gov]
Sent: Thursday, January 10, 2013 12:18 PM
To: Brady Hanson (brady.hanson@pnl.gov); Donghak Kook (syskook@kaeri.re.kr); Zigh, Ghani; hpfeifer@nacintl.com; John Kessler; Keith Waldrop (kwaldrop@epri.com); Kris Cummings (cumminkw@westinghouse.com); Miriam Lloret (mll@enusa.es); Peter.Stefanovic@constellation.com; raheel.haroon@areva.com; Einziger, Robert; Sven Bader (sven.bader@areva.com); Yong Yan (yany@ornl.gov); zimartin@tva.gov
Cc: Albert Machiels; Manuel Quecedo (MQK@enusa.es); Ken Sorenson (kbsoren@sandia.gov); Liu, Yung Y.
Subject: ESCP Fuels-Internals Subcommittee

Dear Colleagues,

Enclosed (EXCEL file) is an updated membership list for the ESCP Fuels-Internals Subcommittee.
Please review it, add missing information, and let me know if you would like to continue to be a participant.

Also enclosed (Word file) is a list of EPRI reports (public and restricted) relevant to our work, as well as a link to enable you to access the publicly available reports.

At Nov. 26-27 ESCP meeting, John Kessler took on the action item of providing the list of EPRI reports (prepared by Albert Machiels) and
of setting up a conference phone call, which would enable Subcommittee members to ask EPRI questions regarding clarification of published data
and regarding "data trends" gleaned from proprietary work. Although the proprietary data itself cannot be shared, trends and 'lessons learned" may be
helpful to us in assessing issues, databases available, and data needs relevant to extended storage and post-storage transport of high-burnup fuel.

I would like to schedule this call for Jan. 22nd at 9 a.m. PST (noon EST).
Please let me know your availability to participate in this call.
We should shoot for 60 minutes for the conference phone call, but we should allow up to 90 minutes

The backup date for the call is Feb. 5th at 9 a.m. PST.

I am preparing a list of questions to ask EPRI (John, Albert, etc.).
I will send it to you shortly.
Please send me your questions to add to my list.

Given the diverse nature of our group, my questions go beyond what EPRI may be able to address.
Some of my questions may be best addressed by Subcommittee members and/or their colleagues.
Please feel free to answer them prior to the call or during the call.

A number of questions I originally planned to ask are answered partially in the enclosed (pdf file) presentation by Albert Machiels.
In particular, please see Slide 15 for measured end-of-life fuel-rod internal pressures vs. burnup.

I look forward to working with you during this new year,

Mike

92

Einziger, Robert

From: Jacobs, Christian
Sent: Tuesday, January 15, 2013 2:28 PM
To: Rubenstone, James; McCartin, Timothy; Michalak, Paul; Horn, Merri; Muir, Jessie; Wentzel, Michael; Rubenstone, James; White, Bernard; Einziger, Robert
Cc: Davis (WCD), Jennifer; Brown, David; Moser, Michelle; Campbell, Tison
Subject: Summary for Meeting (1/15/13) between SFST/SFAS/WCD

All,
For those on the "To" line (i.e., those who attended this morning's meeting), please reply all if I missed something critical (or misstated) below.

- Bob and Bernie indicated that, technically, we can safely store indefinitely using current technology (i.e., even if we have to overpack or repackage including handling bare fuel) given that current regulations require aging management plans be reviewed at least at every license renewal.
- Data for high burn up fuel is less than low burn-up fuel
- High Burn-up fuel may get to degraded fuel quicker (don't know yet).
- SFST working on an ISG document involving a demo project for high burn up fuel. Right now, SFST will only license high burn-up fuel for up to 20 years. After the demo is completed (and if it confirms that the high burn-up remains intact), than SFST will have a data point that may allow them to issue renewals for 20 additional years (regulations allow up to 40 years in a renewal).
- Low burn-up fuel has been licensed for storage up to 40 years in several recent renewals.
- Bob indicated that SFST supported up to 100 years of safe storage for low burn up fuel in the 2010 WC Decision (this did not apply to high burn up fuel). Bob indicated that the analysis for the 60 years post license life in the 2010 decision was only valid for low burn up fuel.
- It may be easier to define "routine maintenance" as an activity which it is not (i.e., when replacement/repackaging activities require the bare handling/transfer of fuel than we have passed the threshold of routine maintenance)

Christian Jacobs
Sr. Project Manager
NRC/NMSS/WCD
(301) 492-3265
Office: EBB-2B26
Mail Stop: EBB-2B02
christian.jacobs@nrc.gov

E96

Einziger, Robert

From: Jacobs, Christian
Sent: Tuesday, January 15, 2013 5:05 PM
To: Stuyvenberg, Andrew; Michalak, Paul; Brown, David; Rubenstone, James; Wentzel, Michael; Muir, Jessie; Jackson, Christopher; Wood, Kent; Jones, Steve; Easton, Earl; McCartin, Timothy; Watson, Bruce; Einziger, Robert; Davis, Jack; McGinty, Tim; Hickman, John; Casto, Greg; Campbell, Tison; Sampson, Michele; Horn, Merri; 'Laplante, Patrick A.'; White, Bernard; Gall, Jennifer
Subject: Summary for NMSS/FSME/NRR meeting (1/15/13) regarding WC EIS (30 yr pool storage post license life)

All,
Below are my notes from this meeting including identified actions:

- SFST (Bob) indicated that as long as water chemistry is maintained in the pool, fuel does not degrade
- **No specific issues identified for high burn up fuel for wet storage (NRR has action to verify this statement). SFST indicated high burn up fuel issue is primarily related to dry storage.**
- NRR working on generic letter regarding degradation of neutron absorbers.
- Primary question (purpose of the meeting) what is the technical basis of the 2010 WC Decision regarding 90 years storage in the pools? **NRR has action to investigate (confirm) basis for timeframe of safe storage in pools as indicated in 2010 WC decision (i.e., can safely store in pools 30 years post license life of the reactors, up to 90 years total wet storage). NRR will provide answer to WC EIS team by mid next week (1/23/13) whether or not the technical basis is still valid.**
- **Tim has action to provide references (regarding pool storage from the 2010 WC Rule and Decision) to NRR by Thursday, 1/17/13.**

If I missed something major, please let me know and reply all so others on this email will be aware.
Thanks,
Chris

Christian Jacobs
Sr. Project Manager
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(301) 492-3265
Office: EBB-2B26
Mail Stop: EBB-2B02
christian.jacobs@nrc.gov

Einziger, Robert

From: Jacobs, Christian
Sent: Thursday, January 17, 2013 8:48 AM
To: Stuyvenberg, Andrew; Einziger, Robert; 'Laplante, Patrick A.'; Brown, David; Campbell, Tison; Davis (WCD), Jennifer; Houseman, Esther; London, Lisa; McCartin, Timothy; Michalak, Paul; Moser, Michelle; Muir, Jessie; Pineda, Christine; Wentzel, Michael
Subject: Follow-up to yesterday's action item concerning how quickly high burn-up fuel fills the pool

All,
Yesterday's action: **Determine how quickly the pool will fill up with high burn up fuel.**

Response: I spoke to Bob this morning and went over some of the assumptions that were discussed with SFST during the 1/15 meeting. Although, it really depends on the pool size, below are some reasonable assumptions.

- It takes 15-20 years for high burn-up fuel to cool in the pool.
- Each core load lasts 5-6 years.
- The minimum pool size needs to accommodate 4 cores + 1 reserve
- Therefore, a reasonable conservative assumption is that the minimum sized pool will fill up in 25-30 years (i.e., 5 cores x 5 to 6 years = 25 to 30 years)

Chris

Christian Jacobs
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(301) 492-3265
Office: EBB-2B26
Mail Stop: EBB-2B02
christian.jacobs@nrc.gov

Einziger, Robert

From: Einziger, Robert
Sent: Tuesday, February 26, 2013 7:32 AM
To: Muir, Jessie; Klementowicz, Stephen; Santiago, Patricia; Folk, Kevin; Fetter, Allen; Compton, Keith; Park, James; Schaaf, Robert; Markley, Christopher; Donnell, Tremaine; Olson, Bruce; Hill, Brittain; Stablein, King; Ruffin, Steve; Easton, Earl; Correia, Richard; Diaz-Toro, Diana; Gavrilas, Mirela; Davis (SFST), Jennifer; Rubenstone, James; Banovac, Kristina
Cc: Michalak, Paul; Einziger, Robert
Subject: RE: ACTION: WC EIS Draft Files to Review
Attachments: Comments on WC.doc

Jessie,

Attached are my comments on the document.

From: Muir, Jessie
Sent: Tuesday, February 19, 2013 12:47 PM
To: Klementowicz, Stephen; Santiago, Patricia; Folk, Kevin; Fetter, Allen; Compton, Keith; Park, James; Schaaf, Robert; Markley, Christopher; Donnell, Tremaine; Olson, Bruce; Hill, Brittain; Stablein, King; Ruffin, Steve; Easton, Earl; Correia, Richard; Diaz-Toro, Diana; Gavrilas, Mirela; Davis (SFST), Jennifer; Einziger, Robert; Rubenstone, James; Banovac, Kristina
Cc: Michalak, Paul
Subject: ACTION: WC EIS Draft Files to Review

All,

Attached for your review are Chapters 1-5 and the Spent Fuel Fires Appendix of the Waste Confidence (WC) EIS. Your review and comments are requested by **COB Tuesday, Feb 26th**. You can provide your comments as a separate document, as "sticky notes" within the PDF, or bulletized in an email.

Please remember these files are preliminary. They are still being worked, they have not been tech-edited, formatted, nor have they been NLO'd by OGC. We are looking for your comments on issues and language that your office can not agree with – either on a regulatory, technical, or NEPA basis. If you have any information/suggestions that would strengthen an argument, please provide.

DO NOT DISTRIBUTE. We would like to keep the review contained to those staff identified by their Office management. If you feel you need to forward to someone else, please let me know before you do so.

You will be receiving a scheduler for a meeting on Wed, Feb 27th. The purpose of that meeting will be to discuss the comments received so all the offices can be aware of issues raised by others. If you have any questions, please let me know.

Thanks,
Jessie

Jessie M. Muir

Jessie M. Muir

EIS Project Manager
Waste Confidence Directorate
U.S. Nuclear Regulatory Commission
Phone: 301.492.3116
Office: EBB2-A24

1- P1-13, Line 6- Are you going to leave Morris out of this analysis?
2- P 1-13, Line 16 – "container' needs to be defined. Dry storage uses the terms cask, canisters, overpacks, DSS, pad but not containers
3- P 1-21, Line 21 – Define DTS the first time it is used
4- P 1-14 fig 1-x – In terms of real years is the figure referring to "licensed life" as yr 40-120, "ST storage" as 100-180 yrs, and "LT storage" as 200-280 yrs, or is there some other interpretation of this figure. The time frames are not clear. In the first bullet of licensed life are you referring to the original reactor or ISFSI license?
5- P 1-14, Line 9 – What about Morris?
6- P 1-15 Line 11 – I disagree with bullet 1. The most likely scenario based on the BRC recommendations is that if there is no repository then the situation to be analyzed is is transportation to a regional or central interim site not to leave at the reactor.
7- P 1-16, Line 14 – Are you implying that you are not considering the case where there is nuclear growth, only replacement of current capability. That makes this analysis sort of sterile as the growth case is realistic
8- P 1, Line 9 – There is other commercial fuel stored at INL besides that which is mentioned, for example the Peach Bottom core, and FSV fuel.
9- P 5, Line 32 – The fuel pellet stack is usually 12 ft long the rods are in general longer in the range of 14 ft. 14 x 14 bundles are still in use
10- P 6, Line 20 – add "oxide" after thorium
11- P 6, Line 26 – Delete this paragraph, it is extraneous to this document. There will be a new waste confidence ruling before this ever becomes a consideration
12- P 7- remove sections on LMFBR and SMR. These aren't relative to this action and can be replaced by a single statement that there will be no operational reactors of these types in commercial operation in the near future.
13- P 12, Line 15 – remove the remainder of the sentence after "demonstrate, and add "demonstrate all the conditions in the license for the approved system used are met". This is more than tornadoes and earthquakes.
14- P 13, Line 22 – This item includes more things, for example: the ability to do required acceptance tests for transportation after storage.
15- P 14 Line 8 – 1200 MTU is for what reactor lifetime operating period. How much will it change the analysis if licenses are extended?
16- P 15, Line 4 - "and are in operation". You need to be careful; this may be true for DTS for canisters but not for bare fuel assemblies.
17- P 15 Line 22 – Why is there a discrepancy between the capacity to load a receiving cask, and I presume you mean "unloading" a source cask?
18- P 15 - This whole idea of moving the equipment from inside a DTS from one facility to another is a pipe dream. Based on my experience running hot cells and trying to get them decontaminated up for release of equipment, 1) decommissioning for transportation would be a nightmare especially if there was any failed fuel that released particulate in the facility, 2) the DTS would have to be an inerted facility so you don't run the risk of oxidizing failed fuel and really crapping up the place, 3) the price tag is probably a factor of 10 too low. 4) The concrete structure would be a decommissioning project that needs consideration in developing a waste estimate, and 5) such a facility won't be available for a minimum of 15-20 years
19- P 26, Sec 2.2.5 and 2.2.6 – Both of these are major efforts and are not as simple as one is led to believe by these brief write-ups.

20- P 27, Line 2 – add "repackaging" after "and"
21- P 3-35, Line 13 – "most common mode" "The most common mode for fuel transport is train not truck
22- P 4-1, Line 7 – Moving the oldest fuel first is probably a poor assumption. In most likelihood the utilities will want to move the hottest fuel possible out of the pool to reduce the heat load and lesson the need to buy more DSSs. What affect on the conclusions will the reverse assumption have?
23- P4-1 As I read the first 17 sections I kept wondering why they consequences were always so low. It wasn't until I got to Sec 4.18 that I realized that the first 17 section sonly dealt with storage under normal conditions and that accident and other events were being considered separately. There should be a statement in the first paragraph of Sec 4.0 explaining the organization of the analysis
24-
25- P 4-1, Line 17 – add "pool" after "fuel"
26- P 4-1, Line 23, 26 – Bullet implies with action will occur repeatedly. For the assumption of LTS of only100 yrs (See Fig 1-x), this will only occur once in LTS.
27- P 4-2, Line 20, GENERAL COMMENT – Refer back to the Section where these terms (SMALL, MODERATE etc) are defined when these terms are used.
28- P 4-11, Line 30. Wherever possible the use of terms like "limited" should be supported with calculation. In this case it is easy to show that "limited" is about 10%.
29- P 4-62, Line 17 – This definition of "accident" does not agree with 10 CFR 71, 72. With this definition if there is no consequence(release) then there is no event
30- P -70, top of page – I find the argument for no consequence very lame. It says that we have no issue because we have a regulation that says we have no issue. The argument should be "We review each application on a case by case basis to ensure there are sufficient supportable design arguments and testing that the regulatory requirements are met thus there are no excessive dose expectations."
31- P 4-70 Line 13 – At least reference relevant tests and analysis supporting this conclusion
32- P 4-70, Line 31 – This was the only drop accident analyzed in the PRA
33- P 4-71, Line 7 – This is a section on accident consequences. This conclusion ca only be supported, and these functions continue to be met, only if the DTS isn't breached during an accident. Basically you are saying if there isn't an accident then there isn't a problem. Aren't you suppose to be looking at the consequences if there is an accident?
34- P 4-71, Line 14 – But there are handling experiences where Lifts get stuck. I know of at least two where there was significant hold up because of a stuck lift within the last 8 years.
35- P 4-71, Paragraph starting line 15. You need to consider an accident that breaches the facility and allows air into the atmosphere when fuel oxidation will occur. I do not find the argument that the environmental risk is small to be persuasive.
36- P 4-71 Sec 4.18.3 – Is this the impact of accidents on climate change or of climate change on accident consequences?
37- P 4-79, Sec 4.19.4 – INSR had Sandia do a study on this very subject supporting the conclusion reached in this section. You should reference the unclassified version of this study.
38- P 5-1 – 1) any comments of a technical nature that I made for Sec 4 would also apply to Sec 5. 2) This section should be cut way back with reference made to the analysis in Section 4 where applicable.

39- P 5-2, Line 13 – While PFS is used as a guide. Everything in this section must be generalized to an arbitrary site so references to PFS should be removed in most places.

40- 5-3, Line 27 "PFSF" – Before (P 5-2, Line 11) PFSF referred to the Utah site. Now it seems to be generalized to mean any away from reactor site. A difference term should be used. Also in all likelihood any such facility will not be privatized but will be run by a government agency, so the term

41- P 5-34, Section 5.16 – This section does not seem to cover the analysis of the actual transportation of the SNF to the away from reactor interim site, specially the effects of restriction of rail, road use due to the activity, and resources for security, rail up grading, etc. There should be ample information in the YM EIS to do this activity. In addition to the one transportation step if there is no interim site, there are now two transportation events 1) to the interim site, and 2) to the repository. This ia a major difference between the AR and AFR scenario.

42- P 5-40, Line 31 – see comment 41

43- App F Sec F.1 1st paragraph, Line 7 – A combustion reaction is an oxidation process

44- . App F Sec F.1 1st paragraph, Line 11 – The rods are going to burst and release fission products and fuel particulate long before a sustainable fire temperature is released

45- Paragraph starting "The behavior of ruthenium", Line 5 – add "Ruthenium oxide which results from the oxidation of Ru in the fuel rods has a very high vapor pressure even at room temperature and will be preferentially released from a breached fuel rod.

Einziger, Robert

From: Jacobs, Christian
Sent: Friday, August 17, 2012 7:57 AM
To: Davis (FSME), Jennifer; Stuyvenberg, Andrew; Dunn, Darrell; Rubenstone, James; Einziger, Robert; Davis, Jack; Wood, Kent; Jackson, Christopher; Ahn, Tae; McCartin, Timothy; Compton, Keith; Miriam Juckett
Cc: Stablein, King
Subject: Extended Wet Storage Meeting follow-up
Attachments: IAEA TECDOC-1012_prn Durability of SNF and Components in pools.pdf

FYI- For those who attended the meeting on Wednesday, Bob thought the attached document may be of interest.

Also FYI:

1) yesterday during an SFST public meeting, Jim Rubenstone asked Rod McCullum if the industry is intending or trending towards the use of wet storage as a long term storage option. Rod answered that industry has no intention of doing that, and that dry storage is the preferred alternative for extended periods of time.
2) There were also questions asked yesterday whether a pool is necessary (to be available) after the decommissioning of a plant, particularly after everything has been moved to dry storage. A former NRC employee (Charlie ??) mentioned that they developed a license for Rancho Seco in the mid 90s that allowed the use of some sort of dedicated dry storage transfer cask (that essentially eliminated the need for a pool after the plant was decommissioned). I believe Jennifer Davis was going to follow up on this.
3) Marc Nichol (NEI) also mentioned that the industry has other methods of handling the need to reopen a cask in a dry environment – should the fuel have to be unloaded and transferred. Although I believe the method he mentioned hasn't been proven or accepted at the NRC.

From: Einziger, Robert
Sent: Thursday, August 16, 2012 9:02 AM
To: Jacobs, Christian
Cc: Rubenstone, James
Subject:

Chris,

The attached document and the reference below might be of interest to the attendees at yesterday's meeting

Survey of Experience With Dry Storage of Spent Nuclear Fuel and Update of Wet Storage Experience

Technical Reports Series 290 Subject Classification: 0803-Spent fuel management

STI/DOC/010/290 (ISBN:92-0-155388-9) 43.50 Euro;

Language: English

Date Published: 1988

Einziger, Robert

From:	Lee, Richard
Sent:	Tuesday, January 22, 2013 9:37 AM
To:	Einziger, Robert
Cc:	Algama, Don; Voglewede, John; Dunn, Darrell; Rubenstone, James; Gavrilas, Mirela
Attachments:	Stress Sources.doc

Hi, Bob:

Per discussion with John Voglewede and Don Algama (staff assigns to work on the NMSS extended storage), enclosed is our thoughts on the sources of post-irradiation cladding stress for long term storage of spend fuel casks. I propose that you take a look at this, and NMSS and RES meet to discuss your thought on this.

Best regards,
Richard

P.S. I think you know that Katie Wagner left RES and move to FSME.

E97

Sources of Post-Irradiation Cladding Stresses

By memorandum dated July 12, 2012 (ML12163A536), the Office of Nuclear Materials Safety and Safeguards (NMSS) amended User Need Request NMSS 2011-002. That amendment requests additional work by the Office of Nuclear Regulatory Research (RES) to address knowledge gaps for emerging technical issues that have been identified for extended storage and transportation of spent nuclear fuel.

As part of Task 7, the amendment states:

> *NMSS is requesting additional detailed technical project planning be performed under Task 7. As described in the draft Technical Information Needs report, many fuel and cladding degradation processes depend upon a source of stress within the cladding from mechanisms such as pellet swelling or fission gas release. These processes may not initiate or propagate if the stress state of the cladding is sufficiently low. A detailed project plan to assess sources of cladding stress is necessary to address this issue.*

In a response dated September 10, 2012 (ML12235A305), RES stated:

> *RES has sponsored extensive characterization and testing of fuel rods after reactor operation. However, the work has been limited to short times (a few years) following discharge from the reactor. RES understands that the goal of the proposed work is to study long-term changes in the fuel and cladding that may take place over many decades following irradiation. In order to better identify the information requirements of this technical issue, RES and NMSS staff will further discuss the information needs for EST before developing a detailed technical project plan.*

Rather than first determining the post-irradiation stress state of the cladding, RES proposes to initially examine the source of these stresses (e.g., "mechanisms such as pellet swelling or fission gas release" according to NMSS"). The reason for this approach is that pellet-to-cladding mechanical interaction is generally greatest during in-reactor operation when fuel temperatures are high and thermal expansion is maximum. Consideration of these stresses is made during the operational (rather than storage) review of a fuel design. Following irradiation, the fuel temperatures drop and alleviate the major source of pellet-to-cladding mechanical interaction. For this reason, the other sources of cladding stress must be considered for extended storage.

RES will determine the extent of post-irradiation fission gas production – a possible source of pellet swelling or fission gas release – using existing methods (e.g., SCALE) but applied over post-irradiation time periods now being considered. For extended storage, the major gas constituent to be considered is decay product helium.

The work will be conducted in-house by RES staff.

From: Einziger, Robert
Sent: Wednesday, January 23, 2013 11:10 AM
To: Lee, Richard
Cc: Dunn, Darrell; Gavrilas, Mirela; Rubenstone, James; Voglewede, John; Einziger, Robert
Subject: RE: Investigation of sources of cladding stresses.

Richard,

Your response left me with a number of questions that I have since resolved with John Vogelwede. I agree with your staff that the differential thermal expansion results in a stress on the cladding that is relieved when the fuel is cooled. It is not the thermal differential that we are trying to assess as a source of pellet stress on the cladding. Rather we are trying to determine the effect of the helium that is generated over time by the decay of the fission products. If the helium remains in the pellet swelling occurs. This swelling can be accommodated either by filling the gaps resulting from the mismatch of fragments when a pellet thermally breaks early in life or by placing a stress on the cladding. Alternatively a portion of the gas can be released to the rod void space increasing the rod pressure and cladding stress. Previous calculations of the gas pressure needed to drive DHC , was that the internal pressure of the rod would have o double or triple at least. Recent work by Rondinello , using ion implantation, was that these effects should start around 100 years after reactor operation. This was the bases for a paper by Ahn and Rondinello predicting cladding stress.

While DOE is working on attacking the cladding degradation mechanisms that might occur at low temperature such as low temperature athermal creep and DHC, I am unwilling to spend resources on these mechanisms until we have some sort of experimental confirmation that the driving stresses actually exist. John indicated that RES would like to do independent in-house verification of the Rondinello results before trying to spend time to brainstorm an experimental verification technique or let an RFP of other organizations to propose experimental methods to determine if a stress will exist.

I agreed with John that an independent evaluation would be useful, and he agreed to send me a schedule when this can be completed taking into account other commitments on your staff that may have higher priority. I would suspect that the RES modeling studies will be no more definitive than the Rondinello implant studies to I am also requesting that your organization, or another organization within RES if you do not have the manpower, develop an RFP for soliciting ideas for obtaining experimental proof or denial of such an operative stress. We would like to be in a position at the start of next fiscal year to let the RFP should you modeling not be definitive.

I look forward to the schedule for your modeling effort and development of the RFP as soon as possible.

Yours

RE Einziger

From: Lee, Richard
Sent: Tuesday, January 22, 2013 9:37 AM
To: Einziger, Robert
Cc: Algama, Don; Voglewede, John; Dunn, Darrell; Rubenstone, James; Gavrilas, Mirela
Subject:

Hi, Bob:

E98

Per discussion with John Voglewede and Don Algama (staff assigns to work on the NMSS extended storage), enclosed is our thoughts on the sources of post-irradiation cladding stress for long term storage of spend fuel casks. I propose that you take a look at this, and NMSS and RES meet to discuss your thought on this.

Best regards,
Richard

P.S. I think you know that Katie Wagner left RES and move to FSME.

95

Rubenstone, James

From: Doolittle, Elizabeth
Sent: Tuesday, January 29, 2013 10:54 AM
To: Rubenstone, James
Subject: FW: Clad Stress: Previous Email
Attachments: Stress Sources.doc

Hi Jim, here's the list Richard referred to.

I'll send you his comments about what they are doing.

Beth

From: Algama, Don
Sent: Monday, January 28, 2013 1:20 PM
To: Doolittle, Elizabeth
Cc: Lee, Richard
Subject: Clad Stress: Previous Email

Beth:

The below email, and attachment, may be helpful to you.

-Don A.

From: Lee, Richard
Sent: Tuesday, January 22, 2013 9:37 AM
To: Einziger, Robert
Cc: Algama, Don; Voglewede, John; Dunn, Darrell; Rubenstone, James; Gavrilas, Mirela
Subject:

Hi, Bob:

Per discussion with John Voglewede and Don Algama (staff assigns to work on the NMSS extended storage), enclosed is our thoughts on the sources of post-irradiation cladding stress for long term storage of spend fuel casks. I propose that you take a look at this, and NMSS and RES meet to discuss your thought on this.

Best regards,
Richard

P.S. I think you know that Katie Wagner left RES and move to FSME.

E99

Sources of Post-Irradiation Cladding Stresses

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As part of Task 7, the amendment states:

> *NMSS is requesting additional detailed technical project planning be performed under Task 7. As described in the draft Technical Information Needs report, many fuel and cladding degradation processes depend upon a source of stress within the cladding from mechanisms such as pellet swelling or fission gas release. These processes may not initiate or propagate if the stress state of the cladding is sufficiently low. A detailed project plan to assess sources of cladding stress is necessary to address this issue.*

In a response dated September 10, 2012 (ML12235A305), RES stated:

> *RES has sponsored extensive characterization and testing of fuel rods after reactor operation. However, the work has been limited to short times (a few years) following discharge from the reactor. RES understands that the goal of the proposed work is to study long-term changes in the fuel and cladding that may take place over many decades following irradiation. In order to better identify the information requirements of this technical issue, RES and NMSS staff will further discuss the information needs for EST before developing a detailed technical project plan.*

Rather than first determining the post-irradiation stress state of the cladding, RES proposes to initially examine the source of these stresses (e.g., "mechanisms such as pellet swelling or fission gas release" according to NMSS"). The reason for this approach is that pellet-to-cladding mechanical interaction is generally greatest during in-reactor operation when fuel temperatures are high and thermal expansion is maximum. Consideration of these stresses is made during the operational (rather than storage) review of a fuel design. Following irradiation, the fuel temperatures drop and alleviate the major source of pellet-to-cladding mechanical interaction. For this reason, the other sources of cladding stress must be considered for extended storage.

RES will determine the extent of post-irradiation fission gas production – a possible source of pellet swelling or fission gas release – using existing methods (e.g., SCALE) but applied over post-irradiation time periods now being considered. For extended storage, the major gas constituent to be considered is decay product helium.

The work will be conducted in-house by RES staff.

Rubenstone, James

From: Dunn, Darrell
Sent: Wednesday, February 13, 2013 5:28 PM
To: Rubenstone, James; Einziger, Robert
Cc: Doolittle, Elizabeth
Subject: EST Cladding Stress
Attachments: Clarified cladding stress work scope PR.docx

Please find the attached file on the RES cladding stress work scope. Patrick Raynaud will have the lead on this effort. His contact information is included below.

Patrick A.C. Raynaud, PhD
Reactor Systems Engineer (Fuels)
U.S. Nuclear Regulatory Commission
RES/DSA/FSCB
Mailstop: CSB-3A07M
Washington, DC 20555
Tel: (+1) 301-251-7542
patrick.raynaud@nrc.gov

Call, email or copy me on email if you have questions. Thanks,

Darrell Dunn
RES/DE/CMB
Phone: 301-251-7621
Fax: 301-251-7420

Subtask 4 of Task 7 under User Need NMSS 2011-002

Gap analyses have identified low temperature creep and delayed hydride cracking (DHC) as potential cladding breach mechanisms after ~100 yr of dry storage. Both of these mechanisms require the presence of a stress in order to be active.

Proposed sources of stress are plenum gas pressure, phase change of the hydrides upon cooling from drying temperatures, and swelling of the fuel due to a buildup of helium decay product (resulting in pellet-cladding mechanical interaction - PCMI). Two public reports were issued in 2012 describing these topics among others: "Review of Used Nuclear Fuel Storage and Transportation Technical Gap Analyses", from DOE, discusses fuel pellet restructuring and swelling in section 3.2.5 and cladding creep at low temperature in section 3.3.4; "Identification and Prioritization of the Technical Information Needs Affecting Potential Regulation of Extended Storage and Transportation of Spent Nuclear Fuel", from NRC, discusses fuel pellet swelling in section A2.3 and cladding creep at low temperature in section A1.4.

Although the rod pressure at the end of irradiation may be too low to drive the cladding breach mechanisms, helium production due to alpha decay and PCMI induced by swelling of the pellets via a buildup of helium have been proposed as sources for cladding stress. Until stress levels in the cladding are evaluated over the extended storage period, it is unclear whether low temperature creep, DHC (induced by rod pressure, PCMI, or both), or simply PCMI cause a regulatory concern. This task will determine if significant levels of cladding stress exists in dry cask storage and the magnitude of the stress if present.

Specifically the task will (staff-time estimates provided):

1- Determine the production of gaseous elements as a function of time after discharge for light water reactor fuel. By 03/22/13
2- Determine how much the overall pressure in the rod will increase if this gas is all released to the plenum. By 04/05/13
3- Determine how much of the gas will release from the fuel pellet to the plenum taking into account the fuel temperature, location of the generation of the gas, and structure of a HBU pellet. By 05/31/13
4- If the gas remains in the rod and the expected amount of gas remains in the pellets, determine the extent of pellet swelling as a function of time. By 07/19/13
5- Compare and contrast these results with studies conducted by other research institutions. By 08/23/13
6- Consider other potential sources of stress. By 09/27/13
7- Should estimates of internal rod pressure or fuel swelling show negligible increase over fuel discharge conditions, work on this activity will cease.
8- If estimates of rod internal pressure change are <u>not</u> negligible, determine the increase in cladding stress due to rod internal pressure and calculate the expected creep strains as a function of time. By 10/25/13
9- If estimates of bulk fuel swelling after discharge are <u>not</u> negligible, determine how much of the fuel swelling will be accommodated by inter-fragment space and how much is available to apply stress to the cladding. By 11/29/13

Completion of the work estimated by 11/29/13

The results of this work will be periodically discussed and a final report produced in a format consistent with other responses to this User Need Request.

View Letter

Close

Date: Feb 11, 2013
To: "Tae M Ahn" tae.ahn@nrc.gov
From: Lars Werme lars.werme@fysik.uu.se
Subject: Your submission to Journal of Nuclear Materials

Ms. Ref. No.: JNM-D-12-00891
Title: Source term analysis in handling canister-based spent nuclear fuel
Journal of Nuclear Materials

Dear Dr. Tae M Ahn,

We have now received the reviewer's comments on the manuscript, Source term analysis in handling canister-based spent nuclear fuel. The reviewer recommends that the manuscript be reconsidered after major rework and resubmission.

The Editors and Publisher encourage you to revise your manuscript without delay beyond the time necessary to complete the work required. Manuscripts not resubmitted within 180 days will be withdrawn automatically.

In your cover letter for the revised version, describe how you have incorporated the comments.

To submit your revision, please do the following:

1. Go to: http://ees.elsevier.com/jnm/

2. Enter your login details

3. Click [Author Login]
This takes you to the Author Main Menu.

4. Click [Submissions Needing Revision]

PLEASE NOTE: Journal of Nuclear Materials would like to enrich online articles by displaying interactive figures that help the reader to visualize and explore your research results. For this purpose, we would like to invite you to upload figures in the MATLAB .FIG file format as supplementary material to our online submission system. Elsevier will generate interactive figures from these files and include them with the online article on SciVerse ScienceDirect. If you wish, you can submit .FIG files along with your revised submission.

I look forward to receiving your revised manuscript.

Yours sincerely,

Lars Werme
Editor
Journal of Nuclear Materials

Reviewers' comments:

Reviewer #1: The paper is about results of a scoping study on radiological impacts of drop/collision of spent nuclear fuel assemblies on site workers. Two models/codes (MELCOR and RSAC) were used, which are both well established and accepted. However, several numbers for release fractions of the radionuclide inventories are shown without details on how these codes were actually used. The reviewer is confident that there is very little traceability in this draft, so that there is no way to know whether these values are reliable or not. The

E100

reviewer did limited search in the internet, and found that a paper with a similar title, "SOURCE TERM ANALYSIS IN HANDLING CANISTER-BASED SPENT NUCLEAR FUEL: PRELIMINARY DOSE ESTIMATE" was published with nearly the same authorship (see: http://pbadupws.nrc.gov/docs/ML1126/ML112640440.pdf). While this published paper has subtitle "Preliminary", it includes more figures and text, resulting in more readable and comprehensive. The reviewer must conclude that the submitted paper is a subset of the already published paper.

Because the subject itself is quite interesting and useful for many readers of the journal, the reviewer strongly recommends that the paper is rewritten by substantially including details of the assessment discussed in the paper. Some of potential improvement includes: (1) a figure that shows the model diagram for assessment, which can be a good material to be used as graphical abstract for the prospective paper, (2) showing example calculation for a limited number of cases, for example for one of the values in Table 3 in "Calculated release fraction" column, (3) discussions on limitations and influential assumptions in this assessment and their impacts on the results, etc.

Some minor comments:
(1) Figure 1: why/how are these points scattered? Give physical/mechanistic discussions on the figure.
(2) Equation in Section 9, including Reynolds number, for the intercept removal efficiency: Show derivation or at least direct reference for this formula; Discuss limitation of applicability of this formula.

Reviewer #2: This paper present an interesting approach to source term analysis in case of accidents handling fuel assemblies with or without a canister. However, in the present format it is not suitable for publication. The description of the study is confusing and very schematic. The results are presented in a very qualitative way. There is no clear description of the procedure used to obtain these results. In fact, it is also not clear which set of data constitutes the original contribution of this work. The understanding is not helped by the language expression, which is, at times, hard to decipher. As it stands now, the manuscript looks like a first draft, which needs to be filled with adequate material. On page 1, Introduction, some text is missing at the beginning of the third paragraph.
It is recommended to thoroughly rework the manuscript, adding the missing information: a clear description of the cases considered, possibly helped by illustration, and a clear description of the original work performed in this work. The description should include the method applied to reach the conclusions stated. The results should be expressed in quantitative terms and the qualitative statements could be left for the conclusions. In the following, some more specific comments.
- the scenarios considered must be better defined; in particular, the assumed locations and modes for radionuclide release and for the dose to persons exposed (workers, public) should be defined clearly, possibly with the help of illustrations: the authors talk about "buildings", but don't give much information on their specific layout
- the assumptions made for the calculations should be stated clearly for each case considered; each case should be appropriately labeled and defined; in the current version it is not clear what set of assumptions/conditions applies to what the authors say
- section 2 needs better description: what is meant with "The outdoor worker dose is estimated by the maximum onsite dose"? What is the "uniform Gaussian plume model"? Why and how the "indoor worker dose is estimated based on the outdoor worker dose, considering event sequences associated with a hypothetical case of no building protection for the indoor worker"? Is the accident occurring inside or outside the building? If there is no protection, what's the relevance of assuming the presence of a building?
- section 4. The bulleted list needs to be better explained: is the release fraction normalized to the inventory of 1 fuel assembly? What is meant with "UO2 matrix"? The fraction increase in case of fuel oxidation must be better described; as it is, it looks more like an attempt to account for higher specific surface area due to the high burnup structure; moreover, the sizes should not be expressed in cm;
- is table 1 entirely from ref [1], or there is data generated in this work? In any case the procedure adopted to determine the values in table 1 should be mentioned; also, the extent of oxidation assumed should be defined.
- material at risk: it is stated that 2 fuel assemblies are considered; Table 1 refers to 1 assembly; again, are the fractions normalized to 1 or 2 assemblies (or to another quantity)? Is a damage ratio of 1 really used for the calculations? How is the building discharge fraction defined?
- leak path factor: it should be better defined and explained; how did the "Calculations of the doses...varied the leak path factor values from 1e-7 to 1e-3"?
- Fig. 1 How were data and curve generated? Is the material showed only from literature or there is also a "this work" contribution?
- section 5; how representative is the data on Table 2? More generally, in the case of the pool we have a

continuous leaching process of the fuel in the breached fuel rods, whereas the fuel assembly drop is a singular event (accident) with a certain (very low) probability: the authors thus compare a "reality" with a hypothetical case; this should be mentioned in the text
- section 5.2: again, the description of what is considered is very confusing; the dose at 1 m and at 10 m or at 100 m are not properly introduced; the "smaller confined volume" and the "ventilation fan" are unclear components of a very undefined configuration; the sentence "the likelihood of indoor worker standing in that direction will be inversely proportional to the smaller confined volume" should be expressed in sounder terms
- section 6: where are discussed the MELCOR and RSAC code results?
- section 8: what is the relevance of "The doses are small and increase linearly with the leak rate" without adequate support from data and scenario description?
- Table 3: what is the "filtering efficiency" referring to? How were the values on the table derived, and for which values of R?

Close

97

Einziger, Robert

From: Dunn, Darrell
Sent: Wednesday, February 13, 2013 5:28 PM
To: Rubenstone, James; Einziger, Robert
Cc: Doolittle, Elizabeth
Subject: EST Cladding Stress
Attachments: Clarified cladding stress work scope PR.docx

Please find the attached file on the RES cladding stress work scope. Patrick Raynaud will have the lead on this effort. His contact information is included below.

Patrick A.C. Raynaud, PhD
Reactor Systems Engineer (Fuels)
U.S. Nuclear Regulatory Commission
RES/DSA/FSCB
Mailstop: CSB-3A07M
Washington, DC 20555
Tel: (+1) 301-251-7542
patrick.raynaud@nrc.gov

Call, email or copy me on email if you have questions. Thanks,

Darrell Dunn
RES/DE/CMB
Phone: 301-251-7621
Fax: 301-251-7420

E101

Subtask 4 of Task 7 under User Need NMSS 2011-002

Gap analyses have identified low temperature creep and delayed hydride cracking (DHC) as potential cladding breach mechanisms after ~100 yr of dry storage. Both of these mechanisms require the presence of a stress in order to be active.

Proposed sources of stress are plenum gas pressure, phase change of the hydrides upon cooling from drying temperatures, and swelling of the fuel due to a buildup of helium decay product (resulting in pellet-cladding mechanical interaction - PCMI). Two public reports were issued in 2012 describing these topics among others: "Review of Used Nuclear Fuel Storage and Transportation Technical Gap Analyses", from DOE, discusses fuel pellet restructuring and swelling in section 3.2.5 and cladding creep at low temperature in section 3.3.4; "Identification and Prioritization of the Technical Information Needs Affecting Potential Regulation of Extended Storage and Transportation of Spent Nuclear Fuel", from NRC, discusses fuel pellet swelling in section A2.3 and cladding creep at low temperature in section A1.4.

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Completion of the work estimated by 11/29/13

The results of this work will be periodically discussed and a final report produced in a format consistent with other responses to this User Need Request.

98

Einziger, Robert

From: CONDE LOPEZ JOSE MANUEL <jmcl@csn.es>
Sent: Thursday, February 14, 2013 9:15 AM
To: Billone, Michael C.; Einziger, Robert
Cc: Manuel Quecedo; ALEJANO MONGE CONSUELO; LLORET LLORCA, Miriam; REY GAYO JOSE MARIA
Subject: RE: ENUSA Creep Samples

Dear Mike:

We are looking for the PWR EOL fuel rod pressure data you requested. For the time being, there are two values available. They belong to the Zirlo rods sent to the CABRI International Project, around 68 MwD/MtU average burnup. The reference is the following:

"Non-destructive Examination of CIP 0-1 and CIP 1-1 father rods". Note CABRI Water Loop 2002/28, report Studsvik/N(H)-01/46.

Both NRC and EPRI have this report. The NRC representative in CABRI was initially Ralph Meyer, and is now John Voglewede. He can pass the report to you so that you have the complete PIE information. And yes, the pressures measured (given at 0C in the report) are higher than those reported in your figure.

Please contact me in case of any doubt or problem. We are looking for more info, and I expect to be back to you in a few days.

Best regards

José M. Conde
Jefe de la Unidad de I+D
Consejo de Seguridad Nuclear
Tel.: +34913460253
Fax: +34913460588

-----Mensaje original-----
De: Billone, Michael C. [mailto:billone@anl.gov] Enviado el: miércoles, 06 de febrero de 2013 19:00
Para: CONDE LOPEZ JOSE MANUEL; Einziger, Robert
CC: Manuel Quecedo; ALEJANO MONGE CONSUELO; LLORET LLORCA, Miriam; REY GAYO JOSE MARIA
Asunto: RE: ENUSA Creep Samples

Jose,

The enclosed description of our met mount sample preparation has no propriety restrictions. We benefit most by sharing information and Argonne is committed to helping our international colleagues generate the best datasets they can. We all benefit from such an exchange.

Now, while I have your attention, I am struggling with a graph EPRI has presented for end-of-life (EOL) PWR fuel rod pressure (at 25C) vs. burnup. I know of at least a dozen valuable data points at 68-60 GWd/MTU that are missing from this plot. All have higher end-of-life pressures than the data points show in the plot. However, the data I am referring to may be AREVA- and Westinghouse-propriety data. Albert Machiels (EPRI) will update this figure within the next few months for the ESCP Fuels-Internals Subcommittee (Miriam and Bob are members).

Does Spain have any publically available data on PWR EOL fuel rod pressures?

E102

We are primarily interested in data for rods with >45 GWd/MTU burnup.
If so, I would like to work with the ESCP Fuels-Internals Subcommittee to do our own updating of the enclosed graph.

Thanks,

Mike

-----Original Message-----
From: CONDE LOPEZ JOSE MANUEL [mailto:jmcl@csn.es]
Sent: Wednesday, February 06, 2013 11:42 AM
To: Billone, Michael C.; Einziger, Robert
Cc: Manuel Quecedo; ALEJANO MONGE CONSUELO; LLORET LLORCA, Miriam; REY GAYO JOSE MARIA
Subject: RE: ENUSA Creep Samples

Dear Bob and Mike,

Thank you both for your prompt response and for the interest shown in this matter. The information given by Mike is already interesting, and we look forward to receiving the procedure. In order to do things properly, you would need to tell me what are the limitations in the use of that information, which I assume might have some parts proprietary to ANL and NRC.

I would be willing to sit down with Bob during the RIC, we can also talk about the forthcoming IAEA's CRP meeting. And it would be nice also to meet Mike if he is there, it has been already some time since we last met.

I thank you again for your collaboration. Best regards

José M. Conde
Jefe de la Unidad de I+D
Consejo de Seguridad Nuclear
Tel.: +34913460253
Fax: +34913460588

-----Mensaje original-----
De: Billone, Michael C. [mailto:billone@anl.gov] Enviado el: miércoles, 06 de febrero de 2013 17:40
Para: Einziger, Robert; CONDE LOPEZ JOSE MANUEL
CC: Manuel Quecedo; ALEJANO MONGE CONSUELO; LLORET LLORCA, Miriam; REY GAYO JOSE MARIA
Asunto: RE: ENUSA Creep Samples

Dear Bob and Jose,

In a subsequent email, I will send you our procedure for grinding, polishing, and etching. We do have to modify etching time depending on the high-burnup cladding alloy (Zry-2, Zry-4, ZIRLO, M5). Also, we have found that we need to use a fresh etching solution for each met mount.

However, what works for us may or may not work for you.
It depends on the power and age of the microscope as well.

Prior to June 2005, our microscope was part of one of the Alpha-Gamma Hot Cell Facility (AGHCF) workstations. It was an old Leitz model. After programmatic work was no longer allowed in AGHCF, we purchased a new Leica DMI5000 M microscope in 2006 and set it up in a Pb-glass-shielded glove box, along with mounting, grinding, polishing, and etching equipment. The procedure we will send you was developed specifically for this microscope. The image quality improved significantly as compared to the in-hot-cell old Leitz microscope we used to operate.

I converted one of your "tif" files to "jpg" so I could adjust contrast and brightness. This did not help to improve the image, which appears to be too dark for the very low level of hydrogen content in the sparse hydrides outside of the liner region. Also, the metal surface appears to be a little too rough (may have to do final polishing step with finer grit paper).

We have been able to get very good images for M5 with 70-100 wppm hydrogen.
I suspect that the hydrogen content in the enclosed image is <50 wppm in visible hydrides. Most of the hydrogen in Zry-2 would tend to migrate to the Zr-liner during long-term creep tests or even in shorter-term simulated drying-storage tests with very slow (about 0.5C/h) cooling rates.

Hopefully, the procedure we will send to you will help. However, there is no single "recipe" that works in all "kitchens". This makes cooking and optical-microscope imaging more of an art than a science.

Mike

-----Original Message-----
From: Einziger, Robert [mailto:Robert.Einziger@nrc.gov]
Sent: Wednesday, February 06, 2013 9:40 AM
To: CONDE LOPEZ JOSE MANUEL; Billone, Michael C.
Cc: Manuel Quecedo; ALEJANO MONGE CONSUELO; LLORET LLORCA, Miriam; REY GAYO JOSE MARIA
Subject: RE: ENUSA Creep Samples

Jose,

I would like to sit down with you when you come to the RIC. I don't do the etching and I don't know if Mike is going to be at the RIC so I am going to forward this e-mail on to him so that he can communicate the etching technique directly with you.

See you in March

REE

-----Original Message-----
From: CONDE LOPEZ JOSE MANUEL [mailto:jmcl@csn.es]
Sent: Wednesday, February 06, 2013 9:59 AM
To: Robert Einziger
Cc: Manuel Quecedo; ALEJANO MONGE CONSUELO; LLORET LLORCA, Miriam; REY GAYO JOSE MARIA
Subject: RE: ENUSA Creep Samples

Dear Bob,

I hope you are doing well. I would like to come back to the micrographs of BWR cladding that Miriam sent you, and which Mike couldn't use to determine hydride continuity due to lack of contrast.

We continue to perform different tests with irradiated cladding, and would like to be aware of the kind of etching that is needed, in order to obtain better results in the future that may allow for the analysis.

As you know, I will be attending the RIC and will participate in the same session than you. I would appreciate if you could find some time during the week to discuss this matter a little more in depth.

Look forward to see you in DC. Thanks in advance

Jose

José M. Conde
Jefe de la Unidad de I+D

Consejo de Seguridad Nuclear
Tel.: +34913460253
Fax: +34913460588

-----Mensaje original-----
De: Einziger, Robert [mailto:Robert.Einziger@nrc.gov] Enviado el: jueves, 31 de enero de 2013 13:15
Para: LLORET LLORCA, Miriam
Asunto: FW: ENUSA Creep Samples

Miriam,

I sent the micrographs onto Mike Billone for analysis. Below is his response.

Thanks for the effort

REE

From: Billone, Michael C. [mailto:billone@anl.gov]
Sent: Wednesday, January 23, 2013 2:49 PM
To: Einziger, Robert
Subject: RE: ENUSA Creep Samples

Bob,

The micrographs are not high enough in contrast for me to determine continuity of radial hydrides.
Very careful etching (HF strength and time) is required, and the recipe varies with material to get better contrast between hydrides and metal

Based on what we can see, the hydrogen content in the Zry-2, in the non-liner area, appears to very low (<70 wppm).

Also, for creep tests, it would be prudent to measure the post-test hydrogen content to ensure that hydrogen was not lost to the end fixtures during time at high temperature.

Sorry I could not be of more help,

Mike

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Einziger, Robert

From: Billone, Michael C. <billone@anl.gov>
Sent: Tuesday, February 05, 2013 12:41 PM
To: Sven Bader (sven.bader@areva.com); Kris Cummings (cumminkw@westinghouse.com); Einziger, Robert; Brady Hanson (brady.hanson@pnl.gov); raheel.haroon@areva.com; John Kessler; Donghak Kook (syskook@kaeri.re.kr); Miriam Lloret (mll@enusa.es); zimartin@tva.gov; Keith Waldrop (kwaldrop@epri.com); Yong Yan (yany@ornl.gov)
Cc: Albert Machiels; Liu, Yung Y.; Han, Zenghu; pvp@enusa.es; Manuel Quecedo (MQG@enusa.es)
Subject: Revised Minutes of Conference Phone Call
Attachments: Minutes_Conf Call_012213_R1.doc

Importance: High

In my haste to send out the minutes, I forgot to remove "Draft" from the title.
Please use enclosed with "R1" at the end of the file name.

Mike

Dear Colleagues,

Enclosed are the minutes for the conference call with EPRI on Jan. 22, 2013 (see last Word file).
I want to thank Miriam Lloret and Albert Machiels for their input to the draft minutes.
Again, I thought it was a very productive exchange and a good way for the Subcommittee to initiate activities for 2013.

The activities of the ESCP Fuels-Internals Subcommittee are ongoing. Feel free to continue to comment on the minutes and/or update us on data that are in the public domain or qualitative trends for data that remain proprietary.

Mike

99

Einziger, Robert

From: Raynaud, Patrick
Sent: Monday, March 25, 2013 6:58 AM
To: Einziger, Robert
Cc: Scott, Harold; Dunn, Darrell
Subject: RE: EST Cladding Stress tasks 1 and 2 draft
Attachments: Cladding Stress v1.xlsx

Bob,
Harold noticed an error: I accidentally used the OD to calculate the inner radius of te cladding, instead of using the ID. This correction brings the stresses down by about 10%. See attached.
Patrick

From: Raynaud, Patrick
Sent: Friday, March 22, 2013 9:46 AM
To: Einziger, Robert
Cc: Scott, Harold; Dunn, Darrell
Subject: RE: EST Cladding Stress tasks 1 and 2 draft

Bob, see attached revised version based on your comments.

I protected the worksheets so that one would not inadvertently make a change, but there is no password, so you can unprotect them by right-clicking on the tab for a given worksheet.
The only unprotected cells that one can change are the ones in red in the "Stress Calculation" worksheet.
The different cases are as follows:

- Base irradiation
  - Gas production to 65 GWd/MTU from either ORIGEN or FRAPCON
    - 0.1682 mol/rod for ORIGEN
    - 0.1447 mol/rod for FRAPCON
  - Fission gas release (FGR) at 65 GWd/MTU (based on the moles of gas in the free void volume of the rod predicted by FRAPCON) from wither ORIGEN, FRAPCON, or 100% gas release (which was the case I sent you previously)
    - 2.56% for ORIGEN
    - 2.97% for FRAPCON
- Extended storage
  - Gas production from either ORIGEN or FRAPCON (Note that the gas produced after discharge is given for ORIGEN, and if you assume the FRAPCON gas production instead of the ORIGEN gas production, then I scaled the ORIGEN production back to match the FRAPCON value at 65GWd/MTU)
  - Gas release fraction to be specified by user between 0% and 100%

So the most conservative case is taking production from ORIGEN and FGR from FRAPCON for the base irradiation, and ORIGEN gas production during storage with 100% release. This case yields a maximum hoop stress in the cladding of 93.2 MPa.

The least conservative case is taking production form FRAPCON and FGR from ORIGEN for the base irradiation, and FRAPCON gas production during storage with 100% release. This case yields a maximum hoop stress in the cladding of 88.6 MPa. The difference in maximum hoop stress predicted is 4.6 MPa (from 93.2 MPa to 88.6 MPa), which is ~5% relative difference

The rod internal pressure is based on the total number of moles of gas in the rod, equal to:

E-103

Total gas = initial gas + released fission gas + gas produced after discharge (assuming in the cases described above that 100% is released to the void volume in the rod)

In the end, if you play with how much of the gas produced after discharge is released to the gap and look at the stress plot, you can see the effect of bounding assumptions such as: 100% release, or 0% release. The difference in stress between these two assumptions after 2000 years is around 35 MPa for the hoop stress.

I think this Excel sheet will allow us to make rather rapid determinations as we move forward. We will talk more on Tuesday and I can answer questions then.

PAtrick

From: Einziger, Robert
Sent: Thursday, March 21, 2013 2:57 PM
To: Raynaud, Patrick
Subject: RE: EST Cladding Stress tasks 1 and 2 draft

The rod pressures seem much too high.

1- 88 mol gas/ MTU is .0168 mol/rod based on 2.2 kg/rod, not .168 mol/rod.
2- What fgr fraction did you use or are the moles of gas the released mols?

From: Raynaud, Patrick
Sent: Thursday, March 21, 2013 1:29 PM
To: Einziger, Robert
Cc: Dunn, Darrell; Scott, Harold
Subject: EST Cladding Stress tasks 1 and 2 draft

Bob,

I was going to send this right after our call, but since you sent me some canister temperature histories, I though I'd update it before sending. We can discuss Tuesday when we meet. For now, I just wanted you to have this. Note that the predictions for canister temperature were not provided beyond 50 years, so I fitted the data to 50 years and extended with a logarithmic function. We can discuss the validity of that assumption. The moles of gas produced over time were provide by ORNL from an ORIGEN calculation for a 17x17 PWR rod at 65GWd/MTU. This is a bounding case in terms of rod internal pressure.

I am assuming the canister fill pressure is 1 atm (0.1 MPa) at 100°C, and this calculation assumes (per the work task descriptions) that ALL the gas produced is released to the gap. This yields huge stresses, beyond the yield stress of the cladding, but is expected with such a conservative assumption. In reality, the fission gas release is small (a few percent), so you can assume that the pressures shown here will decrease by at least a factor of 5 for a more realistic case.

We can talk more Tuesday and I can explain the assumptions and methods, as well as answer questions you have.

Patrick

Patrick A.C. Raynaud, PhD
Reactor Systems Engineer (Fuels)
U.S. Nuclear Regulatory Commission
RES/DSA/FSCB
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Washington, DC 20555

Tel: (+1) 301-251-7542
patrick.raynaud@nrc.gov

Assuming all the gas produced is released to the gap

Burnup	65 GWd/MTU	
R	8.3144621 cm^3 MPa K^{-1} mol^{-1}	
Void volume		13.2 cm^3
Temperature	400 C	673.15 K
Clad OD	0.37 in	9.398 mm
Clad ID	0.322 in	8.1788 mm
Clad thickness	0.024 in	0.6096 mm
Clad Ri	0.161 in	4.0894 mm
Clad Ro	0.185 in	4.699 mm
Canister fill pressure		0.1 MPa
Canister fill temperature	100 C	373.15 K

concentrations moles per MTU (years)	h	he	kr	n	ne	rn
0	5.802E-02	2.604E+00	8.061E+00	9.103E-04	3.587E-04	4.485E-14
1	5.587E-02	2.765E+00	8.029E+00	9.114E-04	3.587E-04	8.959E-14
3	5.192E-02	2.902E+00	7.970E+00	9.136E-04	3.587E-04	1.761E-13
10	4.112E-02	3.268E+00	7.816E+00	9.214E-04	3.587E-04	3.437E-13
20	3.149E-02	3.731E+00	7.689E+00	9.324E-04	3.587E-04	3.898E-13
30	2.600E-02	4.142E+00	7.622E+00	9.434E-04	3.587E-04	4.105E-13
50	2.109E-02	4.850E+00	7.568E+00	9.654E-04	3.587E-04	5.184E-13
100	1.887E-02	6.229E+00	7.549E+00	1.020E-03	3.587E-04	1.320E-12
200	1.872E-02	8.216E+00	7.548E+00	1.129E-03	3.587E-04	5.761E-12
300	1.872E-02	9.685E+00	7.548E+00	1.236E-03	3.587E-04	1.457E-11
400	1.872E-02	1.086E+01	7.548E+00	1.342E-03	3.587E-04	2.799E-11
500	1.872E-02	1.185E+01	7.548E+00	1.446E-03	3.587E-04	4.604E-11
600	1.872E-02	1.270E+01	7.548E+00	1.550E-03	3.587E-04	6.861E-11
800	1.872E-02	1.411E+01	7.548E+00	1.753E-03	3.587E-04	1.267E-10
1000	1.872E-02	1.525E+01	7.548E+00	1.951E-03	3.587E-04	2.008E-10
1200	1.872E-02	1.619E+01	7.548E+00	2.144E-03	3.587E-04	2.897E-10
1400	1.872E-02	1.699E+01	7.548E+00	2.333E-03	3.587E-04	3.920E-10
1600	1.872E-02	1.769E+01	7.548E+00	2.517E-03	3.587E-04	5.066E-10
1800	1.872E-02	1.831E+01	7.548E+00	2.697E-03	3.587E-04	6.325E-10
2000	1.872E-02	1.888E+01	7.548E+00	2.873E-03	3.587E-04	7.686E-10

Gas moles in free volume		
Initial Gas	2.22E-02 mol	
	Production(mol)	
FGR	ORIGEN	FRAPCON
ORIGEN	4.300E-03	3.698E-03
FRAPCON	4.999E-03	4.299E-03
100%	1.682E-01	1.447E-01

Canister Temperature	
11.6788487	-5.018E-07
564.027772	564.027773

	Base Irradiation						
	Production ORIGEN		**FGR** ORIGEN				
	Storage						
	Production ORIGEN		**Release (%)** 100				
	TOTAL	**Increase**	**TOTAL**	**Increase**	**Canister Conditions**		**Rod**
					Temperature	**Pressure**	**Pressure**
xe	**mol**	**mol**	**mol/rod**	**mol/rod**	**K**	**MPa**	**(Mpa)**
7.762E+01	**8.834E+01**	**0.000E+00**	**1.682E-01**	**0.000E+00**	**673.15**	**0.18**	**11.2**
7.764E+01	**8.849E+01**	**1.469E-01**	**1.685E-01**	**2.796E-04**	**659.77**	**0.18**	**11.1**
7.764E+01	**8.857E+01**	**2.209E-01**	**1.686E-01**	**4.206E-04**	**637.22**	**0.17**	**10.8**
7.764E+01	**8.877E+01**	**4.221E-01**	**1.690E-01**	**8.038E-04**	**575.06**	**0.15**	**9.9**
7.764E+01	**8.909E+01**	**7.485E-01**	**1.696E-01**	**1.425E-03**	**532.18**	**0.14**	**9.3**
7.764E+01	**8.943E+01**	**1.087E+00**	**1.703E-01**	**2.070E-03**	**508.18**	**0.14**	**9.1**
7.764E+01	**9.008E+01**	**1.736E+00**	**1.715E-01**	**3.306E-03**	**477.93**	**0.13**	**9.0**
7.764E+01	**9.144E+01**	**3.094E+00**	**1.741E-01**	**5.891E-03**	**436.90**	**0.12**	**8.9**
7.764E+01	**9.342E+01**	**5.080E+00**	**1.779E-01**	**9.673E-03**	**395.86**	**0.11**	**9.0**
7.764E+01	**9.489E+01**	**6.549E+00**	**1.807E-01**	**1.247E-02**	**371.86**	**0.10**	**9.1**
7.764E+01	**9.607E+01**	**7.724E+00**	**1.829E-01**	**1.471E-02**	**354.83**	**0.10**	**9.2**
7.764E+01	**9.706E+01**	**8.714E+00**	**1.848E-01**	**1.659E-02**	**341.62**	**0.09**	**9.3**
7.764E+01	**9.791E+01**	**9.564E+00**	**1.864E-01**	**1.821E-02**	**330.82**	**0.09**	**9.3**
7.764E+01	**9.932E+01**	**1.097E+01**	**1.891E-01**	**2.090E-02**	**313.79**	**0.08**	**9.4**
7.764E+01	**1.005E+02**	**1.211E+01**	**1.913E-01**	**2.307E-02**	**300.58**	**0.08**	**9.4**
7.764E+01	**1.014E+02**	**1.305E+01**	**1.931E-01**	**2.486E-02**	**289.79**	**0.08**	**9.4**
7.764E+01	**1.022E+02**	**1.386E+01**	**1.946E-01**	**2.638E-02**	**280.66**	**0.08**	**9.3**
7.764E+01	**1.029E+02**	**1.456E+01**	**1.959E-01**	**2.772E-02**	**272.76**	**0.07**	**9.3**
7.764E+01	**1.035E+02**	**1.518E+01**	**1.971E-01**	**2.890E-02**	**265.78**	**0.07**	**9.3**
7.764E+01	**1.041E+02**	**1.575E+01**	**1.982E-01**	**2.998E-02**	**259.55**	**0.07**	**9.2**

THIN WALL		
σl = p d / 4 t		
σh = p d / 2 t		
THICK WALL		
σa = (pi ri2 - po ro2)/(ro2 - ri2)		
σc = [(pi ri2 - po ro2) / (ro2 - ri2)] - [ri2 ro2 (po - pi) / (r2 (ro2 - ri2))]		
σr = [(pi ri2 - po ro2) / (ro2 - ri2)] + [ri2 ro2 (po - pi) / r2 (ro2 - ri2)]		
R in thick wall calculations		
fraction of thickness		0 (ID)
R	0.161 in	4.0894 mm

Check Volume (cm3)	THIN WALL STRESS		THICK WALL STRESS		
	Hoop (Mpa)	Axial (Mpa)	Hoop (Mpa)	Axial (Mpa)	Radial (Mpa)
13.2	**75.2**	**37.6**	**79.7**	**34.3**	**-11.2**
13.2	**74.5**	**37.3**	**79.0**	**33.9**	**-11.1**
13.2	**72.4**	**36.2**	**76.7**	**33.0**	**-10.8**
13.2	**66.2**	**33.1**	**70.2**	**30.2**	**-9.9**
13.2	**62.7**	**31.3**	**66.5**	**28.6**	**-9.3**
13.2	**61.2**	**30.6**	**65.0**	**27.9**	**-9.1**
13.2	**60.1**	**30.0**	**63.8**	**27.4**	**-9.0**
13.2	**59.7**	**29.9**	**63.5**	**27.3**	**-8.9**
13.2	**60.4**	**30.2**	**64.4**	**27.7**	**-9.0**
13.2	**61.2**	**30.6**	**65.2**	**28.0**	**-9.1**
13.2	**61.7**	**30.9**	**65.8**	**28.3**	**-9.2**
13.2	**62.1**	**31.1**	**66.3**	**28.5**	**-9.3**
13.2	**62.4**	**31.2**	**66.7**	**28.7**	**-9.3**
13.2	**62.8**	**31.4**	**67.1**	**28.9**	**-9.4**
13.2	**62.9**	**31.4**	**67.2**	**28.9**	**-9.4**
13.2	**62.8**	**31.4**	**67.2**	**28.9**	**-9.4**
13.2	**62.7**	**31.3**	**67.0**	**28.8**	**-9.3**
13.2	**62.4**	**31.2**	**66.8**	**28.7**	**-9.3**
13.2	**62.2**	**31.1**	**66.5**	**28.6**	**-9.3**
13.2	**61.9**	**30.9**	**66.2**	**28.5**	**-9.2**

100

Einziger, Robert

Subject: FW: EPRI ESCP CISCC subcommittee meeting
Location: your phone

Start: Tue 04/16/2013 2:00 PM
End: Tue 04/16/2013 4:00 PM
Show Time As: Tentative

Recurrence: (none)

Meeting Status: Not yet responded

Organizer: BRACEY William (TRANSNUCLEAR INC)

When: Tuesday, April 16, 2013 2:00 PM-4:00 PM (GMT-05:00) Eastern Time (US & Canada).
Where: your phone

Note: The GMT offset above does not reflect daylight saving time adjustments.

~~*~*~*~*~*~*~*~*

Both should sit in on this.

-----Original Appointment-----
From: BRACEY William (TRANSNUCLEAR INC)
Sent: Monday, April 08, 2013 12:00 PM
To: BRACEY William (TRANSNUCLEAR INC); 'allen@engr.wisc.edu'; BADER Sven O (AFS); 'Birk, Sandra'; 'Brad Black'; BROWN James (TRANSNUCLEAR INC); 'bruce.wiersma@srnl.doe.gov'; 'Bryan, Charles'; 'carl.friant@cengllc.com'; 'Caseres, Leonardo J.; 'chuck.merritt@cengllc.com'; 'Connell, Jim'; 'Danner, Tom'; 'darrell.dunn@nrc.gov'; 'darrylbutt@boisestate.edu'; 'Deboi, Kristi'; 'Edwards, Steve'; Einziger, Robert; 'Enos, David G'; 'Farnum, Cathy Ottinger'; 'Garg, Krishan K'; Gavrilas, Mirela; Gordon, Matthew; 'Hollinger, Gary'; 'hvymet@mit.edu'; Jacobs, Christian; 'jeffery.england@srnl.doe.gov'; 'Kessler, John'; 'Kumar Sridharan'; 'Laszlo Zsidai'; 'mark.dupont@srnl.doe.gov'; 'Massari, John'; 'McCULLUM, Rodney; 'Mostafa Mostafa'; Oberson, Greg; Rubenstone, James; 'sara.depaula@nrc.gov'; 'Sebastien P Teysseyre/TEYSSP/FN/INEEL/US'; 'seferry@mit.edu'; SERRES Aurelie (BE/LO); 'SHIRAI Koji; 'Stockman, Christine; 'tanij@criepi.denken.or.jp'; Tarantino, David; 'Todd Mintz'; 'Valenta, Heidi M; 'Waldrop, Keith; 'wataru@criepi.denken.or.jp'; 'Weiner, Ruth'; 'Xihua He'; JUNG Andy (TRANSNUCLEAR INC); 'Paul Gossen'; Randall.Granaas@sce.com; wneuburger@gmail.com
Subject: EPRI ESCP CISCC subcommittee meeting
When: Tuesday, April 16, 2013 2:00 PM-4:00 PM (GMT-05:00) Eastern Time (US & Canada).
Where: your phone

I must apologize to the subcommittee that I have not issued minutes from the Charlotte meeting. I will review my notes and get something out this week. Meanwhile, we need to prepare for St. Petersburg, May 6.

Keith, please set up the conference line, and distribute the information to the mailing list.

Phone call agenda (let me know if you have subjects to add)
Keep your reports brief, there is a lot to cover in only two hours. More detail will be taken up at St. Petersburg:

E104

1. Subcommittee leadership for 2013: Laszlo Zsidai of Holtec was nominated to take my place. Has he accepted? If not, who else can we volunteer?
2. Chemical analysis of Calvert Cliffs samples – Keith Waldrop

3. Dry sampler
4. Wet sampler (SaltSmart)
5. Lessons for future deployments
6. Changes at Louisville Solutions, future support for Salt Smart nuclear applications

7. Status of Calvert Cliffs license extension and response to NRC question on CISCCI – John Massari
8. Status of surface sampling and inspection at Hope Creek/Salem – Laszlo Zsidai
9. Residual weld stresses

10. Preparation and measurement of welded specimens – Ron Ballinger
11. Weld sample clearing house - Sebastien Teysseyre
12. Finite element analysis of residual stresses – Andy Jung or Massari

13. Air sampling

14. Importance of data on composition of aerosols at ISFSIs – Jung
15. Progress on finding correct instrument or filter for volumetric sampler – Charles Bryan

16. Status of research at universities, DOE, NRC, CRIEPI - various

17. What do universities need from industry? Drawings? Other design info?
18. Progress on NEUP projects
19. Any new research at DOE, NRC, CRIEPI, etc.?
20. Upcoming conferences
21. Industry input to DOE NEUP

22. EPRI R&D Roadmap status – Keith Waldrop

104

From: Campbell, Debbie
To: garrido.david@ensa.es; gary.hollinger@fpl.com; gary_cannell@rl.gov; Zigh, Ghani; glenn.grant@pnnl.gov; glenn.schwartz@pseg.com; glez.garmendia.rafael@ensa.es; Oberson, Greg; gregory.hall@icp.doe.gov; Selby, Greg; Haaroen.Sataar@eskom.co.za; haddad@cnea.gov.ar; halimalsaed@environuclear.net; hans.codee@covra.nl; heidi.valenta@cenglic.com; heinz.geiser@gns.de; Graves, Herman; herve.issard@areva.com; hhhsu@iner.gov.tw; Gonzalez, Hipolito; hiung@swri.org; holger.voelzke@bam.de; howardrl1@ornl.gov; hpfeifer@nacintl.com; hvymet@MIT.EDU; ian.wilson@hse.gsi.gov.uk; iaherrera@socoin.es; james.brown@areva.com; james.rubensone@nrc.gov; jan.vanderlee@edf.fr; jap@enusa.es; jconnell@3yankees.com; jeblanco@unionfenosa.es; jedai@khnp.co.kr; jeff.soltis@bwgus.com; jeff.williams@hq.doe.gov; jeffery.england@srnl.doe.gov; jens.schroeder@gns.de; jgab@enresa.es; jhopf@energysolutions.com; Kessler, John; jmcl@csn.es; joe.carter@srs.gov; john.bennett@edfenergy.com; john.massari@cenglic.com; john.shea@edf-energy.com; johnlambert@anl.gov; johnny.floyd@exeloncorp.com; joseantonio.gago@endesa.es; josey.lopez@nuclenor.es; jubinrt@ornl.gov; julian.robertshaw@edf-energy.com; Wall, James; Guttmann, Jack; k.niyogi@holtec.com; kamcmah@sandia.gov; katayama-Jiro@jnes.go.jp; kato-masami@jnes.go.jp; kbsoren@sandia.gov; kcole@nacintl.com; Edsinger, Kurt; keith.norwood@british-energy.com; kemal.pasamehmetoglu@inl.gov; kevin.morris@transnuclear.com; kitamura.takafumi@jaea.go.jp; krishan.garg@cenglic.com; kristi.deboi@sce.com; kumar@engr.wisc.edu; Waldrop, Keith; l.francia@unesa.es
Cc: Hanson, Brady D; Waldrop, Keith; Campbell, Debbie
Subject: ESCP Invitation to ASTM C26 Meeting
Date: Wednesday, March 06, 2013 11:50:32 AM
Attachments: ASTM_Flyer.pdf
C26_Short_Course_Flyer.pdf
Agenda.pdf
Importance: High

Sent on behalf of Brady Hanson, chair, Extended Storage Collaboration Program Fuels and Internals Subcommittee

---

Dear ESCP participants,

As a long time member of the ASTM C26 Committee on the Nuclear Fuel Cycle, I would like to invite you to participate in the upcoming meeting Sunday, June 16 - Friday, June 21, 2013 in Avignon, France. And I need your help to volunteer to present at the workshop!!

Details of the meeting can be found at

http://www.astm.org/C26AvignonFrance0613

and in the attached conference flyer, agenda, and flyer on the short courses.

I have been asked to put together the workshop on Used/Spent Fuel Disposition to take place on Thursday, June 20.

My plans are to have the workshop as follows:

1) Presentation and discussion of ASTM International standards within subcommittee C26.13 of importance to used fuel for storage, transportation, and disposal. Brady Hanson, PNNL, USA

These standards include ones on Spent Fuel Dissolution, Al Fuel Dissolution, Drying, Characterization, Pyrophoricity, Materials for Extended Storage, and the famous C1174 on Prediction of Long-Term Behavior of Materials in Geologic Disposal.

2) Storage and Transportation Status

It would be good to have ~20-30 minute presentations from multiple countries to provide:

- Status of storage and transportation in each county
- Policy issues
- Technical issues
- Technical data gaps
- R&D being performed
- Standards that would be of value to your program

Presentations could be from (looking for volunteers!):

E 105

US Ken Sorenson, SNL
UK
France
Germany
Japan
Korea
Others ????

3) Lunch

4) Geologic Disposal Status

It would be good to have ~20-30 minute presentations from multiple countries to provide:

- Status of storage and transportation in each county
- Policy issues
- Technical issues
- Technical data gaps
- R&D being performed
- Standards that would be of value to your program

Presentations could be from (looking for volunteers!!):

US Peter Swift, SNL
France
Sweden
Finland
Germany
Others ????

5) How Storage and Transportation Options Affect Disposal Options - Tito Bonano, SNL, USA

6) Reprocessing

Status, issues, etc., and if ASTM standards would be of value

UK
France/US Paul Murray, AREVA, US
Japan

If you are interested and think you will be able to attend and give a presentation in one of these areas, please let me know by March 19 so that we can finalize the program. If you are just interested in attending, please let me know as well.

Thanks for your help,

Brady D. Hanson
Staff Scientist
Radiochemical Science & Engineering Group Pacific Northwest National Laboratory
902 Battelle Boulevard
P.O. Box 999, MSIN P7-27
Richland, WA 99352 USA
Tel: 509-375-5051
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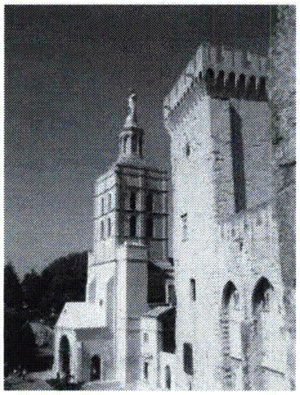
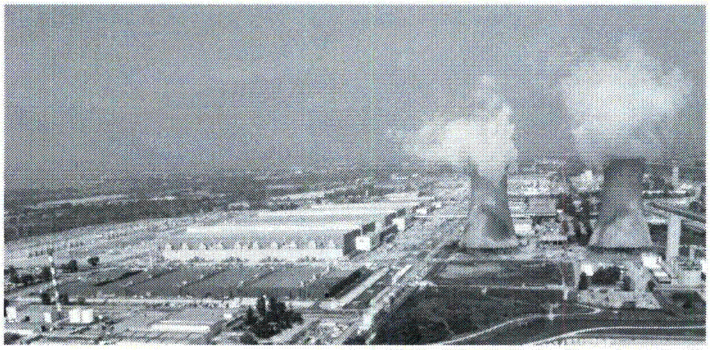
brady.hanson@pnnl.gov
www.pnnl.gov

AN ADVANCED COURSE ON THE NUCLEAR FUEL CYCLE





Avignon June 16-18, 2013

Hotel Mercure Pont d'Avignon

Sponsored by ASTM and INSTN

Scope :

This course will be the opportunity for Scientists, Lab Managers and Experts from the Uranium industry and Academic field to update their knowledge on the Nuclear fuel cycle : current state of the art and new trends.

The workshop can be coupled to a visit of Areva Tricastin and CEA Marcoule sites on Wednesday 19 and to ASTM meetings C26-02, C26-05 and C26-13 on Monday 18 and Tuesday 19.

It will be a exceptional event with international participation from the US and French facilities. This course is designed by experts in the fuel cycle.

Registration :

Through the ASTM webpage: http://www.astm.org/C26AvignonFrance0613

Technical program :

Sunday June 16

13h30 -15h00 Uranium Mining, from leaching and liquid/liquid extraction to precipitation.
AREVA.

15h00-16h30 : Conversion, from the oxides to the fluoride, dry process, wet process.
HONEYWELL and CAMECO

16h30-18h00 : UF6 enrichment with centrifugation,
URENCO, TVEL

Monday June 17

13h30 -15h00 Fuel fabrication, from UF6 to UO2 conversion, pellets and cladding,
WESTINGHOUSE, AREVA

15h00-16h30 : Recycling Highly enriched uranium (HEU),
DOE/Y12

16h30-18h00 New reactors (Gen 4, small modular reactors), new fuels
CEA, INL

Tuesday June 18

13h00 -14h30 Used fuel / spent fuel / recycling
DOE/SNL, AREVA

14h30-16h00 : Spent fuel processing, advanced actinide separation
CEA and ROSATOM

16h00-17h30 Waste processing, waste forms
DOE/PNNL, AREVA

17h30-18h30 : Flexible radwaste packaging
AIEA

ASTM C26 on Nuclear Fuel Cycle
Short Courses and Workshops
Avignon, France

June 16-21, 2013
Mercure Pont d'Avignon Hotel
Avignon, France

Event Contacts: Dale Wahlquist
Idaho National Laboratory
Idaho Falls, ID
USA

Bertrand Morel
Comurhex
Pierrerlatte
France

Sunday, June 16, 2013

Short Courses on Nuclear Fuel Cycle

- 1:30 pm - 3:00 pm: Uranium Mining, from Leaching and Liquid/Liquid Extraction to Precipitation.
- 3:00 pm - 4:30 pm: Conversion, from the Oxides to the Fluoride, Dry Process, Wet Process.
- 4:30 pm - 6:00 pm: UF6 Enrichment with Centrifugation

Monday, June 17, 2013

Short Courses on Nuclear Fuel Cycle

- 12:00 pm – 1:00 pm: Catered lunch provided free of charge for those attending the short courses
- 1:00 pm – 2:30 pm: Fuel Fabrication, from UF6 to UO2 Conversion, Pellets and Cladding
- 2:30 pm – 4:00 pm: Recycling Highly Enriched Uranium (HEU)
- 4:00 pm – 5:30 pm: Advanced Fuels and Reactors (Gen 4, small modular reactors, new fuels)

Tuesday, June 18, 2013

Short Courses on Nuclear Fuel Cycle

- 12:00 pm – 1:00 pm: Catered lunch for those attending the short courses only
- 1:00 pm – 2:30 pm: Used Fuel/Spent Fuel/Recycling
- 2:30 pm – 4:00 pm: Spent Fuel Processing, Advanced Actinide Separation
- 4:00 pm – 5:30 pm: Waste Processing, Waste Forms
- 5:30 pm – 6:30 pm: Flexible Rad Waste Packaging

Workshop on Remote Equipment Design and Short Courses on Nuclear Fuel Cycle

Sponsored by C26.14 on Remote Systems, this workshop will focus on the ASTM guides related to the design of equipment for use in remote facilities. Presentations will focus on hot cell shielding windows, remote handling equipment, general design considerations for remote equipment, radiation hardened cameras and hot cell viewing systems, hot cell measuring equipment, and design of fluid connectors for use in hot cells.

Design of Equipment for Use in Hot Cells

- 8:00 am – 8:30 am: Introduction and Overview of ASTM Remote System Standards
- 8:30 am – 9:00 am: Effective Development of Codes and Standards Nurtures Cost Effective Implementation and Promotes Customer Satisfaction
- 9:00 am – 9:30 am: ASTM C1533 – General Design Considerations for Hot Cell Equipment
- 9:30 am – 10:00 am: ASTM C1554 - Remote Material Handling Equipment Design Considerations
- 10:00 am – 10:15 am: Break
- 10:15 am – 11:00 am: ASTM-1572 – Design and Selection of Hot Cell Shielding Windows for Remote Facilities
- 11:00 am – 11:30 am: Latest Technology in Radiation Hardened Cameras
- 11:30 am – 12:00 pm: ASTM C1615 – Design of Viewing Systems for Hot Cell Facilities
- 12:00 pm – 1:00 pm: Lunch
- 1:00 am – 1:45 am: ASTM DRAFT Standard Guide for Measurement and Test Equipment for Remotely Operated Facilities
- 1:45 am – 2:15 pm: ASTM WK24127 – DRAFT Standard Guide for Fluid Transfer-Containment Components and Systems for Remotely Operated Facilities
- 2:15 pm – 2:30 pm: Break
- 2:30 pm – 3:00 pm: ASTM C1615 – Design of Mechanical Drive Systems for Remote Facilities
- 3:00 pm – 3:30 pm: ASTM C1725 - Design of Specialized Support Equipment and Tools for Hot Cells
- 3:30 pm – 4:15 pm: Lessons Learned From Hot Cell Window Failures

Wednesday, June 19, 2013

Tours

Tour two facilities: Tricastin and Melox (Marcoule). Two tour groups. One group tours Tricastin while the other tours Melox, then switch after lunch. Leave hotel at 8:00 am. Each group meets separately for lunch (lunch cost included with tour fee).

- Tricastan is in Peirrelatte. It has a conversion UF6 plant and UF6 enrichment facility. It also has activities on reprocessed uranium and UF6 de-conversion to U3O8.
- In Marcoule, there is a MOX fuel fabrication facility (requires masks), Melox, and hot cells in Atalante where RD is performed on actinides.
- Those taking the tours will need to provide a pdf of their passport in addition to information such as current work affiliation, current home address, etc. Information must be provided three months prior to the tours.

Thursday, June 20, 2013

Workshop on Long Term Glass Performance

- 8:30 am to 5:00 pm: Detailed information on this workshop will be available soon.
- 12:30 pm to 2:00 pm: Catered lunch provided for those attending the workshops.

Workshop on Spent Fuel Disposal

- 8:30 am to 5:00 pm: Detailed information on this workshop will be available soon.
- 12:30 pm to 2:00 pm: Catered lunch provided for those attending the workshops.

Workshop on Analytical Development and Reference Materials in the Nuclear Fuel Cycle

Reference Materials and Proficiency Testing

- 8:30 am – 9:00 am Evaluating the Needs and Challenges for Reference Materials in the Nuclear Industry.
- 9:00 am – 9:45 am: Development of Reference Materials for Isotopic Analysis and PTS
- 9:45 am – 10:30 am: Reference Materials and Precision and Bias (GUM)
- 10:30 am – 11:00 am: Break
- 11:00 am – 11:45 am: Validation of Test Method Using Reference Materials
- 11:45 am – 12:30 pm: Metrology for Activity and Dose Measurement

- 12:30 pm – 2:00 pm: Lunch

Analytical Developments

- 2:00 pm – 2:45 pm: New Trends in Analytical Development for Nuclear Laboratories
- 2:45 pm – 3:15 pm: Hyphenation, State of the Art
- 3:15 pm – 3:45 pm: Speciation in Nuclear Matrices

Actinide Analysis

- 4:15 pm – 4:45 pm: Summary of Actinide Characterization Capabilities
- 4:45 pm – 5:15 pm: Analysis of Trace Actinides and Other Radionuclides in Bulk Uranium
- 5:15 pm – 5:45 pm: Development of MOX Analysis

Friday, June 21, 2013

Workshop on Analytical Development and Reference Materials in the Nuclear Fuel Cycle

- 8:30 am – 9:00 am: On-line UF6 Isotopic Analysis
- 9:00 am – 9:30 am: UF6 On-line Analysis with ICP-MS
- 9:30 am – 10:00 am: Application of Microfluidic Analysis

- 10:00 am – 10:30 am: Break

Nuclear Measurement

- 10:30 am – 11:00 am: Spectroscopic On-line Monitoring of Radiochemical Streams
- 11:00 am – 11:30 am: New Developments in Neutronics Activation
- 11:30 am – 12:00 pm: Enhanced Gamma Imaging Spectroscopy Technologies
- 12:00 am – 12:30 pm Developments in K edge analysis

Workshop on Long Term Glass Performance

- 8:30 am to 5:00 pm: Detailed information on this workshop will be available soon.
- 12:30 pm to 2:00 pm: Catered lunch provided for those attending the workshops.

From:	Dunn, Darrell
Sent:	Monday, March 19, 2012 5:22 PM
To:	Roberto Pabalan; Oberson, Greg
Cc:	Todd Mintz; Xihua He; Ahn, Tae
Subject:	RE: dilution of chloride solution above DRH
Attachments:	Omega DRH data for salts.pdf

I have not confirmed the actual source of the data in the attached file (listed as Wexler and Hasegawa) but it does beg the question on whether sea salt would be less than MgCl2. It certainly seems close but I personally I think the details of the measurements and calculations need to be better understood to make a comparison. Obviously sea salt is less than NaCl and that is not surprising. Does your calculation for sea salt include CaCl2?

From: Roberto Pabalan [mailto:rpabalan@cnwra.swri.edu]
Sent: Monday, March 19, 2012 5:08 PM
To: Oberson, Greg
Cc: Todd Mintz; Xihua He; Dunn, Darrell; Ahn, Tae
Subject: RE: dilution of chloride solution above DRH

Greg,

The 2% value is incorrect – the very small amount of water was causing funky results. I redid the calculations. Below are the results for sea salt deliquescence (MDRH) at 25 to 80 C. The sea salt values are slightly lower compared to a binary NaCl+MgCl2 mixture.

--Bobby

Temperature (°C)	Sea Salt MDRH (%)	$NaCl+MgCl_2$ MDRH (%)
25	33.66	33.94
30	33.41	33.70
35	33.14	33.45
40	32.84	33.16
45	32.49	32.83
50	32.10	32.46
55	31.66	32.04
60	31.15	31.56
65	30.58	31.02
70	29.93	30.41
75	29.20	29.72
80	28.35	28.93

From: Oberson, Greg [mailto:Greg.Oberson@nrc.gov]
Sent: Monday, March 19, 2012 9:13 AM
To: Roberto Pabalan
Cc: Todd Mintz; Xihua He; Darrell Dunn
Subject: RE: dilution of chloride solution above DRH

Bobby,

E106

Are you available for an early phone call, 9 AM ET/8 AM CT tomorrow to discuss this further? Also, the attached email where you calculated DRH 2% for sea salt.

Thanks,
Greg

Oberson, Greg

From: Oberson, Greg
Sent: Monday, April 30, 2012 4:35 PM
To: Dunn, Darrell; Ahn, Tae
Subject: CNWRA SCC update
Attachments: April 30 update.pdf

I am going to be out of the office from Wednesday through next Tuesday. Here is a brief update for what's going on at CNWRA. Nothing seems to be happening with the Task 2 tests (non-coastal atmospheric species). We should consider what to do. They recommend adding ammonium chloride. I would like some more information first concerning the basis for that species. It may be helpful to test the industrial + marine conditions, but to be thoughtful how we go about it. You see for task 1 there is a fair amount of corrosion on the flat specimens and the 52C cycling U-bends. Xihua also told me she thinks they see some cracking on the 0.1 g/m2 specimens at 35C but to be confirmed by cross section. Task 3 tests are ongoing at 60 C/30%RH and 80C/35% RH. The expanded scope contract mod should go to CNWRA this week or next.

Greg

April 5, 2012

MEMORANDUM TO: Brooke D. Poole, Acting Director
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety and Safeguards

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

SUBJECT: STATUS OF USER NEED REQUEST NMSS-2003-003, "DEVELOPING HUMAN RELIABILITY ANALYSIS CAPABILITY SPECIFIC TO MATERIALS AND WASTE APPLICATIONS"

The purpose of this memorandum is to inform you that user need request NMSS-2003-003, "Developing Human Reliability Analysis Capability Specific to Materials and Waste Applications," has been completed. The user need requested RES to work in two phases: (1) a feasibility/scoping phase in which Human Reliability Analysis (HRA) capability needs were identified, and (2) an implementation phase in which identified HRA products were developed based upon the priorities of NMSS staff.

The following products were completed as a part of this user need:

1. S.E. Cooper, "Phase 1: Feasibility Study for Waste Arenas," September 30, 2004
2. J.E. Brewer, P.J. Amico, S.E. Cooper, and S.M. Hendrickson, *Preliminary, Qualitative Human Reliability Analysis for Spent Fuel Handling*, NUREG/CR-7017 (in press)
3. J. Brewer, S. Hendrickson, S. Cooper, and R. Boring, *Human Reliability Analysis-Informed Insights on Cask Drops*, NUREG/CR-7016 (in press)

During this project, RES worked closely with cognizant NMSS experts, including report reviews and feedback. With the completion of the feasibility study and imminent publication of the two NUREG/CRs noted above, the user need response has been completed.

RES has established an online quality survey with which user offices can evaluate the usefulness of RES products and services. This survey can be found at http://portal.nrc.gov/edo/res/OfficeWide/RESQualitySurvey/Lists/RES%20Quality%20Survey/NewForm.aspx?Source=http://portal.nrc.gov/edo/res/OfficeWide/RESQualitySurvey/default.aspx. If your office has not yet completed this brief survey, I would appreciate your support in ensuring its completion (which will take about 5 minutes) within the next 10 working days.

CONTACT: Susan E. Cooper, RES/DRA
301-251-7604

April 5, 2012

MEMORANDUM TO: Brooke D. Poole, Acting Director
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety and Safeguards

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

SUBJECT: STATUS OF USER NEED REQUEST NMSS-2003-003, "DEVELOPING HUMAN RELIABILITY ANALYSIS CAPABILITY SPECIFIC TO MATERIALS AND WASTE APPLICATIONS"

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CONTACT: Susan E. Cooper, RES/DRA
301-251-7604

DISTRIBUTION:
DRA r/f

ADAMS Accession No.: ML120890077

OFFICE	RES/DRA/HFRB	RES/DRA/HFRB	RES/DRA
NAME	S. Cooper	S. Peters	R. Correia
DATE	4/2/12	3/30/12	4/5/12

OFFICIAL RECORD COPY

April 5, 2012

MEMORANDUM TO: Lawrence E. Kokajko, Director
Division of Spent Fuel Alternative Strategies
Office of Nuclear Material Safety and Safeguards

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

SUBJECT: STATUS OF USER NEED REQUEST NMSS-2005-004, "RESEARCH ASSISTANCE IN PREPARING FOR AND PREVIEWING HUMAN RELIABILITY ISSUES EXPECTED IN THE U.S. DEPARTMENT OF ENERGY LICENSE APPLICATION FOR THE YUCCA MOUNTAIN REPOSITORY"

The purpose of this memorandum is to inform you that user need request NMSS-2005-004, "Research Assistance in Preparing for and Previewing Human Reliability Issues Expected in the U.S. Department of Energy License Application for the Yucca Mountain Repository," has been completed. The user need requested RES to assist with the following items related to human reliability:

- Preparing NMSS staff to review the DOE license application
- Reviewing the license application
- Interactions with DOE regarding the NRC's review

Examples of products and services provided by RES to address the assistance requested by NMSS include:

1. Assistance in developing Interim Staff Guidance HLWRS-ISG-004, "Preclosure Safety Analysis – Human Reliability Analysis"
2. Presentation of a seminar titled "HRA Knowledge Transfer for High-Level Waste"
3. Development and documentation of review HRA comments for both pre- and post-closure safety evaluation reports
4. Development of requests for additional information (RAIs)
5. Interactions with DOE, especially with respect to their responses to RAIs

CONTACT: Susan E. Cooper, RES/DRA
301-251-7604

During this project, RES worked closely with cognizant NMSS staff. With the delivery of the products and services listed above, the user need response has been completed.

RES has established an online quality survey with which user offices can evaluate the usefulness of RES products and services. This survey can be found at http://portal.nrc.gov/edo/res/OfficeWide/RESQualitySurvey/Lists/RES%20Quality%20Survey/NewForm.aspx?Source=http://portal.nrc.gov/edo/res/OfficeWide/RESQualitySurvey/default.aspx. If your office has not yet completed this brief survey, I would appreciate your support in ensuring its completion (which will take about 5 minutes) within the next 10 working days.

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DISTRIBUTION:
DRA r/f

ADAMS Accession No.: ML120890068

OFFICE	RES/DRA/HFRB	RES/DRA/HFRB	RES/DRA
NAME	S. Cooper	S. Peters	R. Correia
DATE	4/2/12	3/30/12	4/5/12

OFFICIAL RECORD COPY

Oberson, Greg

From: Dunn, Darrell
Sent: Thursday, July 12, 2012 12:59 PM
To: Rubenstone, James; Compton, Keith; Einziger, Robert; Ahn, Tae; DePaula, Sara; Gordon, Matthew
Cc: Oberson, Greg; Gavrilas, Mirela; Csontos, Aladar; Lin, Bruce
Subject: Calvert Cliffs inspection

I participated in the EPRI ESCP NDE subcommittee phone call this morning. During the call John Massari from Constellation Energy provided results from the Calvert Cliffs inspection. He indicated that the Salt Smart results for the cold canister indicated the surface concentration of soluble salts was 543 mg/m2. I asked if this was all soluble salts. The response from Massari and Jim Brown (AREVA-Transnuclear) is that they believe this indicates concentration of soluble chloride salts on the surface. They have not analyzed the sample collected with the scotch brite scouring pad/filter. I am not sure that will provide a confirmation of the surface concentration but it should give information on what chloride species are present.

Please treat these results as preliminary at this point. I am sure we will hear more later.

Darrell Dunn
RES/DE/CMB
Phone: 301-251-7621
Fax: 301-251-7420

E107

Oberson, Greg

From: Dunn, Darrell
Sent: Wednesday, September 19, 2012 9:37 AM
To: Oberson, Greg
Subject: RE: SCC Analog and an Addition

It is not clear to me that any of these references are applicable to plausible range of conditions expected for dry storage casks. The paper by Barber et al. might be relevant but I did not find much on the environment or the source of chlorides in my quick scan of this paper. With so many available references that are actually applicable, it is puzzling why someone would choose references that will only lead to confusion and actually degrade the paper.

I'll bet $20 that the Corrosion 2002 paper was not actually presented at the conference. It is almost impossible for people in Iran to get a visa in time to travel to the US for such purposes. Besides, the paper is terrible. I would have rejected it. The BWR stub tube case really is a stretch.

From: Oberson, Greg
Sent: Wednesday, September 19, 2012 9:04 AM
To: Dunn, Darrell
Subject: FW: SCC Analog and an Addition

From: Ahn, Tae
Sent: Monday, September 17, 2012 1:46 PM
To: DePaula, Sara; Oberson, Greg
Subject: SCC Analog and an Addition

Randy said the concrete could retain water. He think that a max of 30 g/m^3 could be an underestimate.

E108

Oberson, Greg

From: He, Xihua <xihua.he@swri.org>
Sent: Thursday, November 15, 2012 9:39 AM
To: Oberson, Greg
Subject: SCC project weekly update

Greg,

For Task 5, we have bent the samples to 1.5% strain and placed them in the chamber yesterday. We will start the deposition today and the specimens may be ready tomorrow. Other on-going tests are running smoothly.

For Task 3, we looked at the last two samples pulled from the 60 °C and 30%RH test. We didn't see any more cracking and only pits. We may need to re-examine the 60 °C-25% RH test specimens to confirm what we have seen.

My schedule is open today, please call if you want.

Thanks,
Xihua

E109

Oberson, Greg

From: He, Xihua <xihua.he@swri.org>
Sent: Wednesday, December 12, 2012 1:44 PM
To: Oberson, Greg
Subject: pulled specimen
Attachments: 304S-20C #2.jpg

Greg,

Just a quick update that we pulled one as-received and one sensitized C-ring specimens from the 52 °C-32% RH test at yield strength with 10 g/m2 salt. Crack was clearly observed from the sensitized specimen (see one photo attached), but not from the as-received specimen.

Thanks,
Xihua

E110

Oberson, Greg

From: He, Xihua <xihua.he@swri.org>
Sent: Friday, December 14, 2012 6:15 PM
To: Oberson, Greg
Subject: FW: SCC Task 1 Specimens Pulled

Greg,

Probably Todd mentioned to you that we stopped Task 1 tests. The following are some preliminary results.

- 60 °C samples with 10 g/m^2 exposed for 6.5 months (6 as received and 6 sensitized samples) –Looks like cracking may be present on at least one sensitized sample.
- 35 °C samples with 0.1 g/m^2 exposed for 1-year (3 as received, 3 sensitized, and 2 welded samples) – Cracking looks like it is present on at least one sensitized sample.
- 45 °C samples with 0.1 g/m^2 exposed for 1-year (3 as received, 3 sensitized, and 2 welded samples) – Cracking looks like it is present on at least one sensitized sample.

We will continue to examine the specimens next week and keep you informed on the progress.

Thanks and have a good weekend,
Xihua

E111

Oberson, Greg

From: He, Xihua <xihua.he@swri.org>
Sent: Wednesday, January 02, 2013 6:15 PM
To: Oberson, Greg
Subject: SCC project update

Greg,

Happy New Year to you and your family! Hope you had a good holiday!

The SCC project is moving on smoothly. We terminated the Task 4 test after the longest exposure for 4 months. We observed cracking from the CaCl2 and MgCl2 specimens, not from the NaCl. The sea salt specimens still need to be examined.

We pulled the Task 5 45C 44% RH 1g/m3 samples last Friday (3-Month Exposure). We only saw pits on the surface of the samples. We can try and cut them up.

For Task 1 , we examined the 52C 1g/m2 samples exposed for 8 months. We saw what looked like potential shallow cracks on the sensitized samples, but we would need cross-sectioning to confirm. We only saw pits on the as-received samples.

Chapter 2 for Nureg report is uploaded to SharePoint after Todd and Bobby reviewed it.

Will you please give me a call tomorrow? I would like to discuss with you on the NACE conference.

Thanks,

Xihua

E112

Oberson, Greg

From: He, Xihua <xihua.he@swri.org>
Sent: Tuesday, January 22, 2013 10:36 AM
To: Oberson, Greg
Subject: SCC project update

Greg,

Just a quick update on the SCC project: We pulled all the specimens out yesterday which ends all the SCC tests. We observed cracking from the C-ring specimens with 1.5% strain exposed to 52 °C-32% RH, but not from the 45 °C-44% RH test. Some specimens are still under examination. I will send you some photos tomorrow. Please let me know if there are any questions.

Thanks,
Xihua

E113

7/26/12

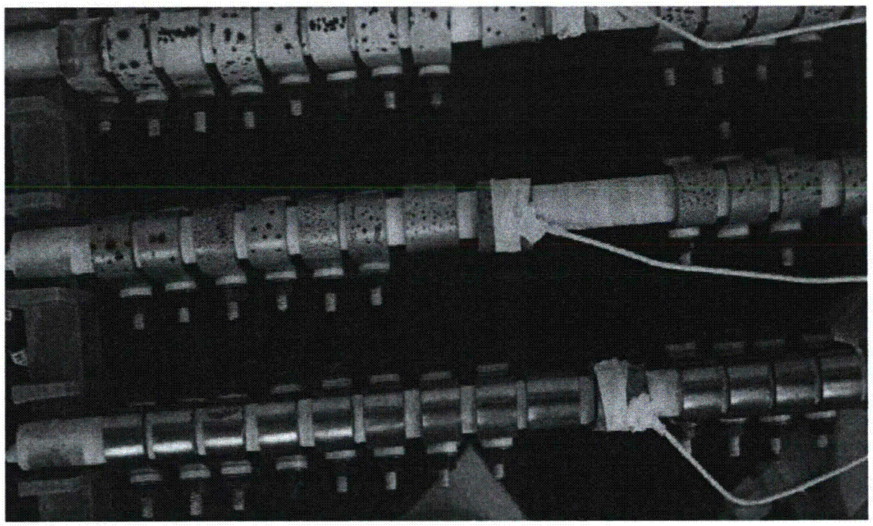
May 21, 2012 SCC Project Update

E114

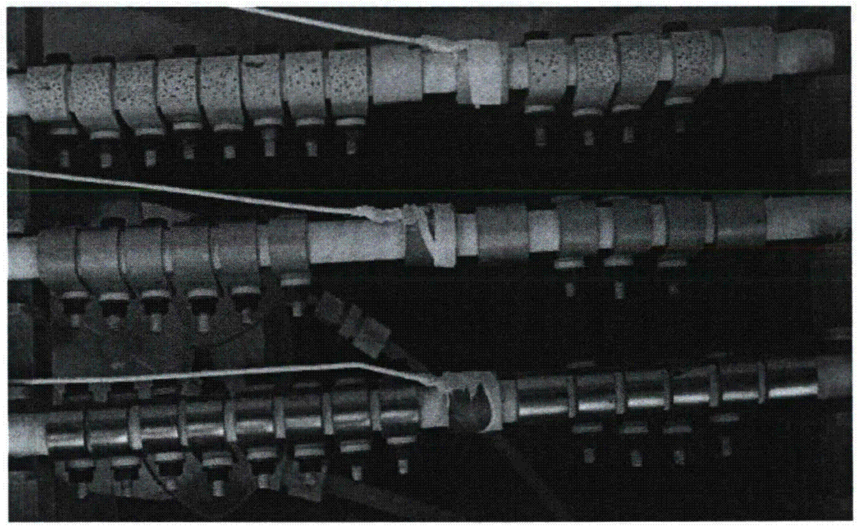
Task 1 — 4-month Pull of 72 Specimens

- Removed all the remaining 35 and 45 °C specimens deposited with 1 and 10 g/m^2 simulated sea salt on the surface and half of the remaining specimens with 0.1 g/m^2 salt
- Salt remained on the removed 0.1g/m^2 specimens surface
- Pitting was observed from all the specimens. Cracks were observed visually from the surface of some specimens, but there were no through wall cracks on any specimen.

35 °C



45 °C



Summary of Specimens Exposed for 4 Months at 35 and 45 °C at Absolute Humidity Below 30 g/m^3

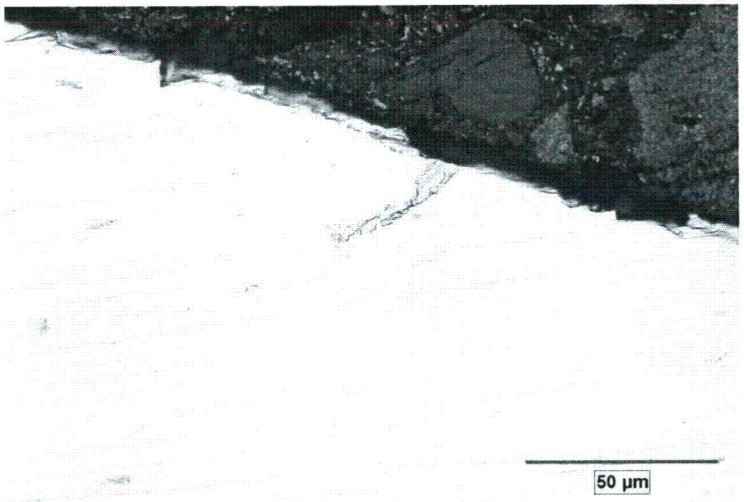
Temperature (°C)	Salt concentration (g/m^2)	Fabrication	Surface examination		Cross section examination for cracking
			Pitting	Cracking*	
35	0.1	As-received	3/3	0/3	0/2
		Sensitized	3/3	2/3	0/1
		Welded	2/2	1/2	0/1
	1	As-received	6/6	5/6	N/A
		Sensitized	6/6	4/6	N/A
	10	As-received	6/6	3/6	N/A
		Sensitized	6/6	6/6	N/A
		Welded	4/4	1/4	2/3
45	0.1	As-received	3/3	1/3	0/2
		Sensitized	3/3	1/3	2/2
		Welded	2/2	0/2	1/2
	1	As-received	6/6	3/6	N/A
		Sensitized	6/6	2/6	N/A
	10	As-received	6/6	6/6	N/A
		Sensitized	6/6	6/6	N/A
		Welded	4/4	4/4†	1/1

*Cracks were observed without additional cleaning to remove the corrosion products or cross sectioning

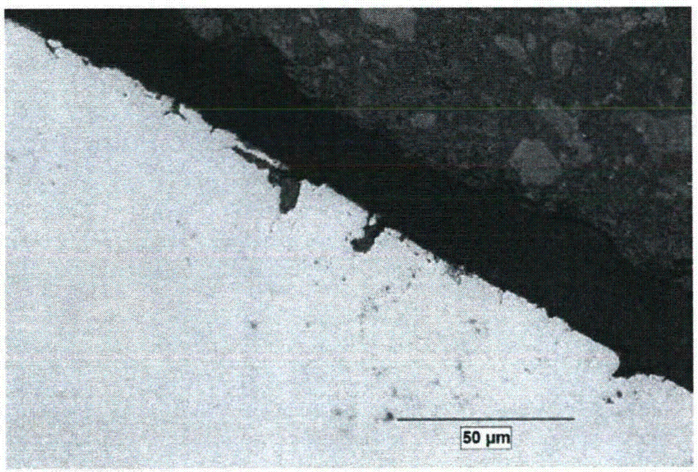
†Cracks observed on welded specimens were in the base material

4-month Pull — 0.1 g/m^2 salt, 35 °C

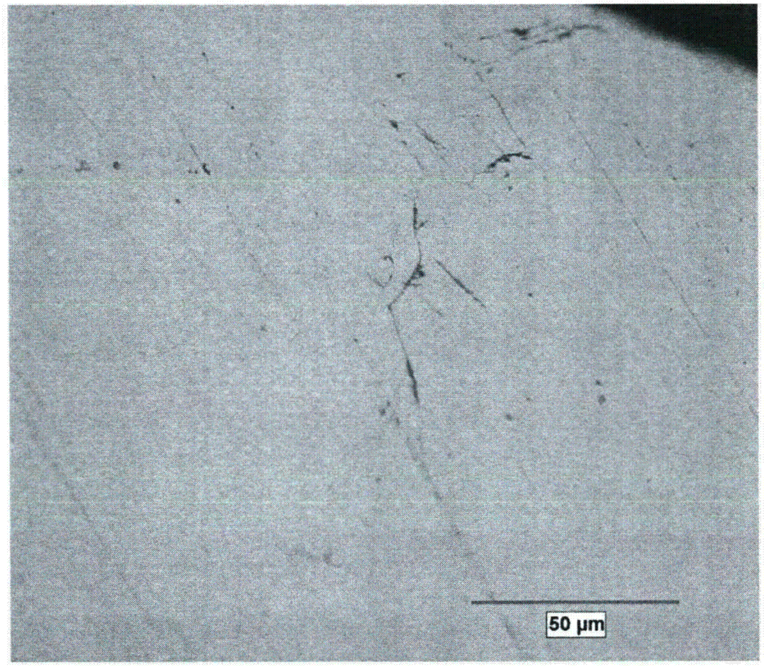
As-received: no cracking found



Welded: no cracking found



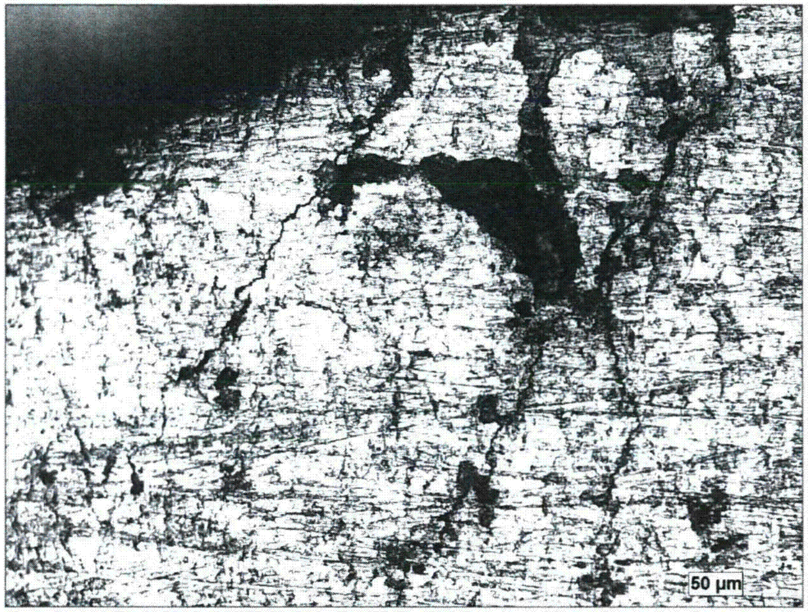
Sensitized: cracks observed



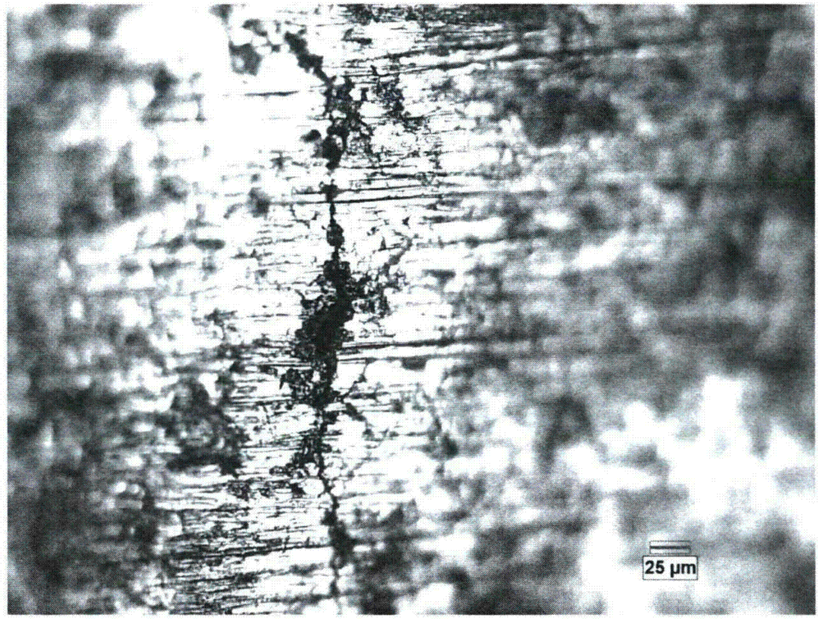
- Cracking observed on sensitized specimen, not from cross section of as-received and welded

4-month Pull — 1 g/m^2 salt, 35 °C

As-received



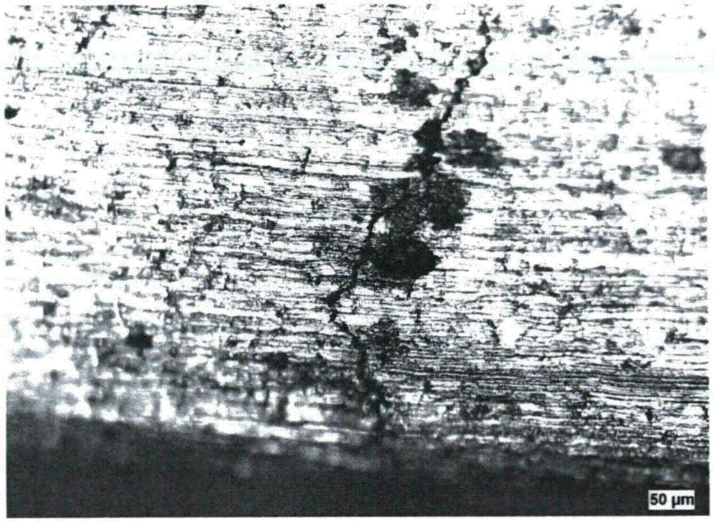
Sensitized



- Cracks observed

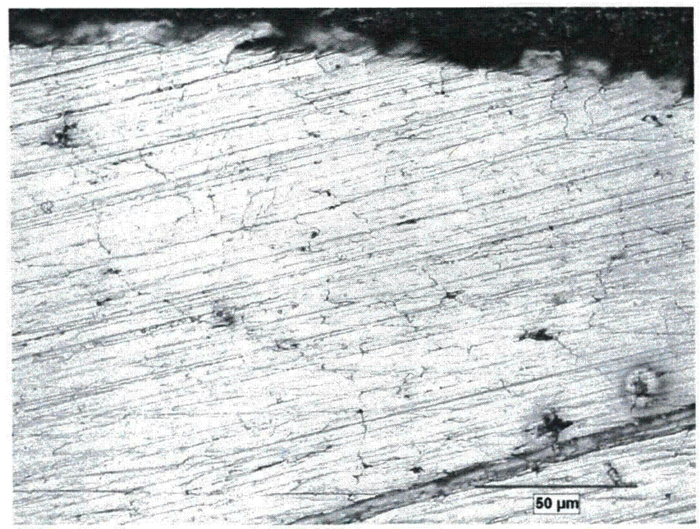
4-month Pull — 10 g/m^2 salt, 35 °C

As-received



Sensitized

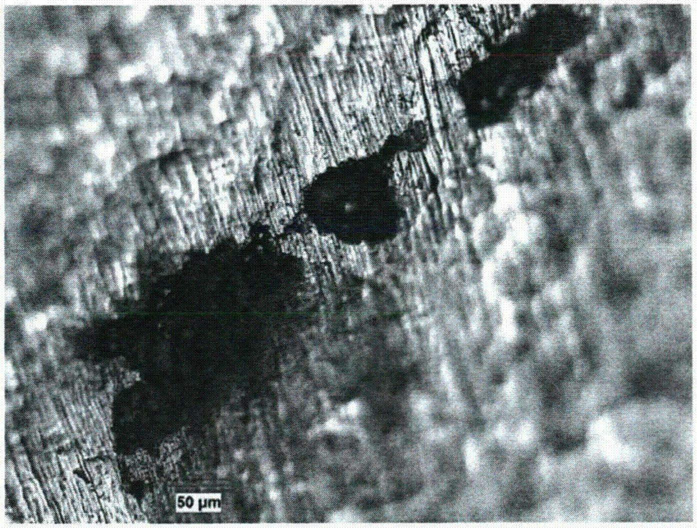
Welded



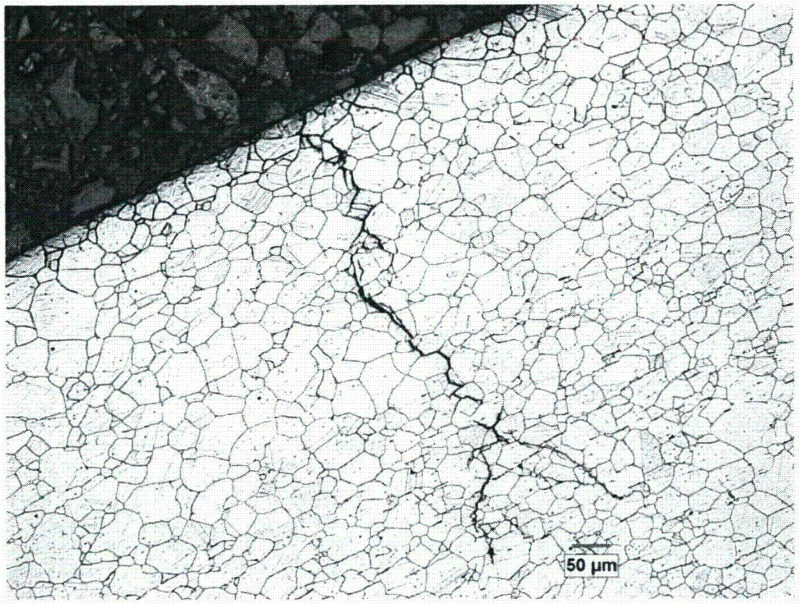
- Cracks from as-received and sensitized
- Interdendritic attack in the weld

4-month Pull — 0.1 g/m^2 salt, 45 °C

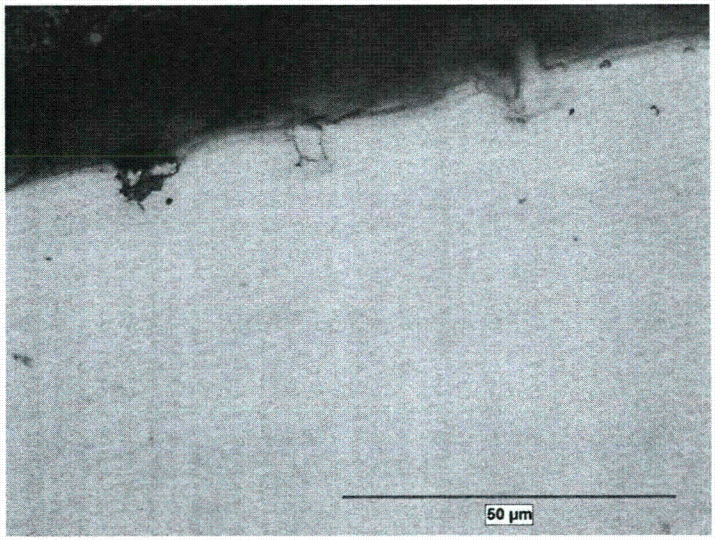
As-received



Sensitized (etched)



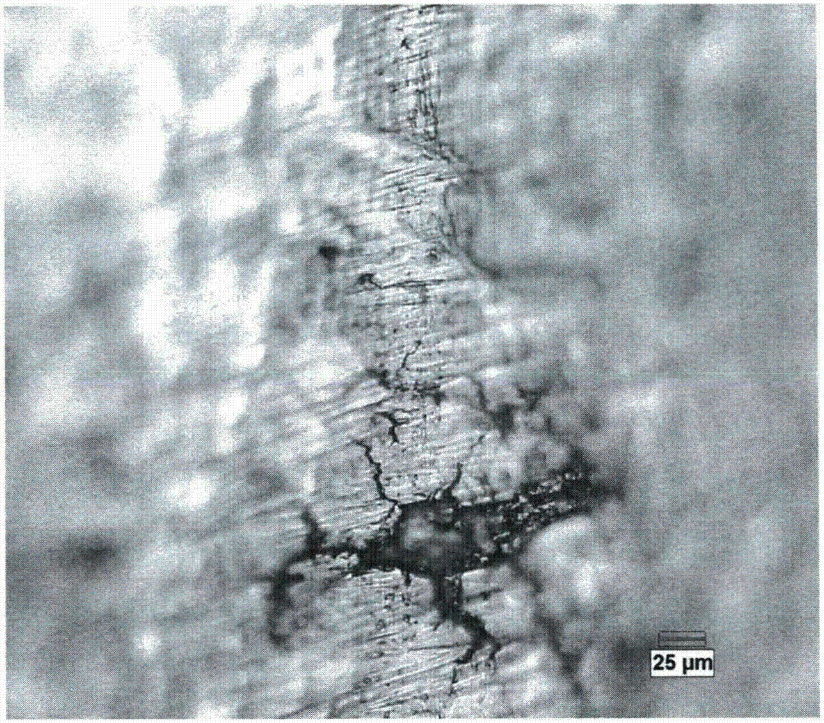
Welded, 0.1 g/m^2 salt, 45 °C



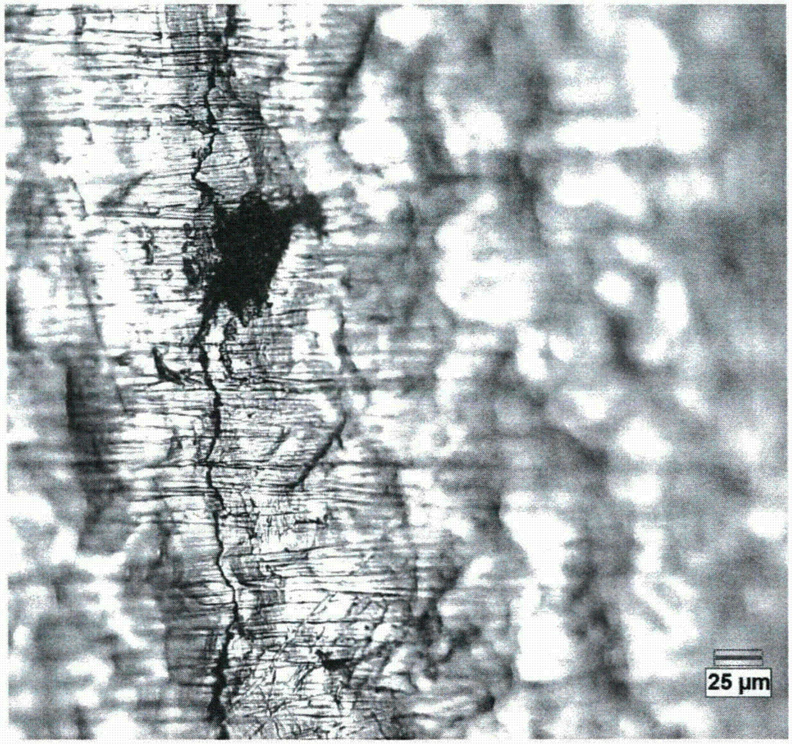
- Cracks from as-received, sensitized, and welded

4-month Pull — 1 g/m^2 salt, 45 °C

As-received



Sensitized



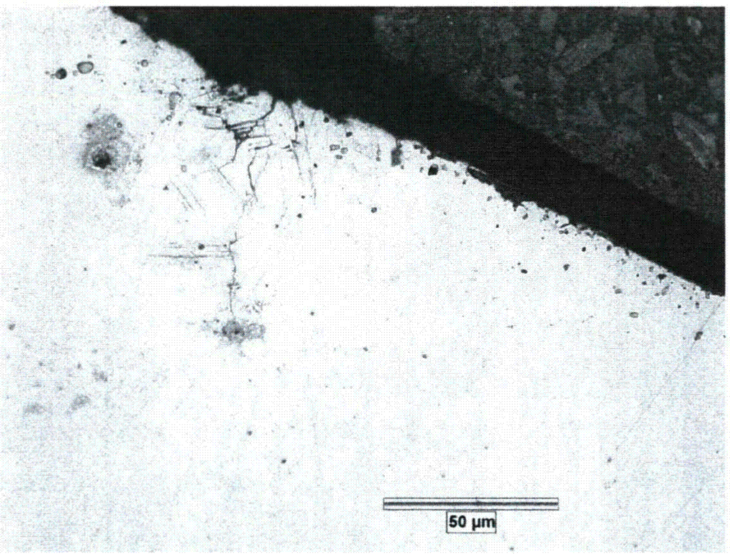
- Cracks observed

4-month Pull — 10 g/m^2 salt, 45 °C

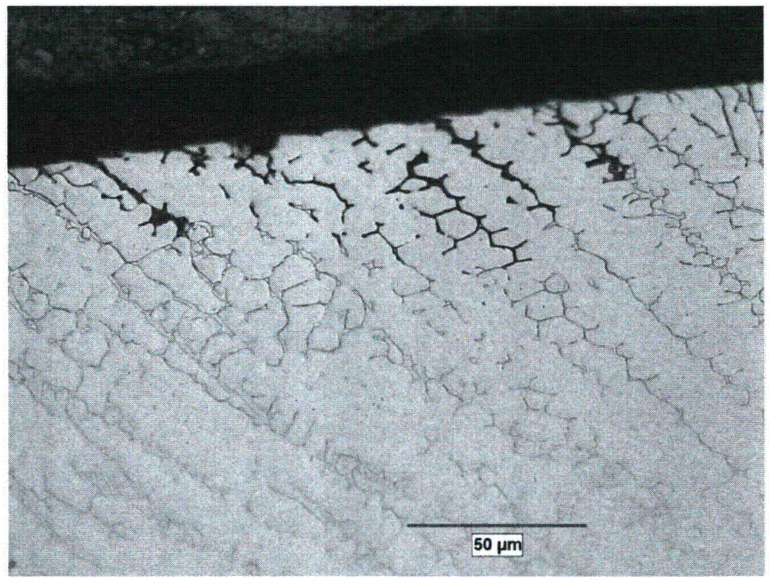
As-received, 10 g/m^2 salt, 45 °C (no photo available)

Sensitized, 10 g/m^2 salt, 45 °C (no photo available)

Welded, 10 g/m^2 salt, 45 °C

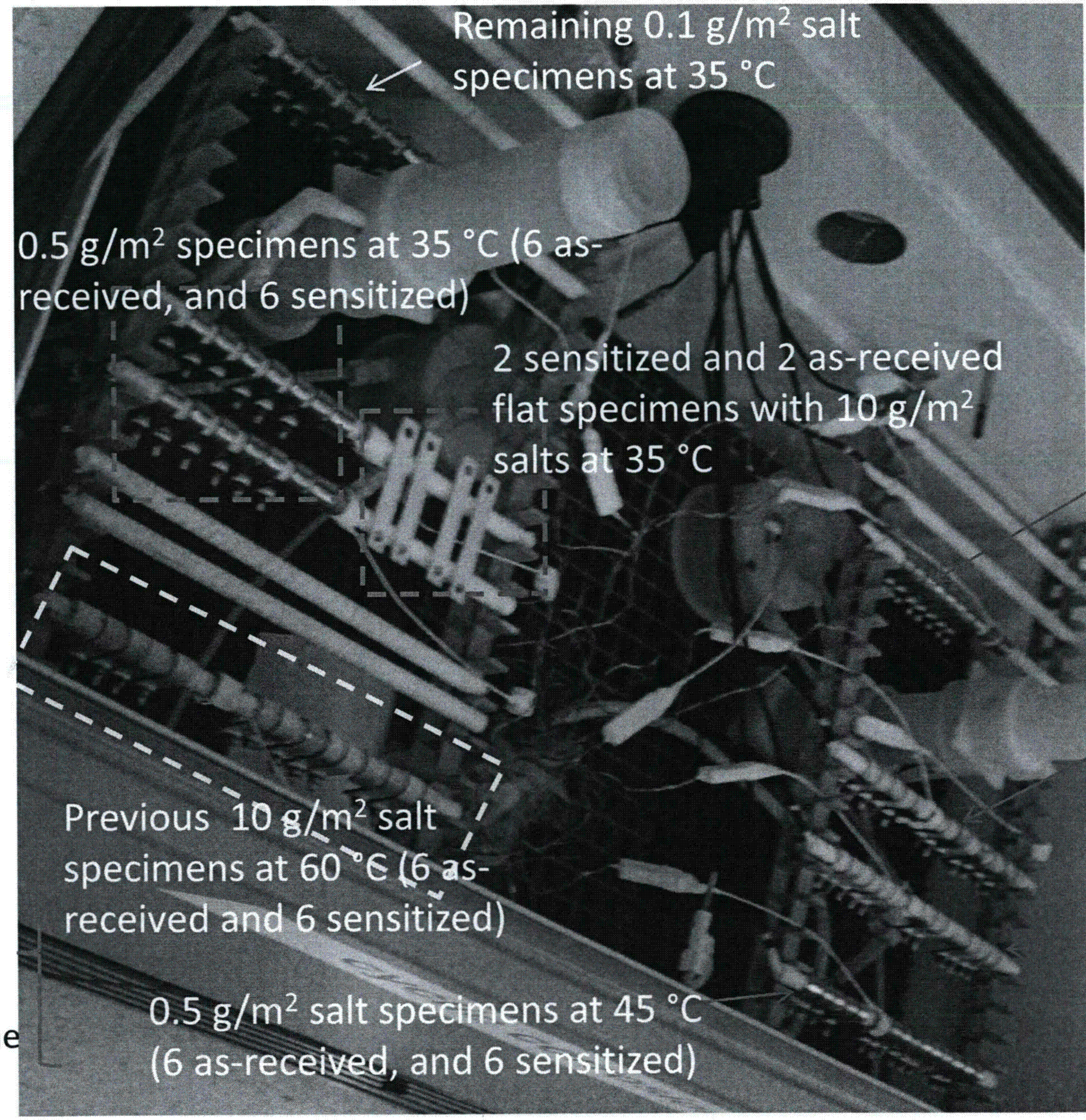


etched



- Cracks observed from as-received and sensitized and cracks from welded

Task 1 Additional Tests



Remaining 0.1 g/m^2 salt specimens at 45 °C

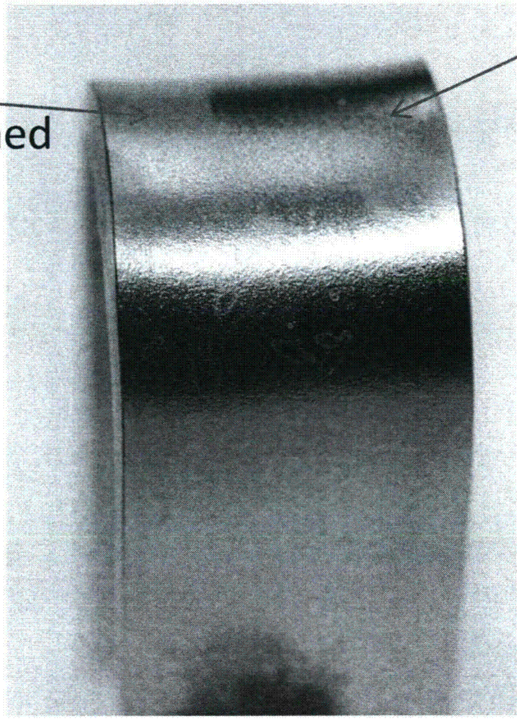
10 g/m^2 salt specimens at chamber temperature (6 as-received, and 6 sensitized)

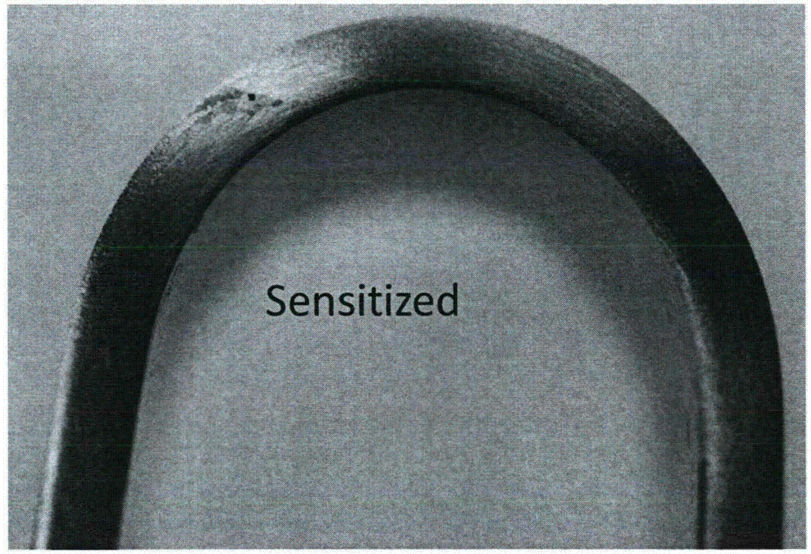
10 g/m^2 salt specimens at 52 °C (6 as-received, and 6 sensitized)

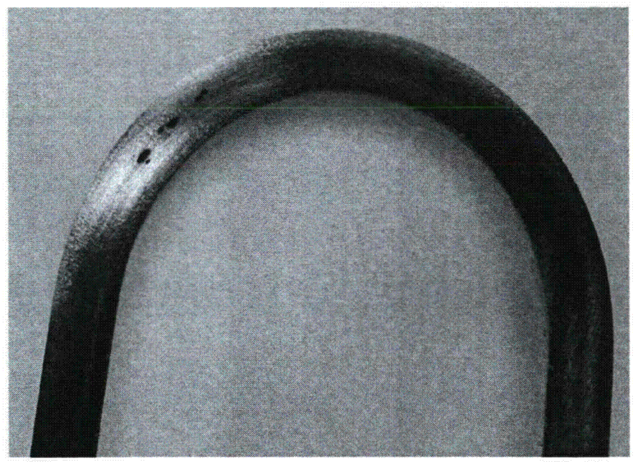
Note: Affected by dripping, moved to the center later

10

1-month Pull of Additional Tests—35 °C, 0.5 g/m^2 salt



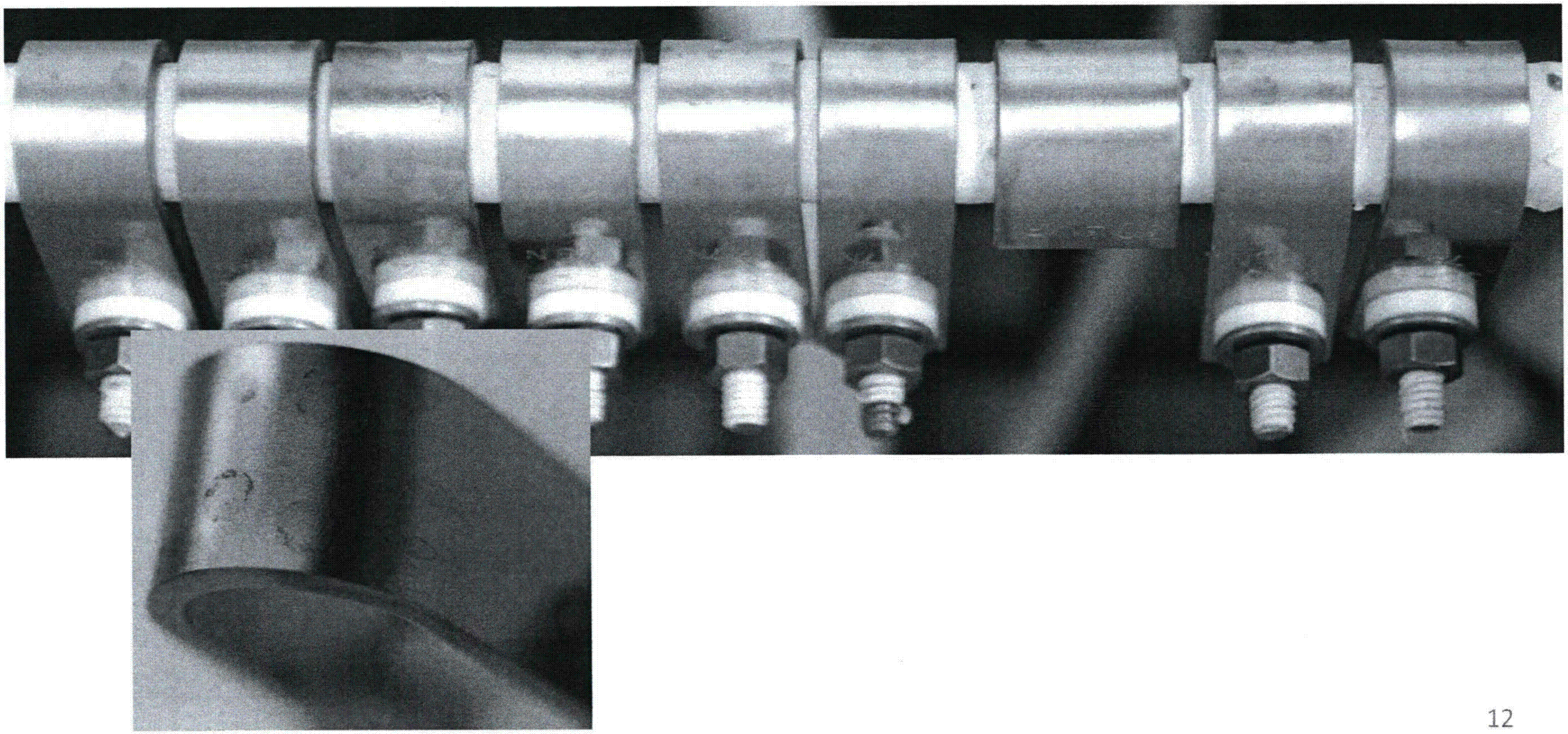




- Salt remained on 0.5 g/m^2 specimens
- Pits observed at sides. Minor pits at top surface. More pits observed for sensitized specimens
- No cracking observed from surface. Plan to cross section to observe

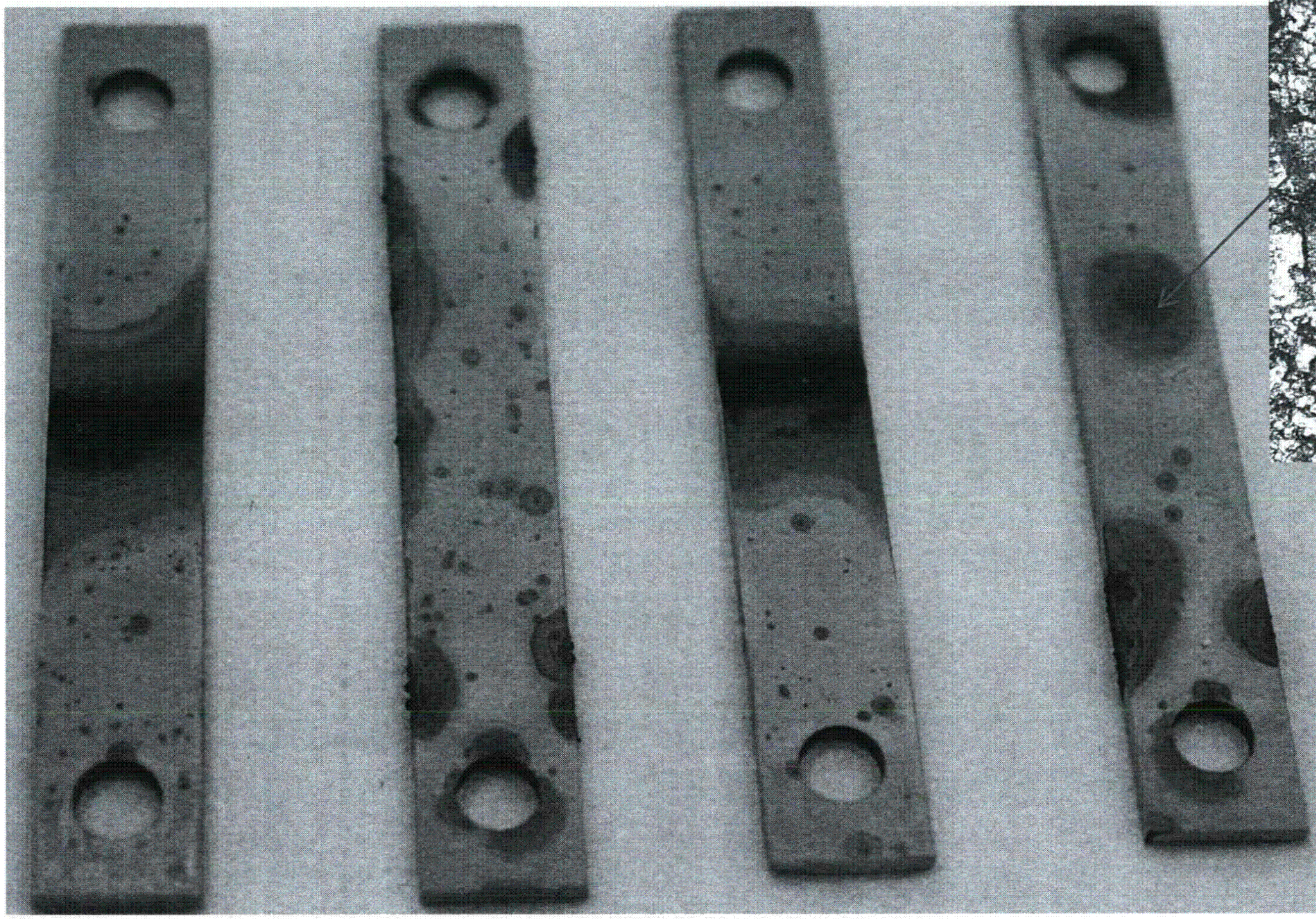
1-month Pull of Additional Tests—45 °C, 0.5 g/m² salt

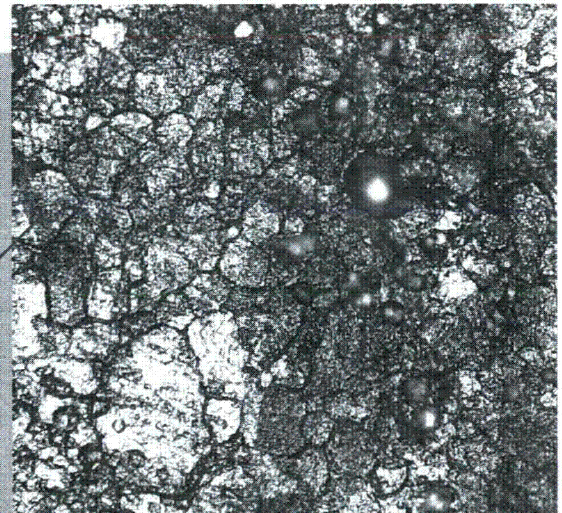
- Specimens seem to be dripped by condensation from the chamber and dripping may have induced minor pitting
- Results may be compromised by dripping, but the overall outcome for this project is not compromised because cracking was observed at 0.1 g/m² salt specimens



1-month Pull of Additional Tests — Flat Specimens, 35 °C, 10 g/m² salt

welded Sensitized welded Sensitized



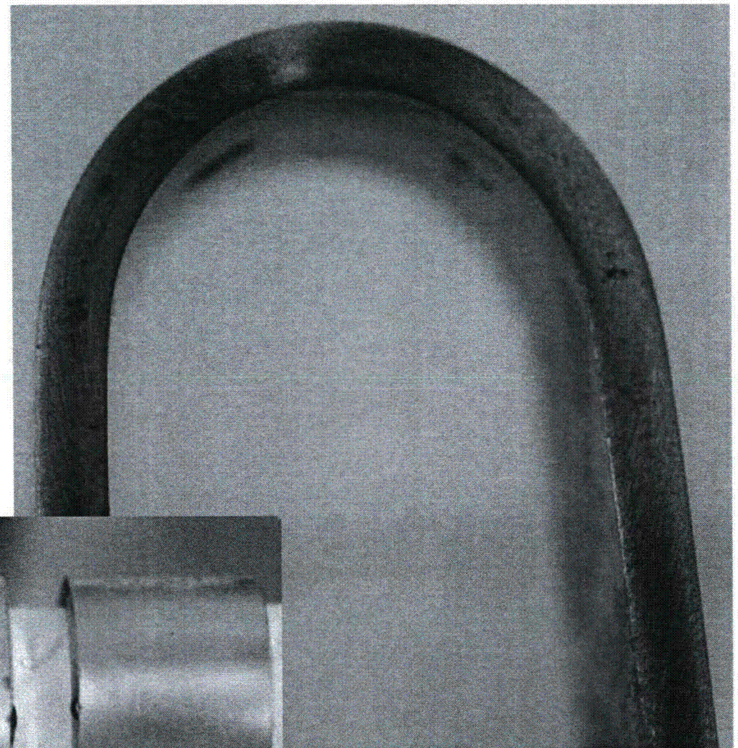


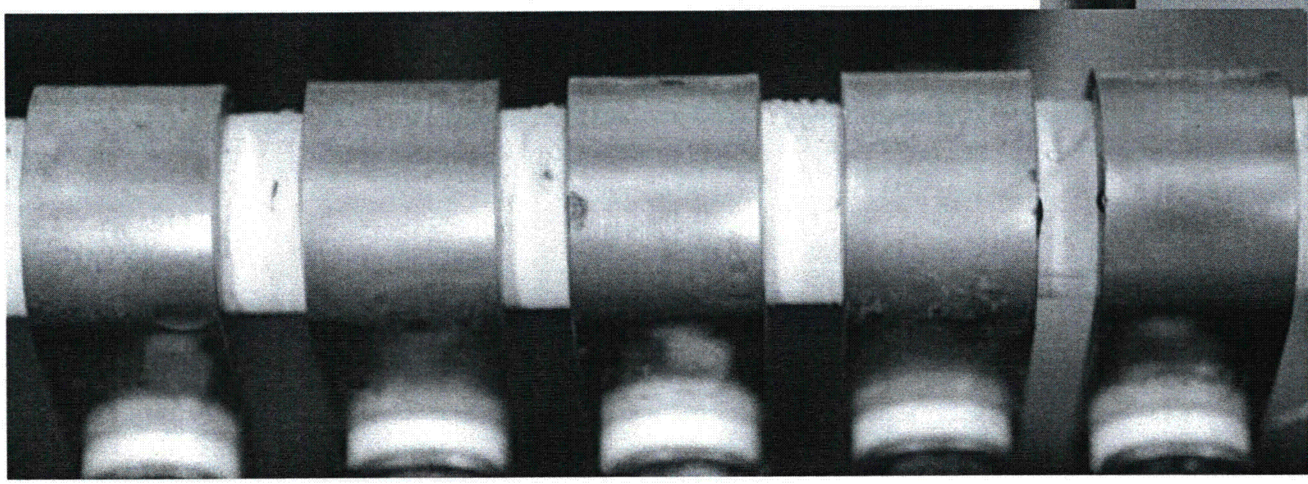
Intergranular etching

- Extensive pitting, no cracking from surface
- Further action: cross section welded region and pits to examine

1-month Pull of Additional Tests — Chamber Temperature (~27 °C), 10 g/m^2 salt

- Salt deliquesced (RH>75%) and partially drained
- Pitting observed at sides and top, no cracking observed
- More pitting on sensitized than as-received
- Further action: cross section to examine further

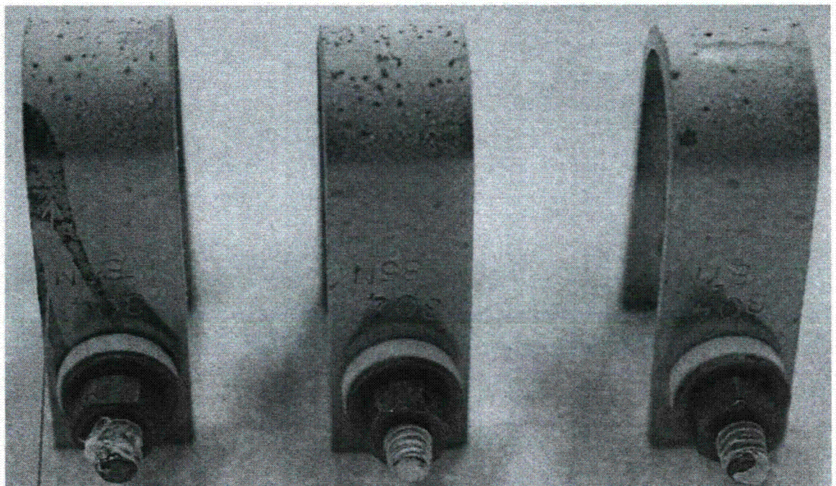




1-month Pull of Additional Tests — 52 °C, 10 g/m^2 salt

- Pitting observed on all specimens
- Cracking observed from sensitized specimen surface

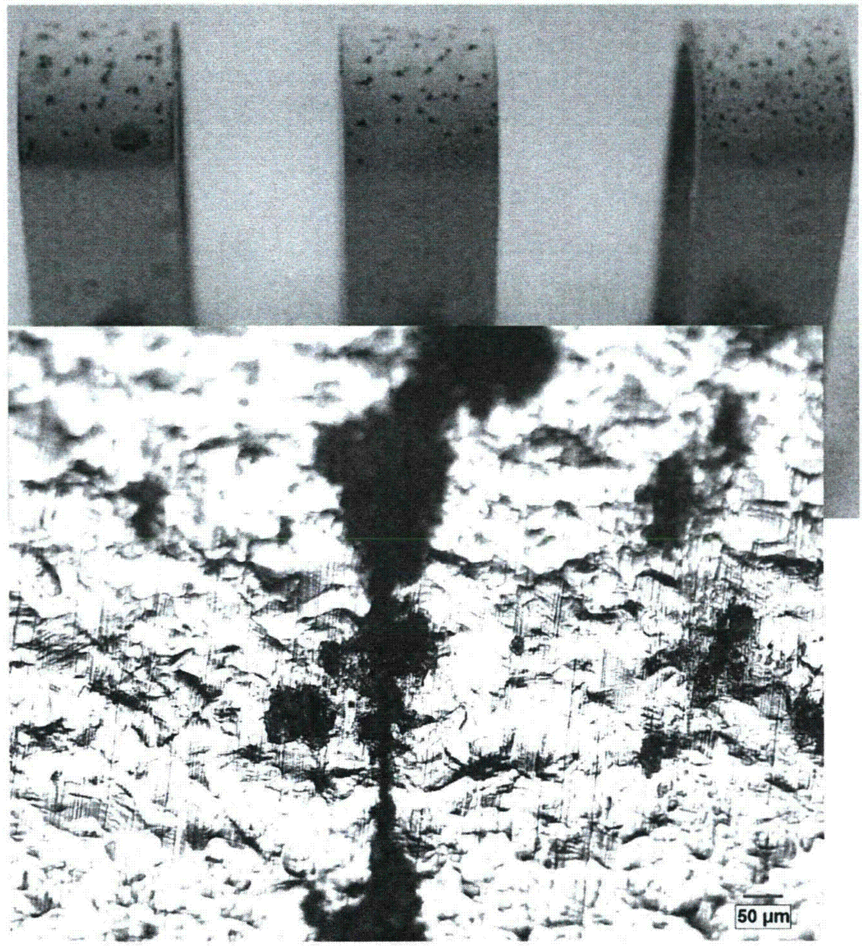
As-received



This specimen was dripped, but cracking was not observed from surface.

- Further action: Cross section some specimens to examine further

Sensitized

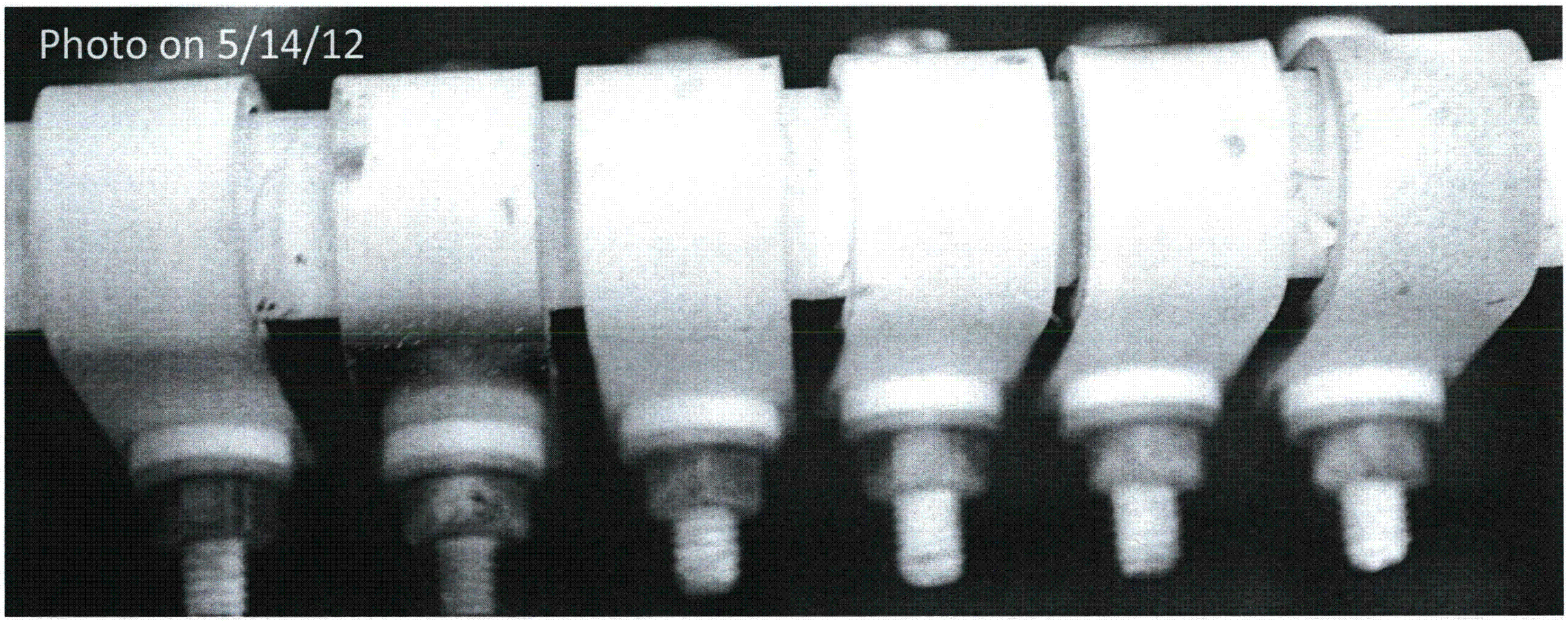


Summary of Specimens Exposed for 1 Month

Temperature (°C)	Salt concentration (g/m^2)	Fabrication	Surface examination		Cross section examination of specimens if cracking not observed from surface
			Pitting	Cracking	
35	0.5	As-received	2/3	0/3	Planned
		Sensitized	3/3	0/3	Planned
45	0.5	As-received	Data compromised, no further action		
		Sensitized			
35	10	Sensitized flat	2/2	0/2	Planned
		Welded flat	2/2	0/2	Planned
Chamber temperature (~27 °C)	10	As-received	2/3	0/3	Planned
		Sensitized	3/3	0/3	Planned
52	10	As-received	3/3	0/3	Planned
		Sensitized	3/3	1/3	Planned

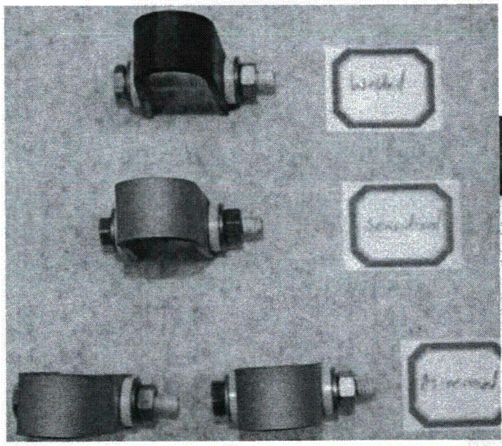
Path Forward for Task 1

- Continue to examine the 4-month pull specimens focusing on cross section of as-received and welded 0.1 g/m^2 specimens
- Cross section the new 1-month pull specimens to examine further
- Pull the 60 °C-10 g/m^2 salt specimens on 5/25/12 if pitting observed (pitting not observed yet).



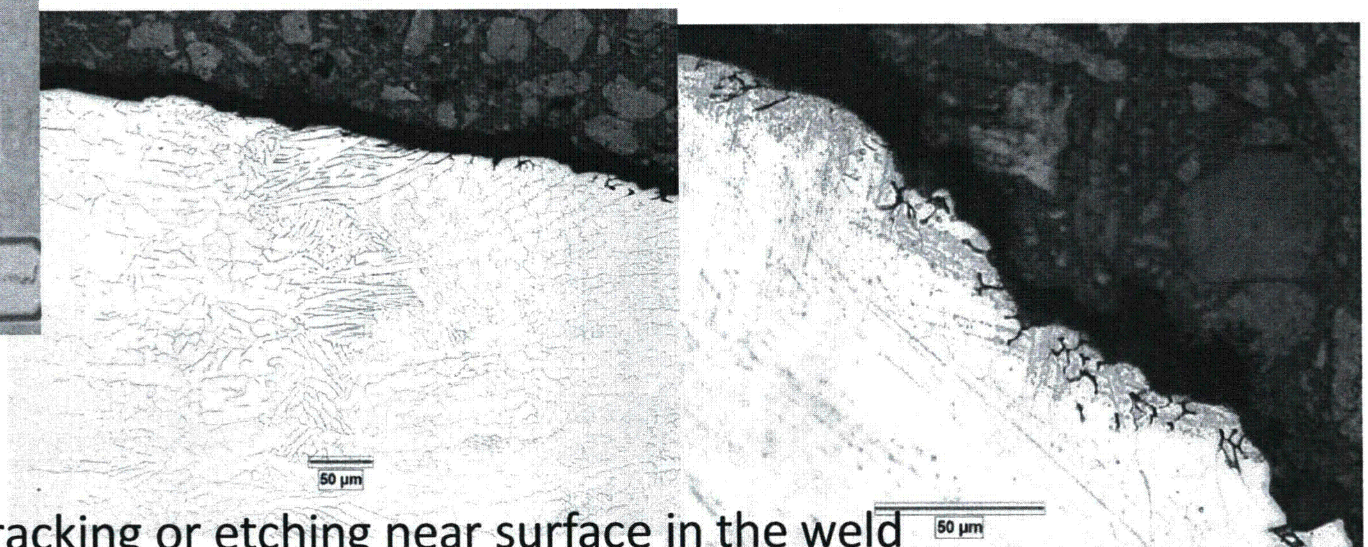
Task 2 SCC Tests at 45 °C-44% RH and 35 °C-72% RH

- SCC tests at 35 °C and 72% RH were terminated in 1 month after exposure at 45 °C-44% RH for 6 weeks
- Except for extensive non-uniform general corrosion observed from specimens exposed to NH_4HSO_4 and very minor corrosion on some specimens, other specimens remained pristine. The specimens exposed to flyash did not show any corrosion and flyash remained dry.



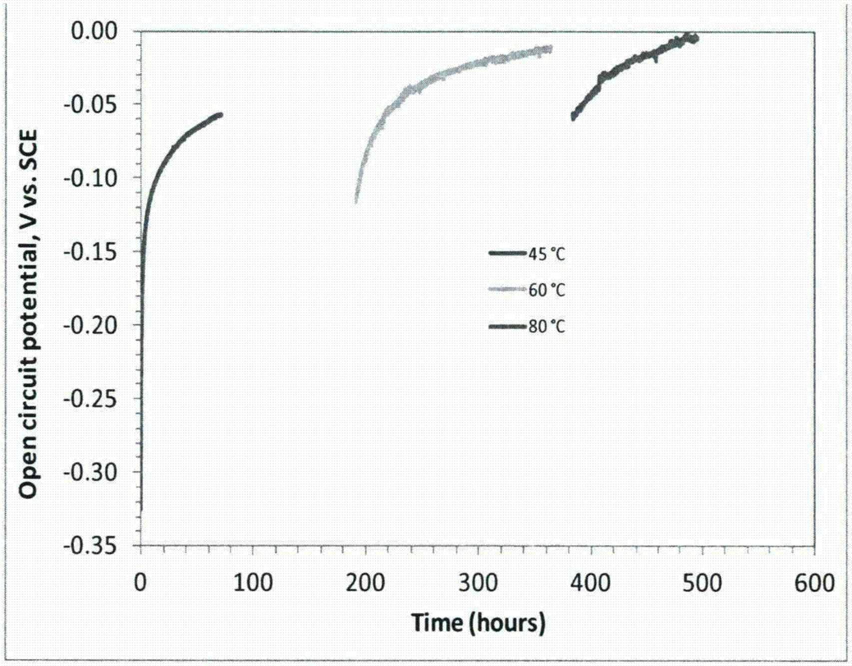
NH_4HSO_4 exposure

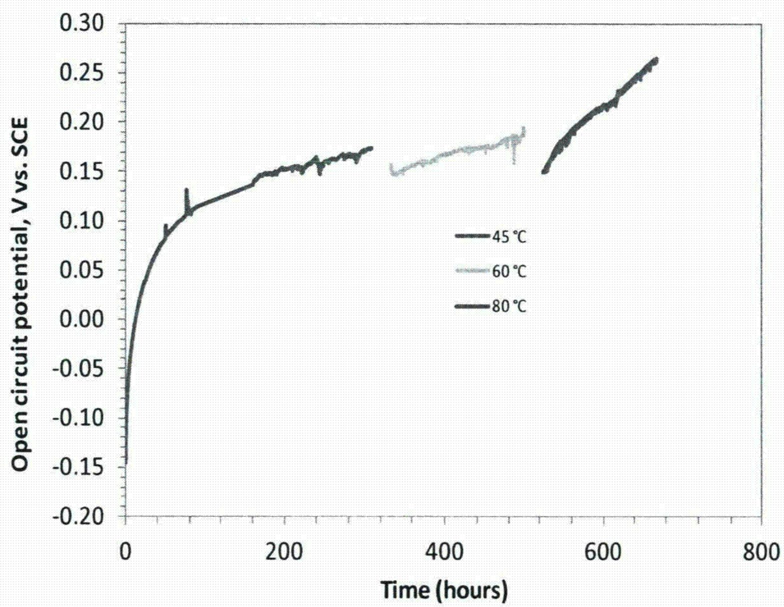
Welded in the welding zone (etched)



- Intergranular cracking or etching near surface in the weld
- Further action: EDS line scan to examine element distribution for possible dealloying

Open circuit potential measurement in (a) $(NH_4)_2SO_4$ and (b) NH_4NO_3 solutions without deaerating at 45, 60, and 80 °C

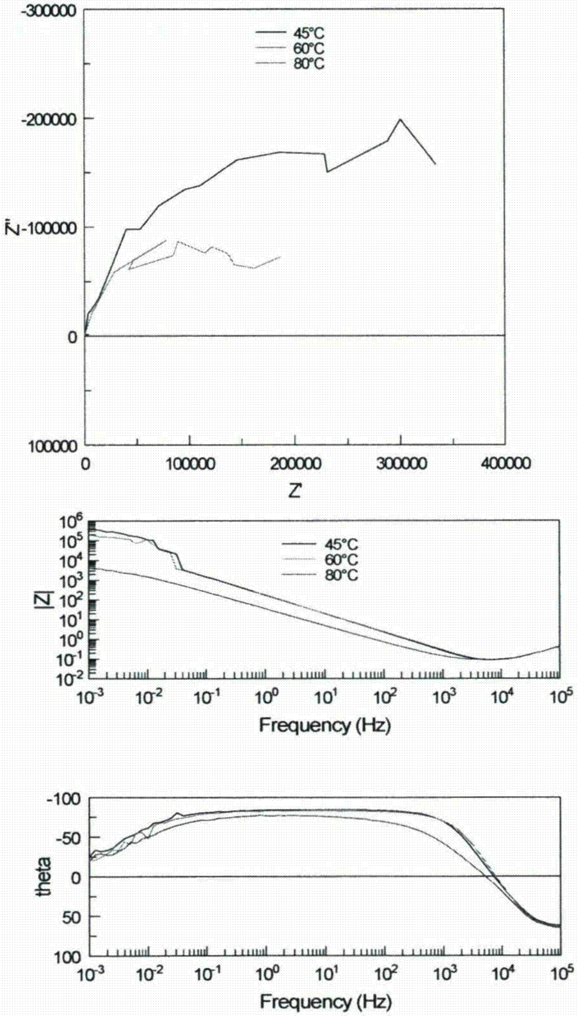




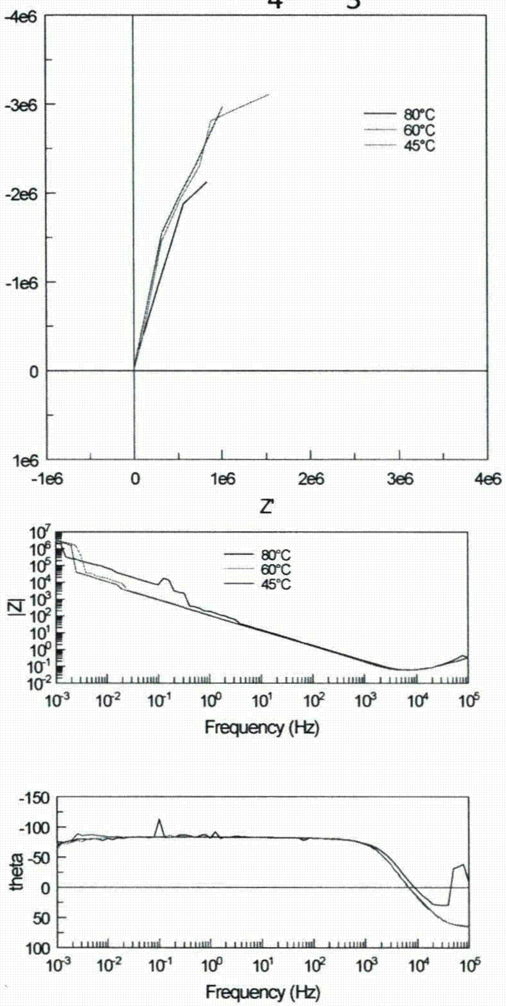
- After resetting to each higher temperature, the open circuit potential decreased momentarily, but it shifted to more noble potential indicating passivation of the oxide film. Overall, no oxide film breakdown events were observed in both solutions over the 1-month monitoring period

Electrochemical impedance spectroscopy of 304 stainless steel at 45, 60, and 80 °C

$(NH_4)_2SO_4$



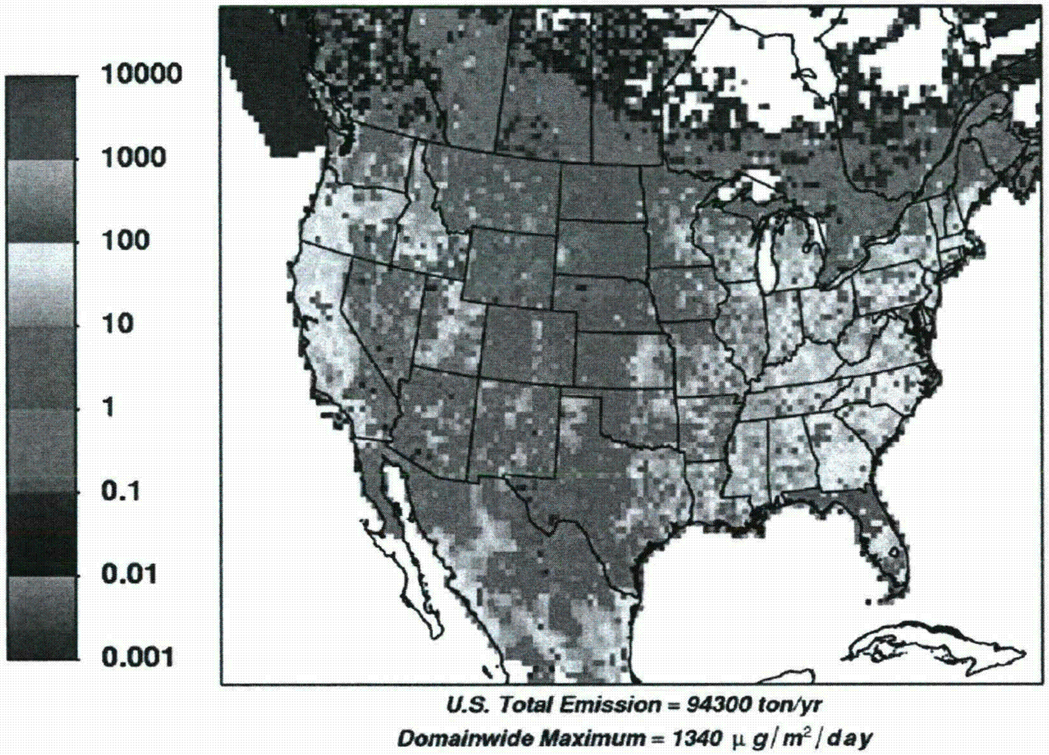
NH_4NO_3



- In both solutions, the oxide film resistance remained high indicated by the high impedance and near constant phase angle in the frequency range of 10^3 to 10^{-1} Hz

Technical Basis for Task 2 SCC Tests With Addition of Chloride

- Spatial distribution of Cl in fine particulate matter ($PM_{2.5}$) emissions derived from the 2001 U.S. EPA National Emissions Inventory (Reff, et al., 2009)



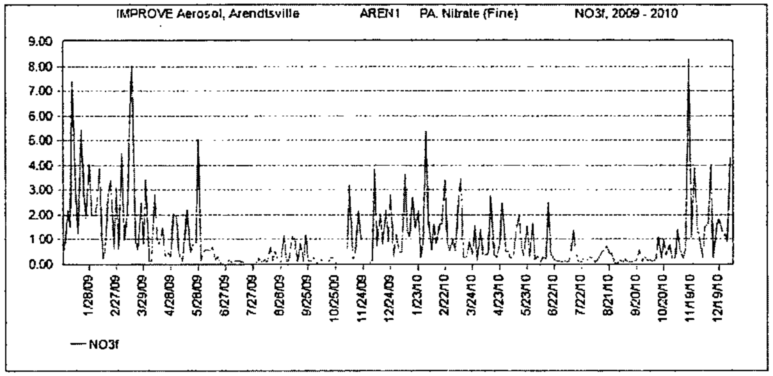
(a) Cl

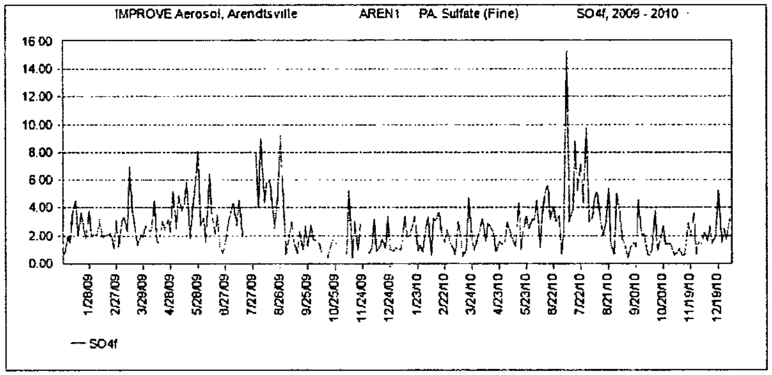
Technical Basis for Task 2 SCC Tests With Addition of Chloride

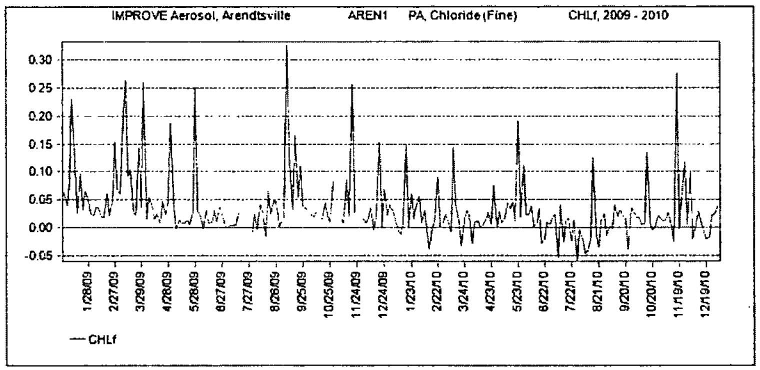
- Location of Monitoring Sites in the IMPROVE network. Red Arrows Show the Sites Selected for Task 2 Experimental Design: (i) Arendtsville, Illinois; (ii) Big Bend National Park, Texas; (iii) Bondville, Illinois; and (iv) Great Smoky Mountains National Park, Tennessee.

Plots of the $PM_{2.5}$ nitrate, sulfate, and chloride data for the four sites

- Nitrate, Sulfate, and Chloride Concentration ($\mu g/m^3$) in Fine Particulates Collected at the Arendtsville, Pennsylvania, Monitoring Site During the Period January 1, 2009 to December 31, 2010. The Data Were Taken From the IMPROVE Database Available at http://views.cira.colostate.edu/web/

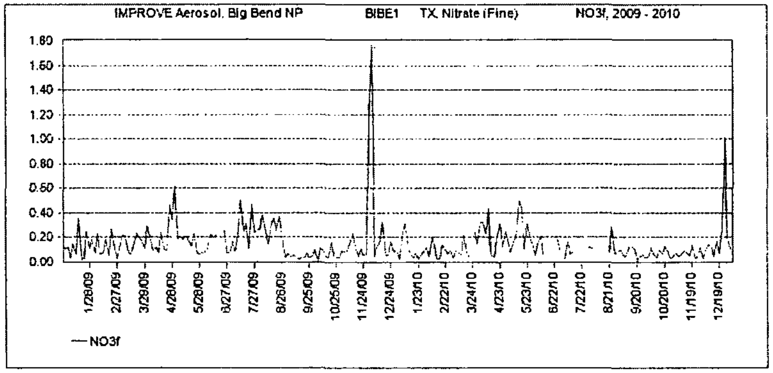


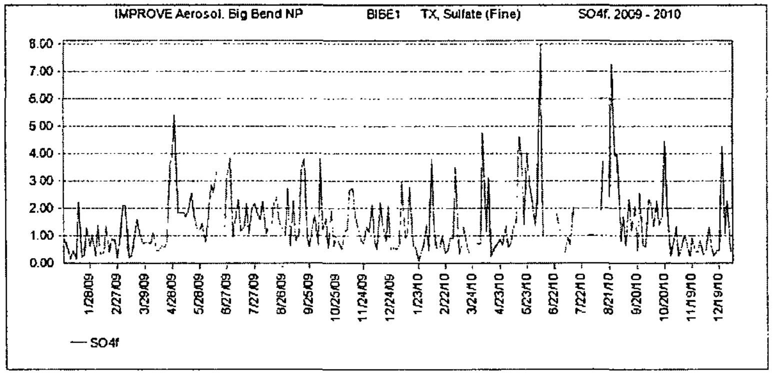


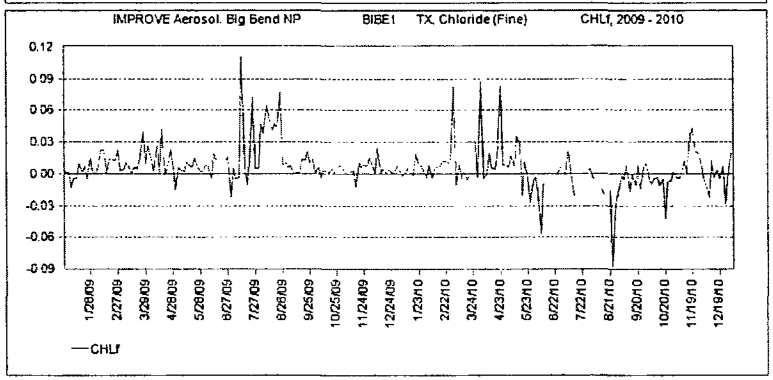


Plots of the $PM_{2.5}$ nitrate, sulfate, and chloride data for the four sites

- Nitrate, Sulfate, and Chloride Concentration ($\mu g/m^3$) in Fine Particulates Collected at the Big Bend National Park, Texas, Monitoring Site During the Period January 1, 2009 to December 31, 2010. The Data Were Taken From the IMPROVE Database Available at http://views.cira.colostate.edu/web/

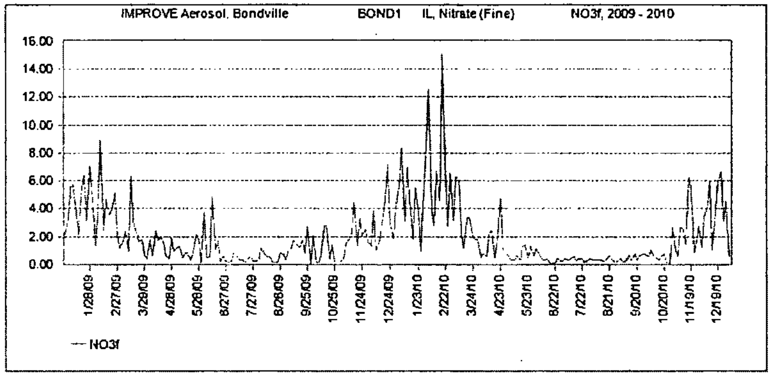


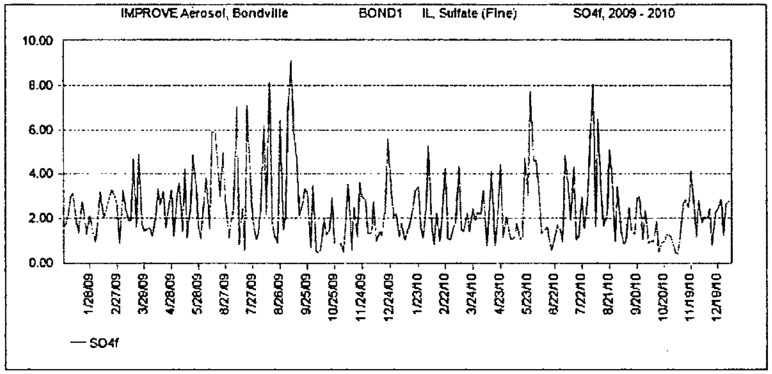


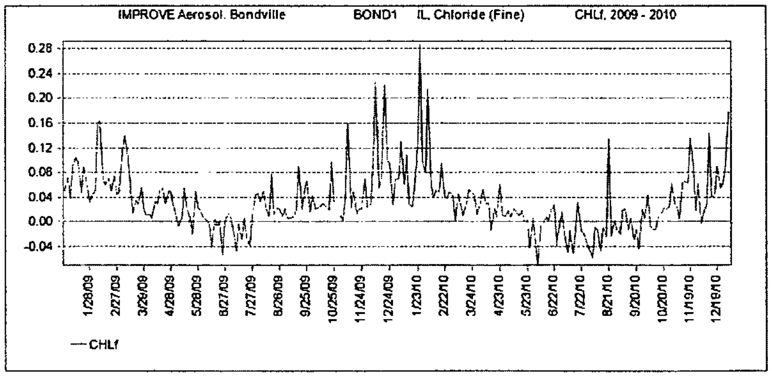


Plots of the $PM_{2.5}$ nitrate, sulfate, and chloride data for the four sites

- Nitrate, Sulfate, and Chloride Concentration ($\mu g/m^3$) in Fine Particulates Collected at the Bondville, Ilinois, Monitoring Site During the Period January 1, 2009 to December 31, 2010. The Data Were Taken From the IMPROVE Database Available at http://views.cira.colostate.edu/web/

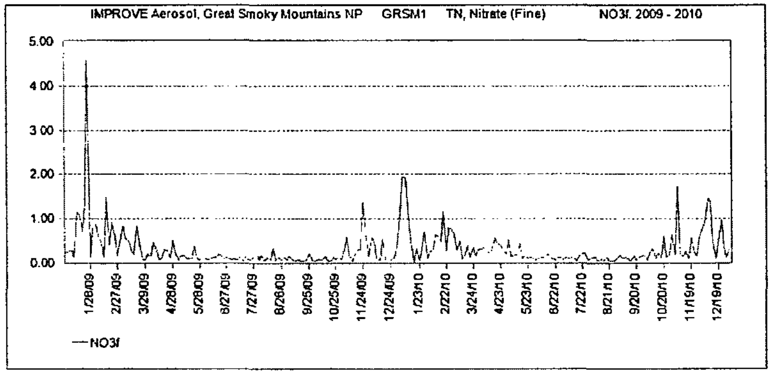


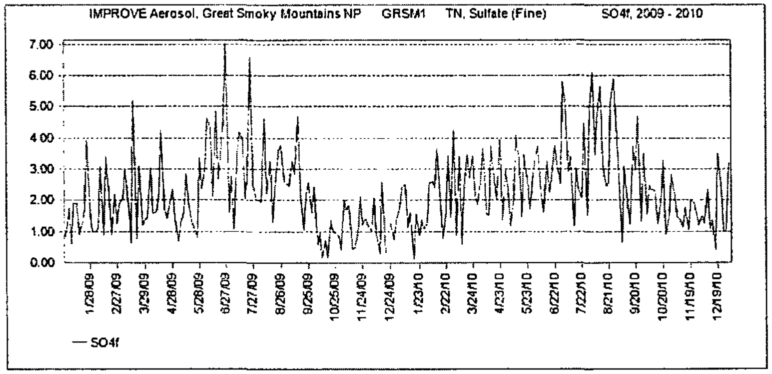


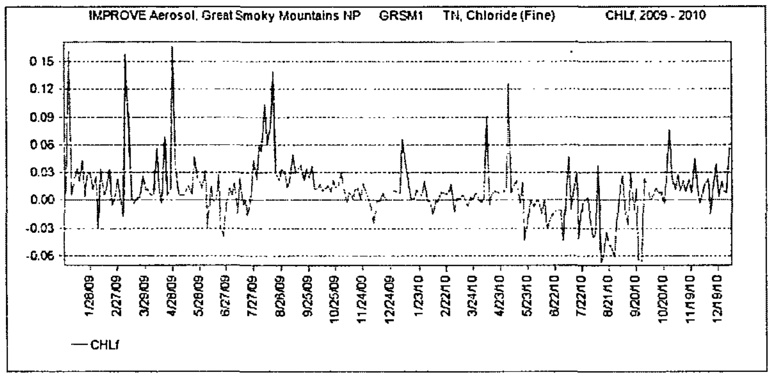


Plots of the $PM_{2.5}$ nitrate, sulfate, and chloride data for the four sites

- Nitrate, Sulfate, and Chloride Concentration ($\mu g/m^3$) in Fine Particulates Collected at the Great Smoky Mountains National Park, Tennessee, Monitoring Site During the Period January 1, 2009 to December 31, 2010. The Data Were Taken From the IMPROVE Database Available at http://views.cira.colostate.edu/web/







Technical Basis for Task 2 SCC Tests With Addition of Chloride

Table 1. Nitrate, Sulfate, and Chloride Concentration in Fine Particulate Matter Collected at Four IMPROVE Monitoring Sites for the Period January 1, 2009 to December 31, 2010.*

Site Location	NO_3^- Concentration Median and Range ($\mu g/m^3$)	SO_4^{2-} Concentration Median and Range ($\mu g/m^3$)	Cl^- Concentration Median and Range ($\mu g/m^3$)
Arendtsville, PA	0.5349 (0.0529 to 8.300)	2.2702 (0.366 - 15.2673)	0.0253 (0.0002 to 0.3252)
Big Bend National Park, TX	0.1068 (0.0084 to 1.7787)	1.1139 (0.098 to 8.0446)	0.0086 (0.0001 to 0.1637)
Bondville, IL	1.1627 (0.0662 to 8.9192)	2.0517 (0.4084 to 9.0997)	0.0315 (0.0006 to 0.2855)
Great Smoky Mountains National Park, TN	0.1482 (0.0382 to 4.5818)	2.0497 (0.1252 to 7.0209)	0.0145 (0.0007 to 0.1657)

* IMPROVE data was accessed using the VIEWS Version 2.0 data query wizard available at http://views.cira.colostate.edu/web/. Negative values in the database were excluded.

Table 2. Mole Ratio of Nitrate to Chloride and Sulfate to Chloride in Fine Particulate Matter Collected at Four IMPROVE Monitoring Sites*

Site Location	NO_3^-/Cl^- Mole Ratio	SO_4^{2-}/Cl^- Mole Ratio	SO_4^{2-}/NO_3^- Mole Ratio
Arendtsville, PA	12.1	33.1	2.7
Big Bend National Park, TX	7.1	47.8	6.7
Bondville, IL	21.1	24.0	1.1
Great Smoky Mountains National Park, TN	5.8	52.2	8.9

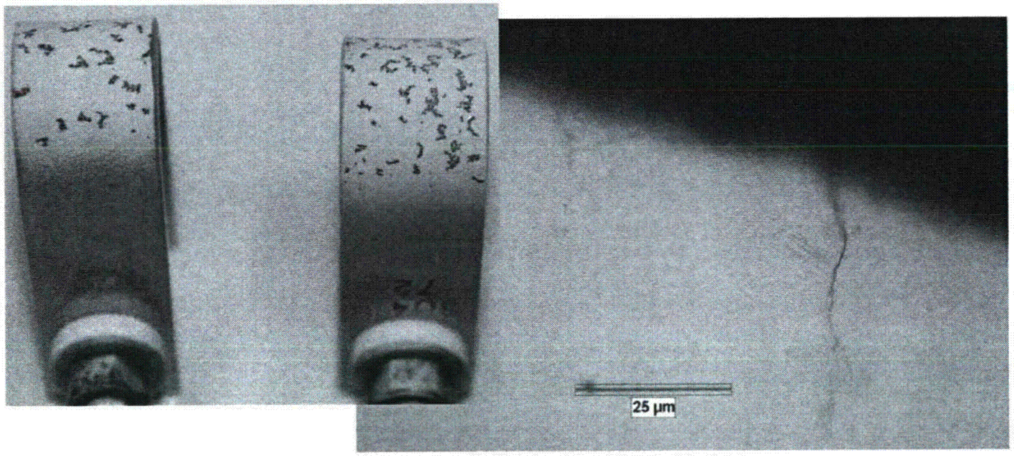
*Based on median NO_3^-, SO_4^{2-}, and Cl^- aerosol concentration listed in Table 1.

Proposed Task 2 Test Matrix with Addition of Chloride

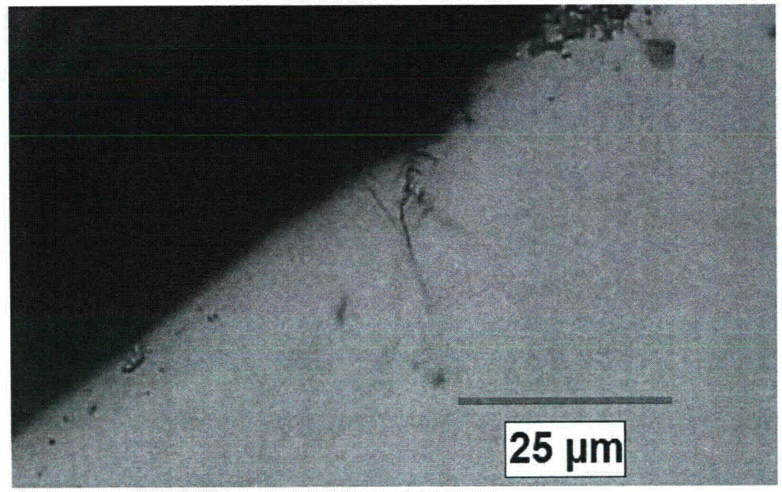
Temperature, °C	Relative Humidity, %	Salts	Specimens	Test duration
45	44	Mole ratio 1. NH_4NO_3/NH_4Cl or NaCl = 6 2. NH_4NO_3/NH_4Cl or NaCl= 6 and $(NH_4)_2SO_4/NH_4NO_3$= 9	3 sensitized and 3 as-received U-bends for each salt	1-2 months

Task 3 — 60 °C-35% RH SCC test (terminated)

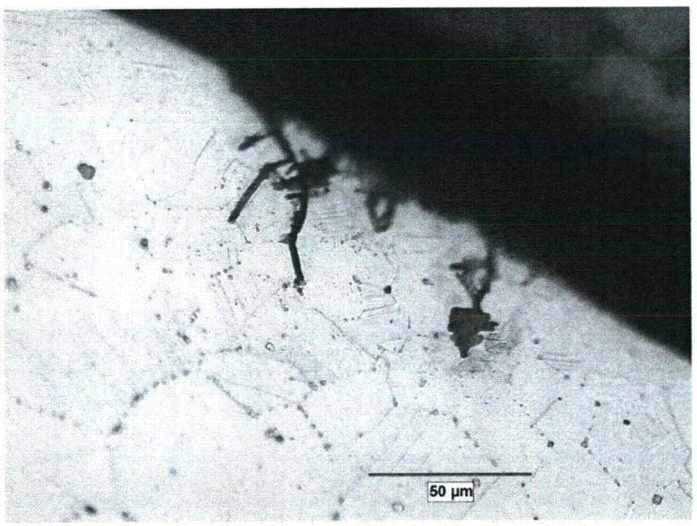
As-received



Sensitized



As-received and etched



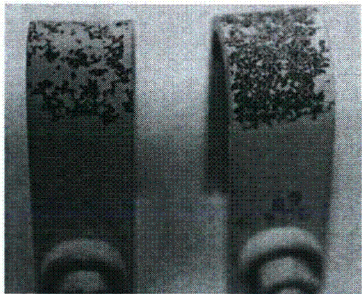
- Mix of intergranular and transgranular cracking

Path Forward for Task 2

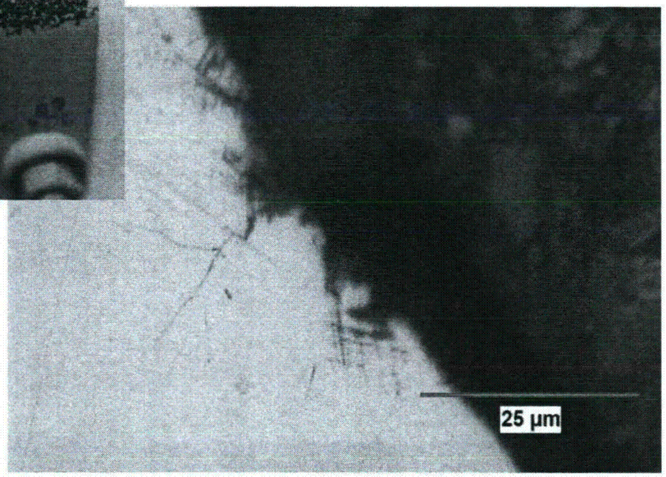
- Discuss and finalize the SCC test matrix with addition of small amount of chloride into the non-coastal salts to evaluate the stress corrosion cracking susceptibility
- Conduct polarization tests in salts proposed

Task 3 — 80 °C-40% RH SCC test (terminated)

As received   Sensitized

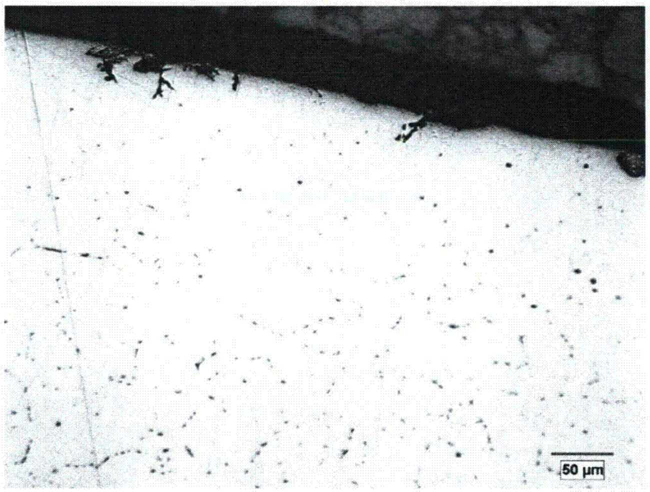


As-received

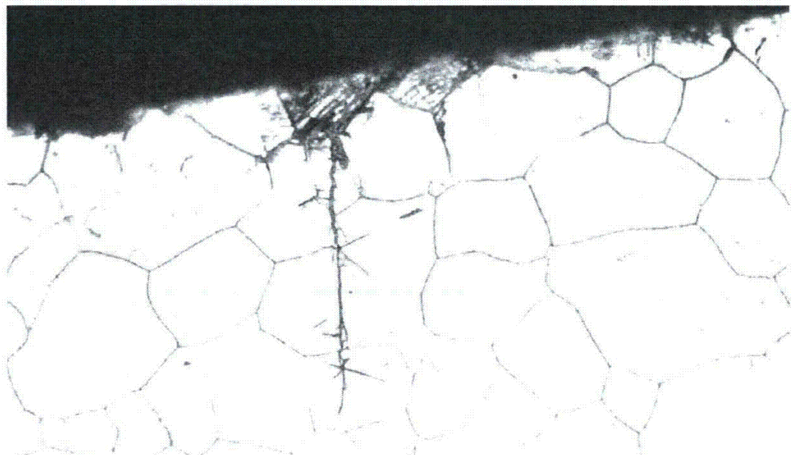


Sensitized

As-received and sensitized



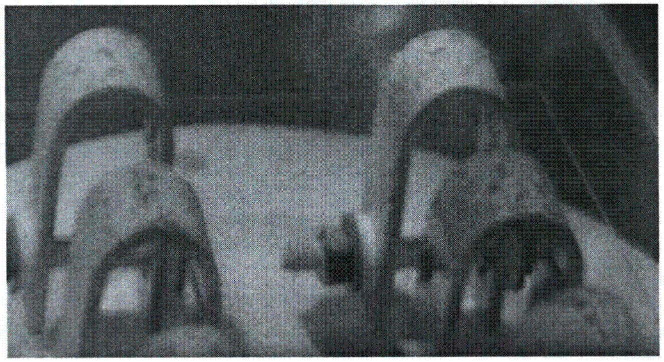
Sensitized and etched



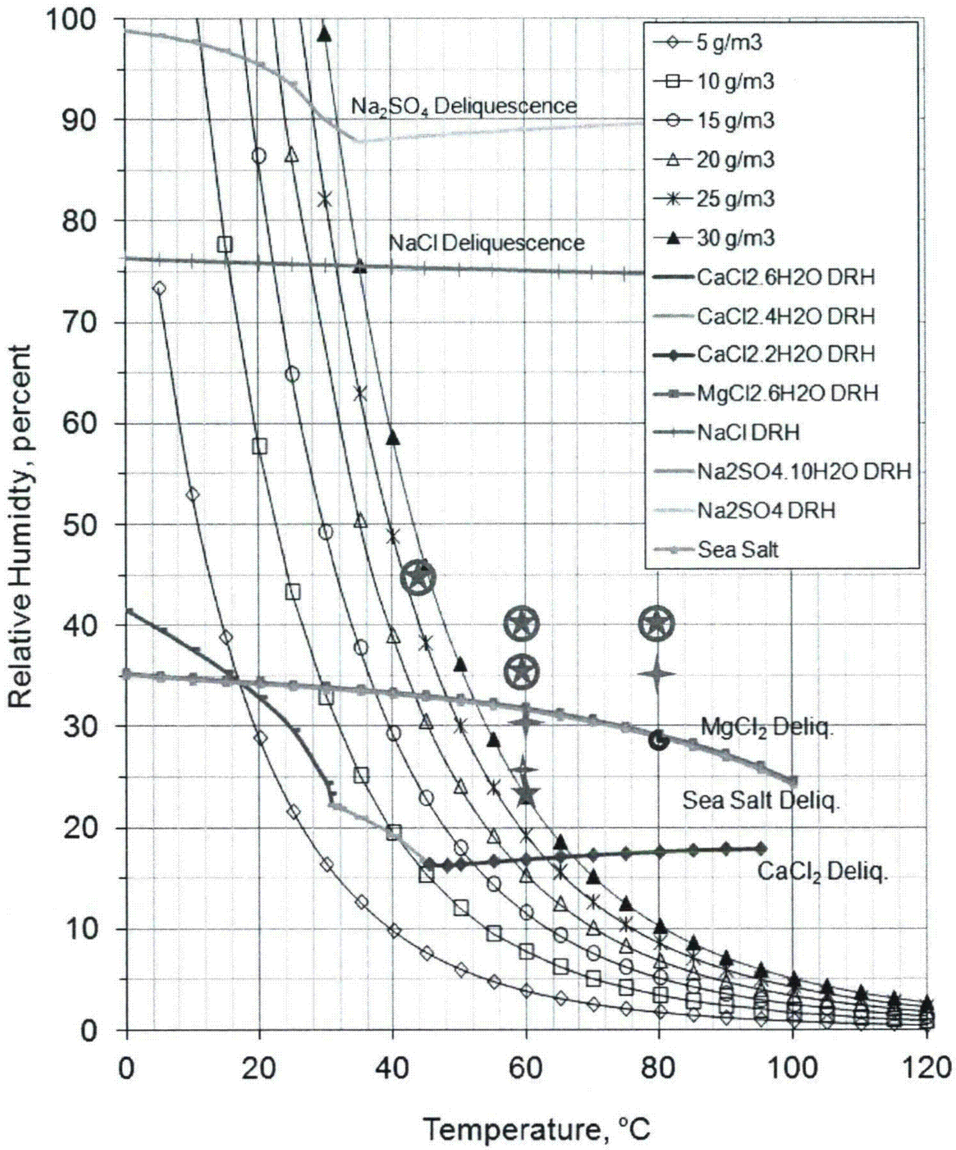
Mix of intergranular and transgranular cracking

Three On-going Task 3 SCC tests

- 80 °C-35% RH
  - Set up on 4/20/12
  - Extensive pitting observed (see photos after 18 days)
- 60 °C-30% RH
  - Set up on 4/23/12
  - Minor pitting observed
- 60 °C-25 % RH test
  - Set up on 5/15/12



Summary and Path Forward of Task 3



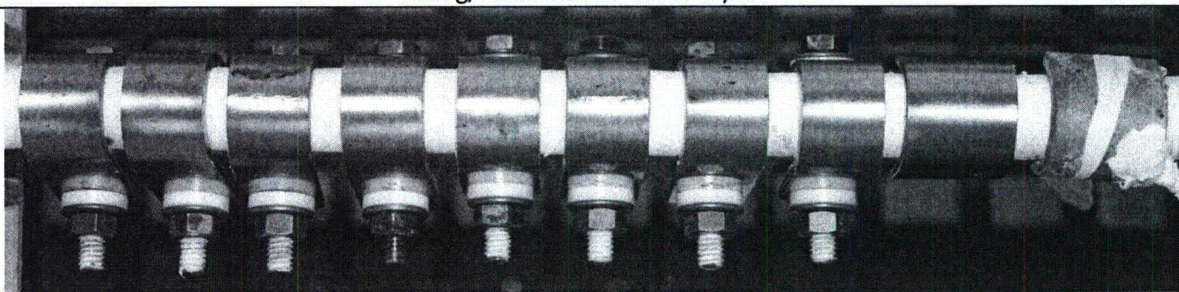
- Terminated and cracking observed
- Terminated, but cracking not observed
- On-going and pits observed
- On-going and pits not observed yet
- Planned
  - 80 °C-28% RH

12/19/12 SCC Project Update

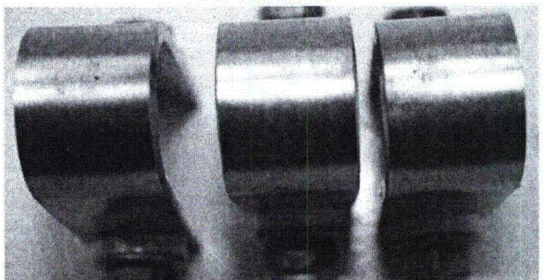
Task 1

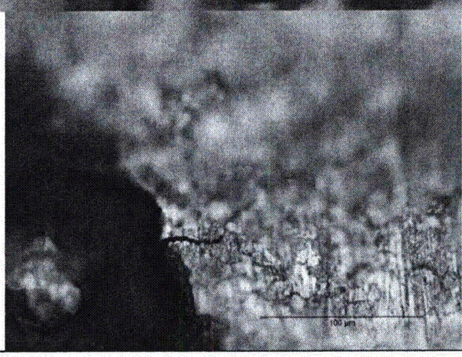
All the remaining Task 1 specimens were pulled last week. Some photos are shown in following figures.

0.1 g/m^2 salt at 35 °C for 1 year

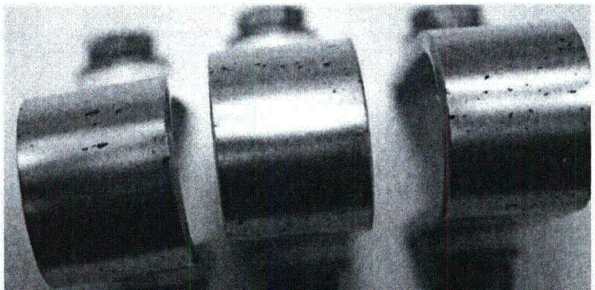


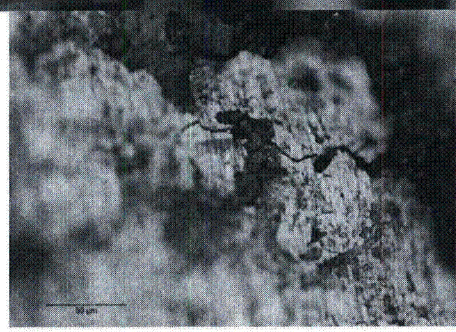
As-received



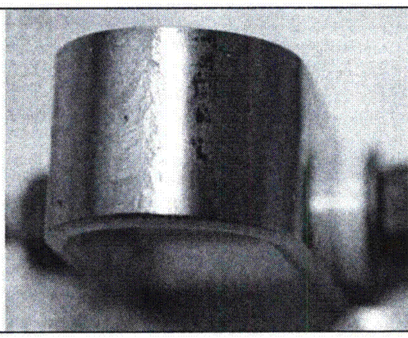


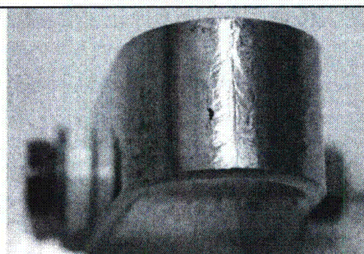
Sensitized

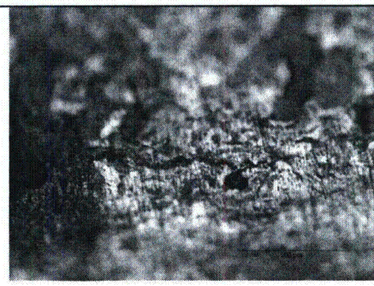




Welded

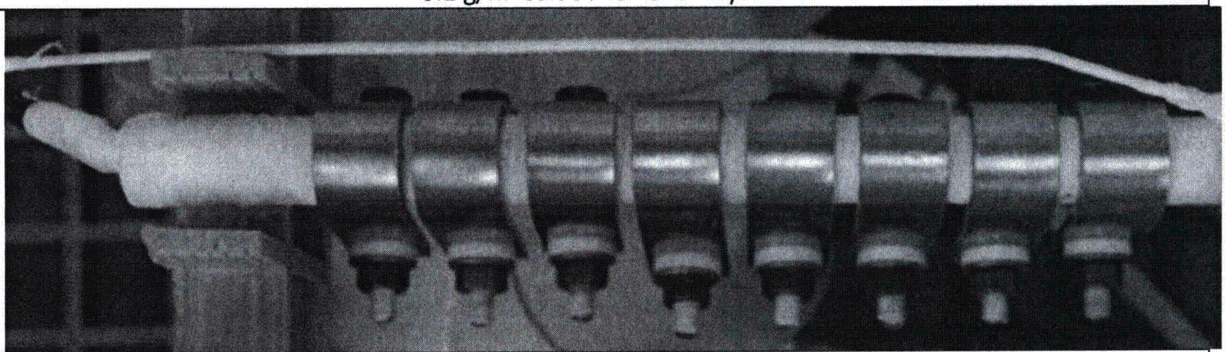






E115

0.1 g/m^2 salt at 45 °C for 1 year



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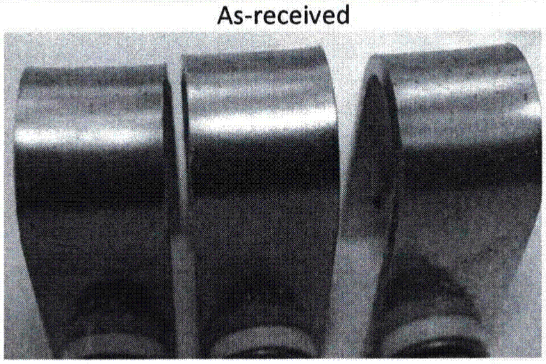

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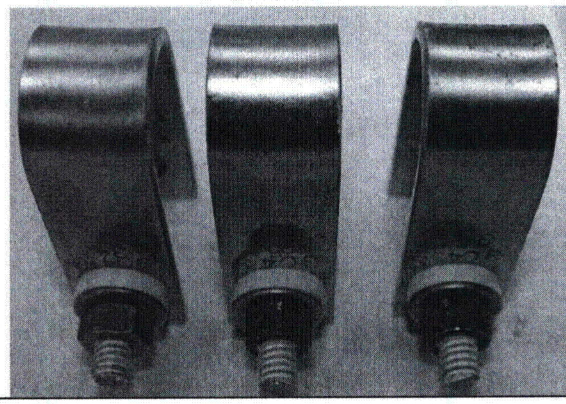
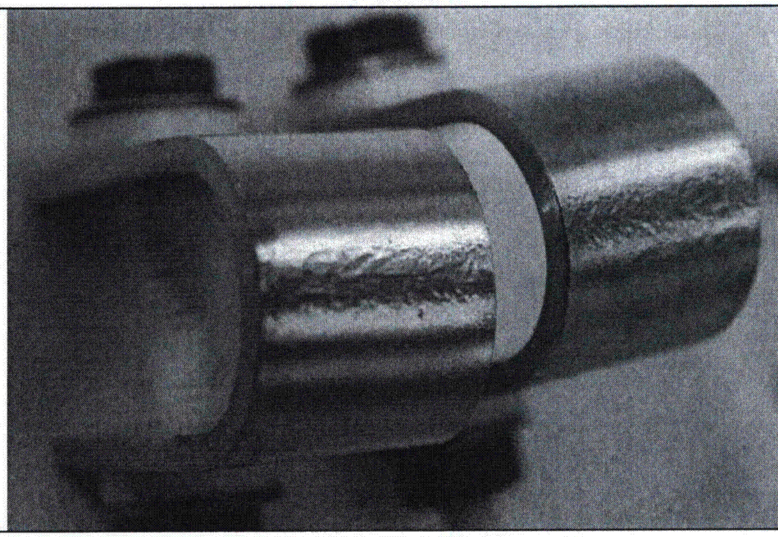
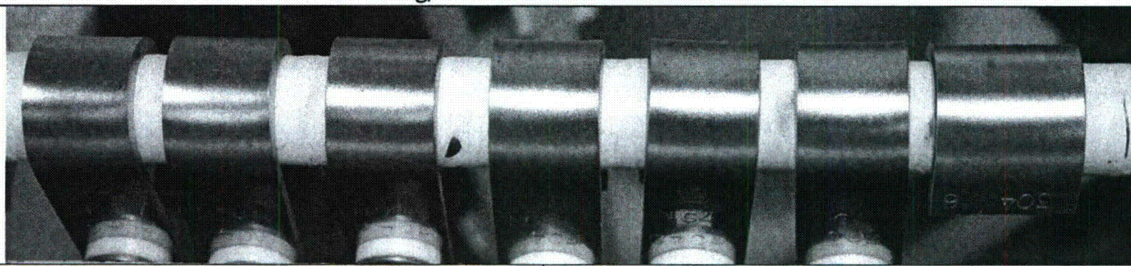


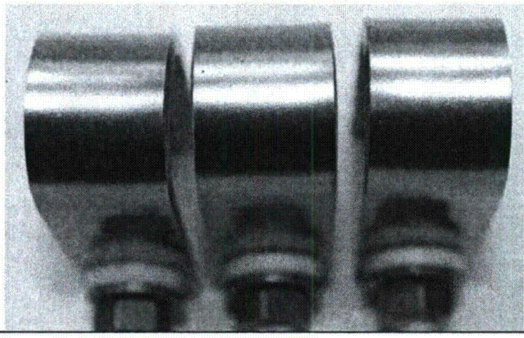
Welded



0.5 g/m^2 salt at 35 °C for 8 months



As-received



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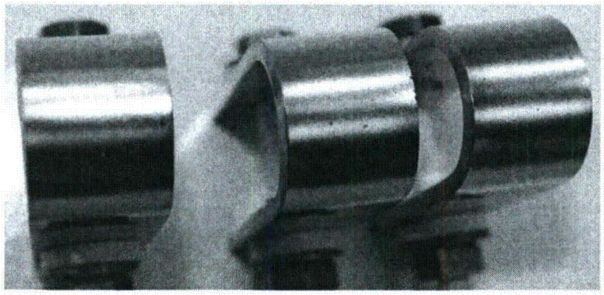
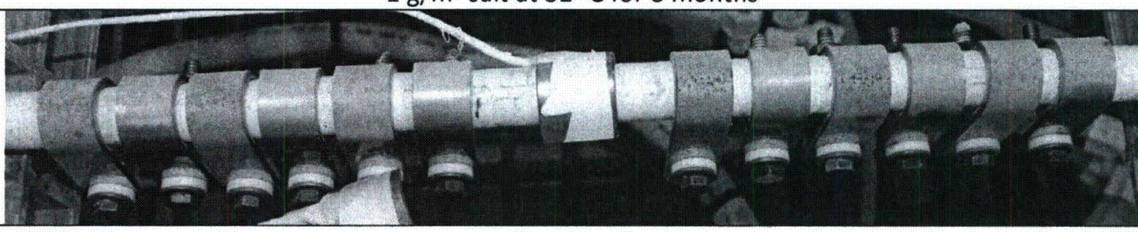
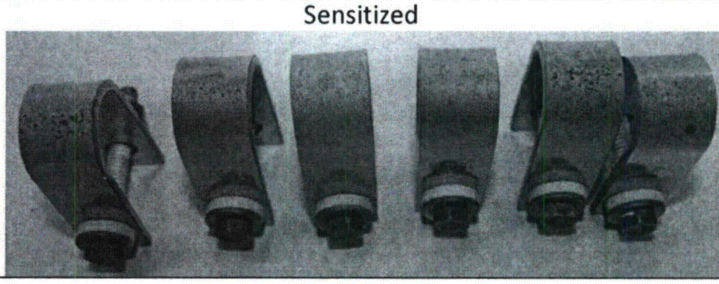


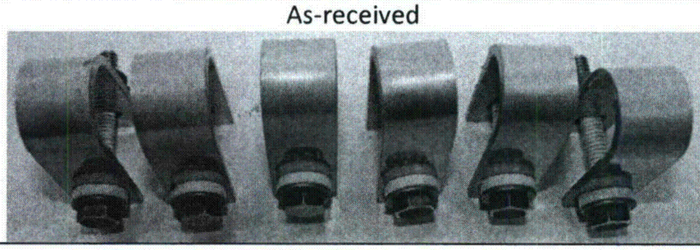
1 g/m^2 salt at 52 °C for 8 months



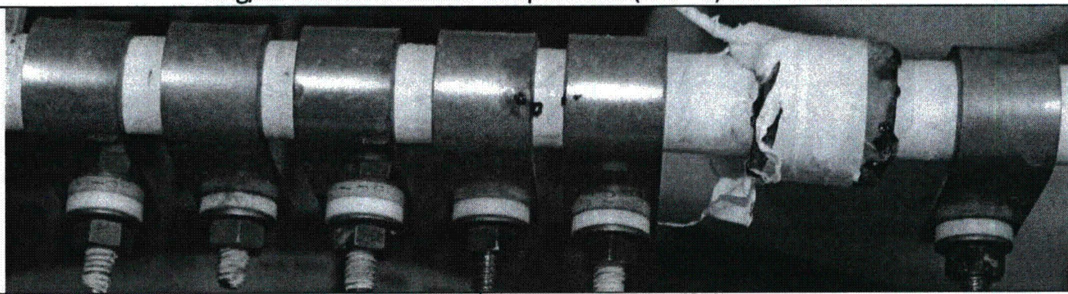
Sensitized



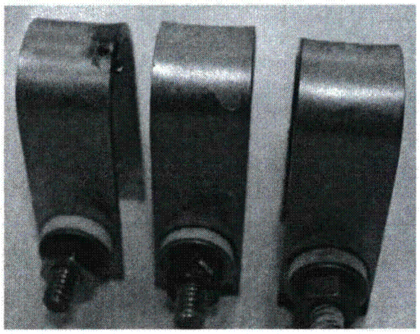
As-received



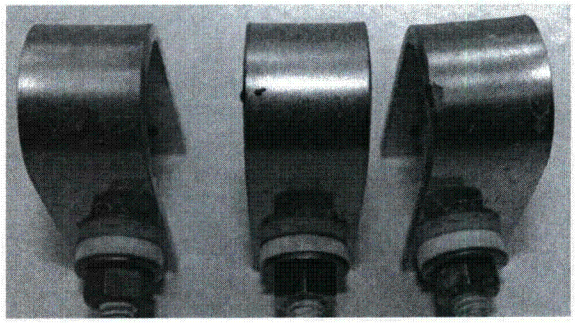
10 g/m^2 salt at chamber temperature (~27 °C) for 8 months



As-received

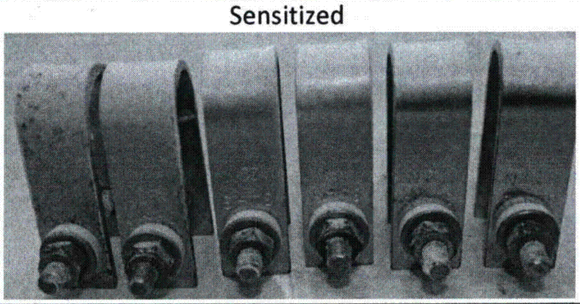


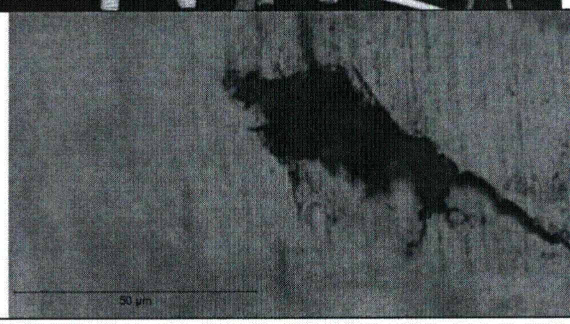
Sensitized



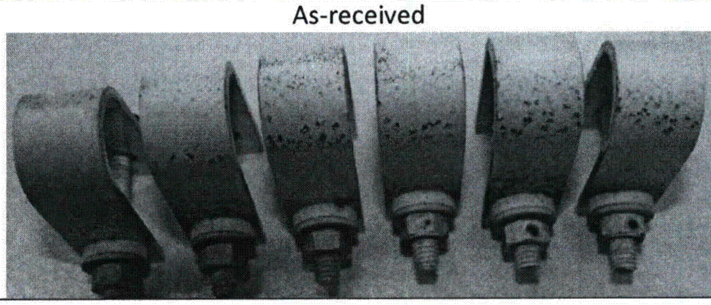
10 g/m^2 salt at 60 °C for 6.5 months

Sensitized





As-received



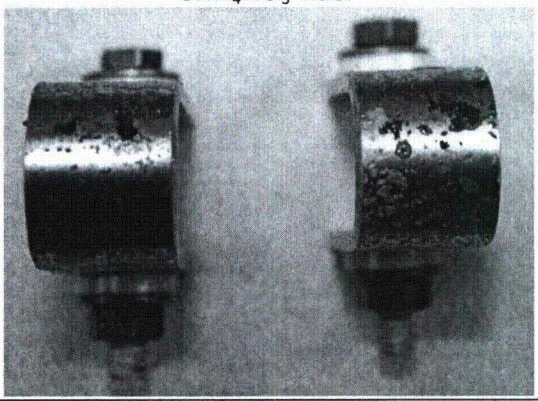
SCC Project Update on 11/19/2012

Task 2

The remaining Task 2 as-received specimens were retrieved after 4-month exposure at 45 °C-44% RH. The photo is shown in Figure 1. There is corrosion on the specimens with $3NH_4NO_3 \cdot NaCl$ salt, but not from the $6NH_4NO_3 \cdot NaCl$ salt. The specimens will be examined further for cracking.

So far, all the Task 2 tests are terminated.

$3NH_4NO_3 \cdot NaCl$



$6NH_4NO_3 \cdot NaCl$

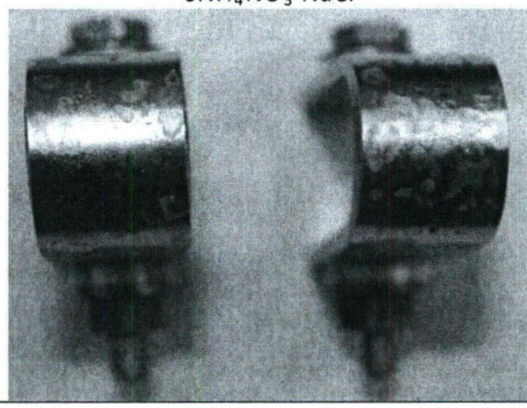
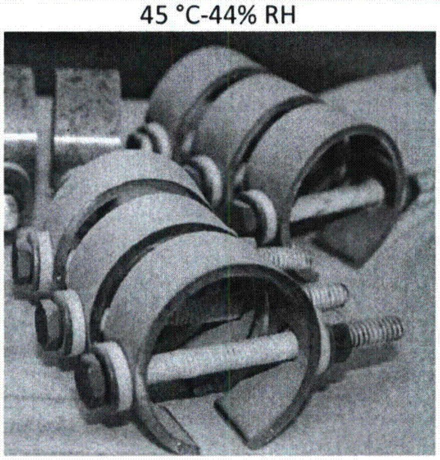


Figure 1. Task 2 as-received specimens after 4-month exposure at 45 °C-44% RH

Task 5

The C-rings with 10 g/m^2 salt stressed to 1.5% strain are in the 52°C-32% RH and 45 °C-44% RH chambers today. The photos are shown in Figure 2.

45 °C-44% RH



52 °C-32% RH

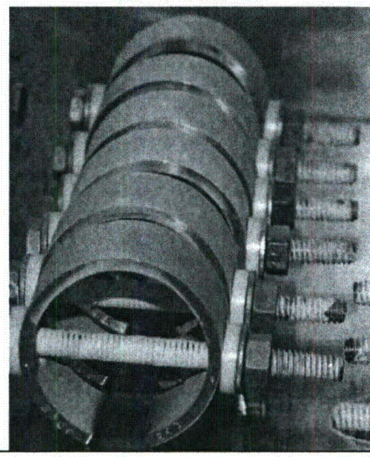
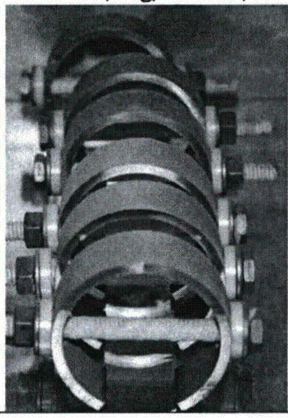


Figure 2. Task 5 specimens with 10 g/m^2 salt stressed to 1.5% strain in the 52°C-32% RH and 45 °C-44% RH chambers

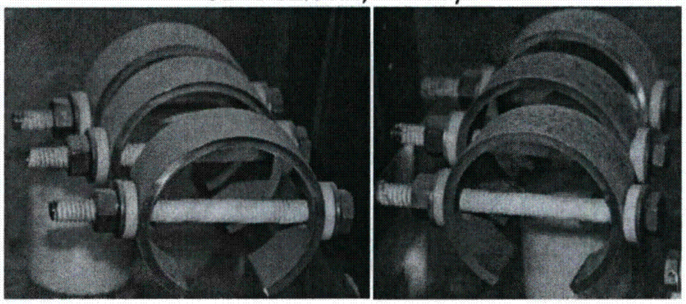
E116

Figure 3 shows the status of other specimens in the chamber.

52 °C-32% RH, 1 g/m^2 salt, 10 days



52 °C-32% RH, ~40 days



45 °C-44% RH

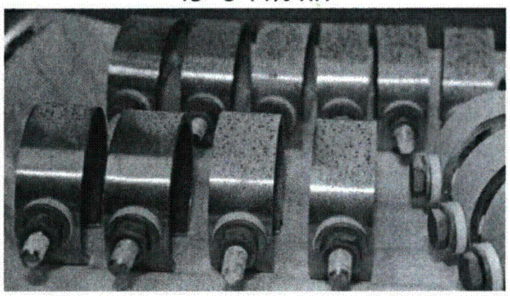
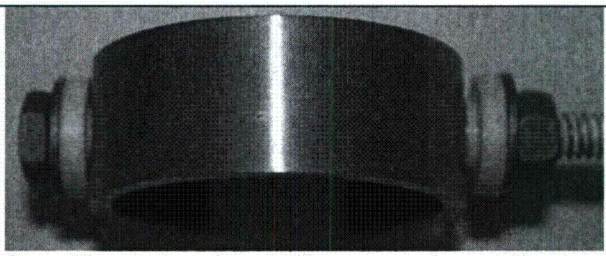


Figure 3. Task 5 specimens with 1 and 10 g/m^2 salt stressed to yield strength in the 52°C-32% RH and 45 °C-44% RH chambers

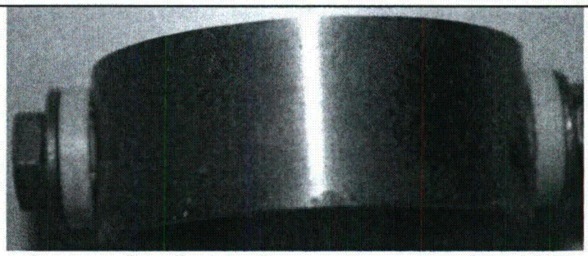
SCC Project Update on 12/4/12

One as-received and one sensitized specimens were pulled from 45 °C/44% RH and 35 °C/72%RH tests at yield strength with 10 g/m^2 salts after 2 month-exposure.

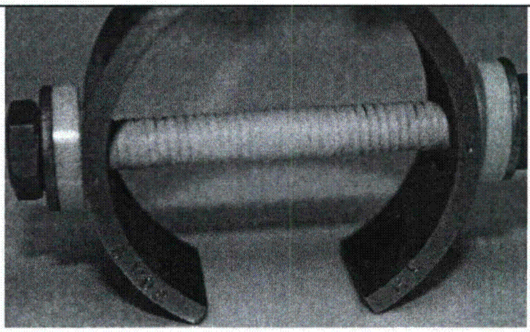
The surface of all the specimens was examined. The photos are shown in Figures 1 and 2 respectively.



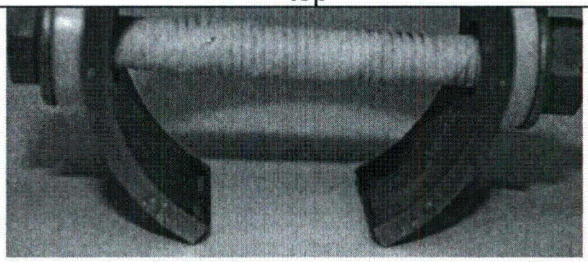
Sensitized showing pitting corrosion at the top



As-received showing some corrosion at the top

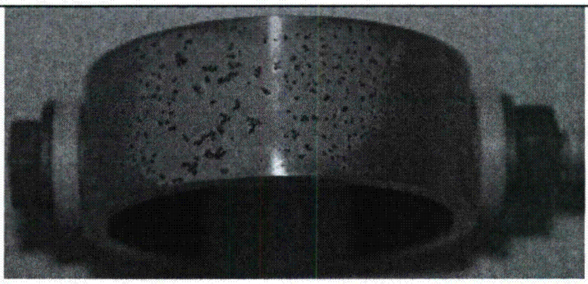


Sensitized showing corrosion inside the C-ring



As-received showing corrosion down at the leg

Figure 1. Specimen photos after two months exposure at 35 °C/72%RH



Sensitized showing extensive pitting corrosion

As-received showing extensive pitting corrosion

Figure 1. Specimen photos after two months exposure at 45 °C/44%RH

The 45 °C samples had pitting all over the surface but no crack and no indication of crack were observed. They will be cross sectioned for further examination. The 35 °C samples did not have much corrosion on the top. There were only a couple of pits, but had more corrosion down the

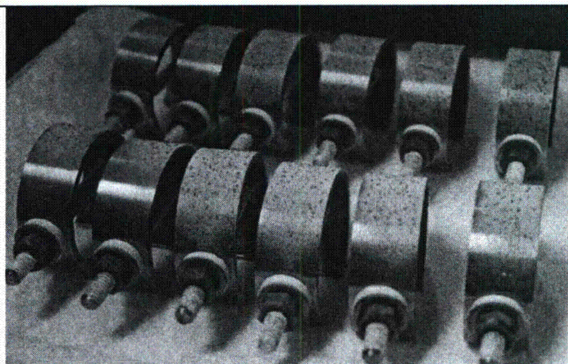
E117

legs indicating the salt seemed to run off the sides. They will not be cross sectioned as the pits are shallow and no indication of crack.

11-7-12-SCC Project Update

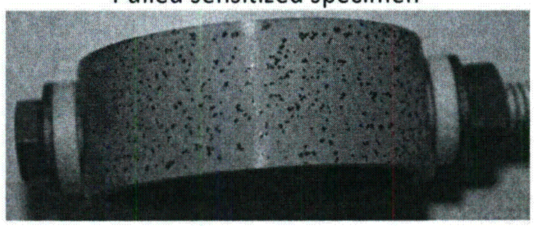
Task 5

One as-received and one sensitized specimens stressed to yield strength were pulled from 45 °C-44% RH and 35 °C-72% RH tests with 10 g/m^2 salts. The photos of the specimens are shown in Figures 1 and 2. Extensive pitting was observed from the 45 °C-44% RH test specimens, but no cracking was observed from both the surface and cross section (cross section of the as-received specimen is not available because the specimen was lost during cutting). For the test at 35 °C-72% RH, in addition to deliquescent salt droplets on specimen surface, a thin layer of salt remained. pits are shown on the surface, but they are not as extensive as the ones from the 45 °C-44% RH tests. Cracks were observed from the sensitized specimen exposed at 35 °C-72% RH.

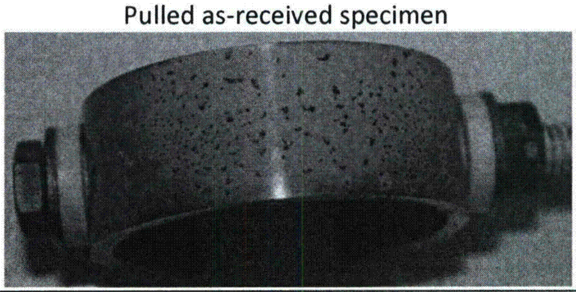


(Top row: Specimens with 1 g/m^2 salt;
Bottom row: Specimens with 10 g/m^2 salt)

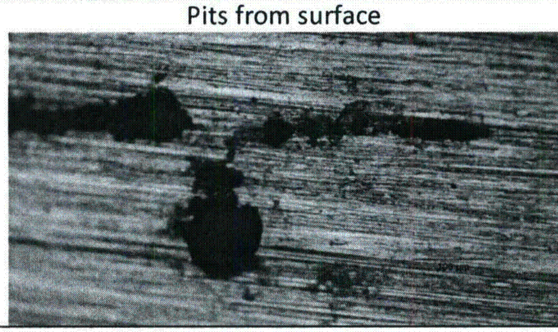
Pulled sensitized specimen



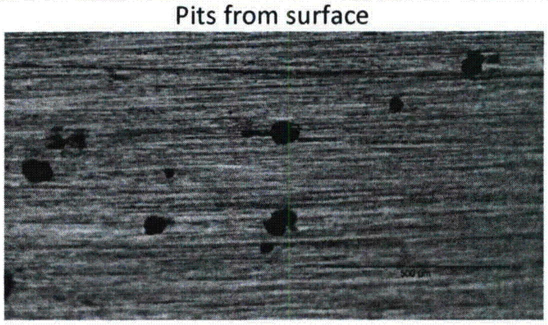
Pulled as-received specimen



Pits from surface



Pits from surface



Pits from cross section

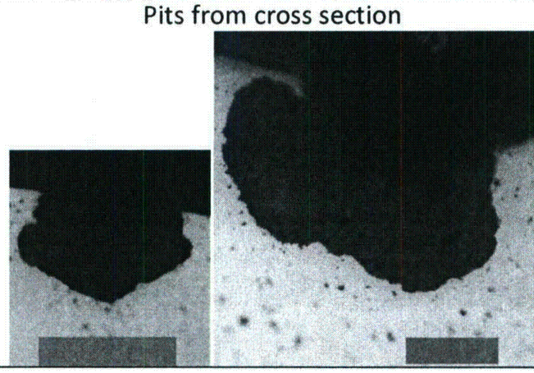
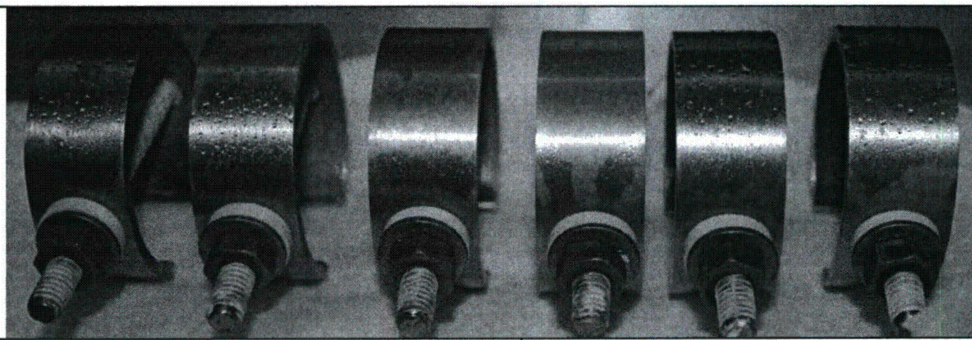
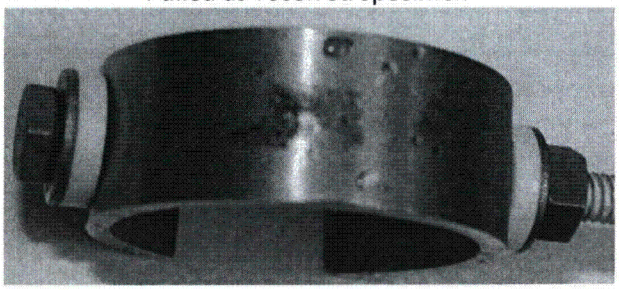


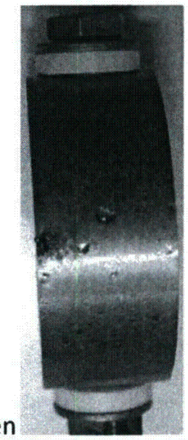
Figure 1. Test at 45 °C-44% RH with 10 g/m^2 salts

E118



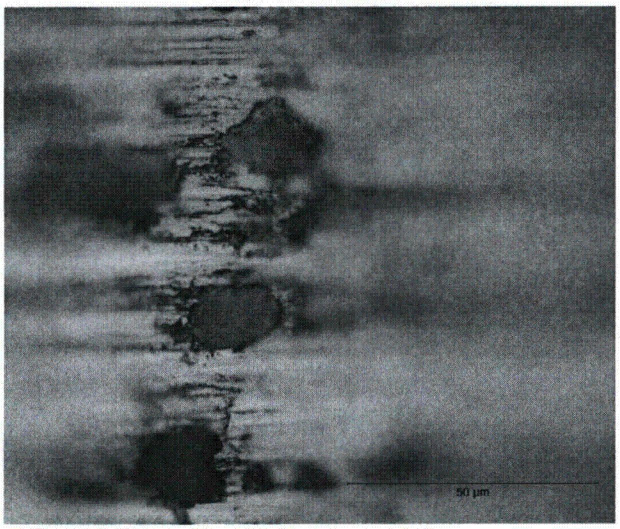
Pulled as-received specimen





Pulled sensitized specimen

Pits from surface



Pits and cracks from surface

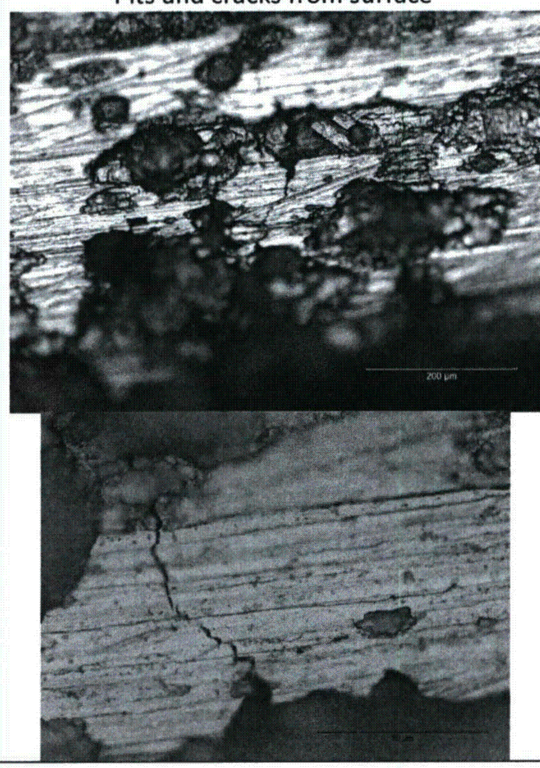


Figure 2. Test at 35 °C-72% RH with 10 g/m^2 salts

Path forward on Task 5:

More specimens will be pulled from the 45 °C-44% RH and 35 °C-72% RH tests with 10 g/m^2 salts in 1 month to continue to examine.

1 as-received and 1 sensitized specimen will be pulled from the 52 °C-32% RH test with 10 g/m^2 salts on 11/12/12 (1-month test duration).

6 as-received and 6 sensitized specimens were stressed to yield strength. The specimens will be deposited with 1 g/m^2 salt. Half of the specimens will be exposed to 52 °C-32% and the other half will be exposed to 35 °C-72% RH. The tests will start this week.

10/5/12

Stress Corrosion Cracking of Spent Nuclear Fuel Dry Storage Canisters in Marine Environments

Greg Oberson
RES/DE/CMB

Division of Engineering Monthly Meeting
March 19, 2012

Participants

- RES: G. Oberson, D. Dunn
- NMSS: T. Ahn, B. Einziger, S. DePaula, J. Rubenstone
- Center for Nuclear Waste Regulatory Analyses (CNWRA): X. He, T. Mintz, B. Pabalan, L. Casceres

Outline

- Overview and regulatory framework
- Conditions for stress corrosion cracking (SCC)
- Results of previous NRC-sponsored work
- Focus of current research
- Summary

The Issue

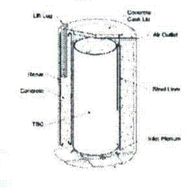
- Many stainless steel spent nuclear fuel (SNF) storage canisters are in coastal areas of the U.S.
- Stainless steel is susceptible to stress corrosion cracking (SCC) in salt-rich environments like the coast.
- We know SCC of the canisters requires deposition of salt on the canisters, moisture, and surface tensile stress.
- We don't know:
  - How much, if any, salt is on canisters.
  - How much salt is a "problem."
  - Conditions where there could be moisture on the canisters.
  - Stresses on canisters.

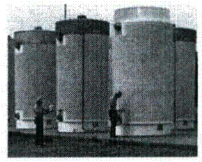
E119

Dry Storage of SNF

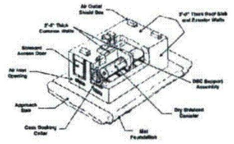
- Used fuel from the reactor is initially placed in the spent fuel pool.
- Spent fuel pools have reached or may approach their capacity in the coming years.
- Used fuel that has cooled for several years in the pool may be removed for dry storage to free up additional room in the pool.
- Dry cask storage involves placing fuel in sealed stainless steel canister in a concrete or metal shell or overpack.

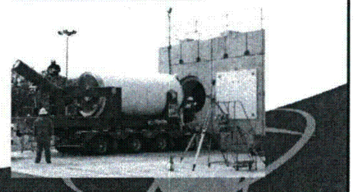
Dry Cask Storage System Configurations





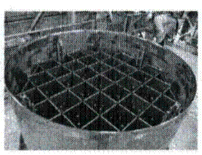
Overpacks have air inlets and outlets open to atmosphere for passive cooling.

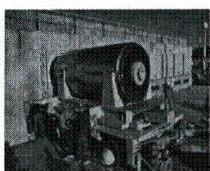




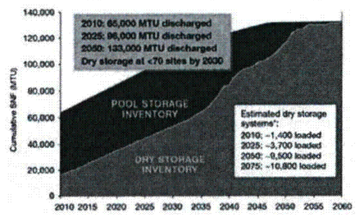
SNF Storage Canisters

Material	Fe	Cr	Ni	Mo	Mn	C	S	P	N	Si	Cu
Type 304	Bal	18.19	8.07	N/A	1.21	0.039	0.002	0.026	0.042	0.55	N/A
Type 304L	Bal	18.14	8.07	0.18	1.29	0.025	0.001	0.025	0.032	0.34	0.27
Type 316L	Bal	16.43	10.13	2.06	1.35	0.019	0.0006	0.027	0.022	0.51	0.32





Usage of Dry Storage



- First dry storage casks loaded in 1986 at Surry.
- Currently less than 25% of domestic SNF inventory is in dry storage.
- EPRI expects all operating plants to implement dry storage by 20[illegible]

Regulatory Framework for Dry Storage

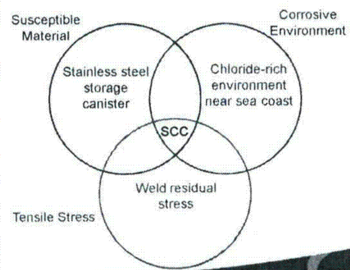
U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

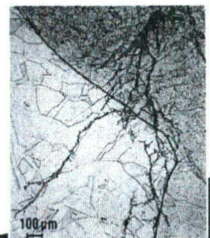
- 10 CFR 71, "Packaging and Transportation of Radioactive Material"
- 10 CFR 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater than Class C Waste"
  - 20 year initial licensing term with subsequent 40 year license renewals
  - License renewal requires description of aging management programs
- 10 CFR 51.23(a) – Waste Confidence Decision
  - SECY-09-0090, "Final Update of the Commission's Waste Confidence Decision
  - Generic finding that SNF can be stored safety without significant environmental impacts at least 60 years beyond licensed operating life
- SRM-COMDEK-09-0001 – staff directed to evaluate adequacy of regulatory programs for storage and transportation of SNF beyond 120 year timeframe
- COMSECY-10-0007 – staff presents project plan for extended storage and transportation (EST) regulatory program review
  - Consider 300 year analytical timeframe for evaluating EST programs
  - Develop of technical gap assessment for dry cask EST
  - Identify potential research activities to address technical gaps

Chloride-Induced SCC of Stainless Steel Storage Canisters

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

NRC technical gap analysis report for extended dry storage indicated that one issue that should be addressed is SCC of stainless steel canisters in locations with chloride-rich atmospheres, such as coastal areas.





ISFSI Locations in the U.S.

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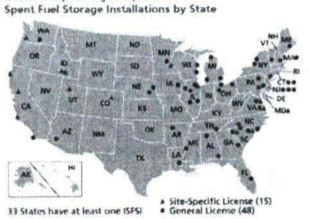

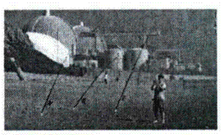

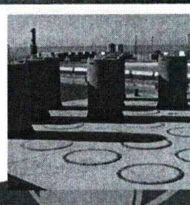

Many independent spent fuel storage installations (ISFSI) are located in coastal areas.

Atmospheric Chloride Deposition

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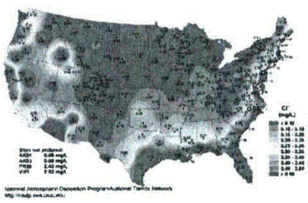

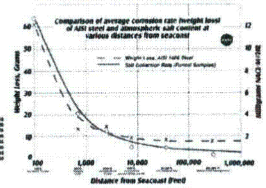

- Atmospheric chloride content can be relatively high in coastal areas.
- Deposition rate on canister is likely to have strong dependence on local environmental conditions, overpack design, canister temperature, canister configuration, and other factors.
  - → Implication for resolving issue by washing canister.

Deliquescence and Efflorescence

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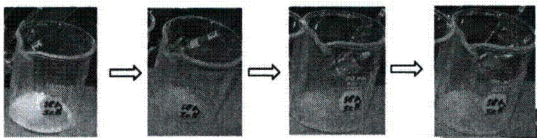
- Moisture must be present on canister need moisture for SCC to occur.
- Salts can absorb atmospheric moisture to form a solution at relative humidity (RH) at or above the deliquescence relative humidity (DRH).
- DRH depends on the chemical composition of the salt and temperature. Salt mixture may behave differently than constituent pure salts.
- Salt will precipitate out of solution, or effloresce, as RH drops below DRH.



60ºC – 10% RH Dry sea salt ⇨ 60ºC – 25% RH Sea salt is moist ⇨ 60ºC – 35% RH Sea salt mostly in solution ⇨ 60ºC – 10% RH Sea salt is mostly dry

Conditions for Deliquescence

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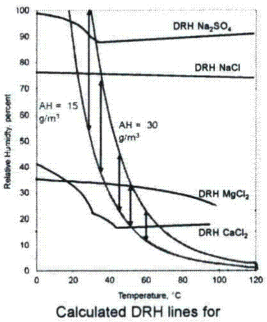

Calculated DRH lines for sea salt constituents

- Main constituents of sea salt are NaCl (58%), $MgCl_2$ (26%), Na_2SO_4 (10%), $CaCl_2$ (2.7%), KCl (1.6%), and $NaHCO_3$ (0.47%)
- Upper bound of absolute humidity (AH) in nature is about 30 g/m^3.
- At AH = 30 g/m^3, RH is low at high temperature, but can be very high at low temperature.
- At temperatures below about 60-70ºC, ambient RH could exceed calculated DRH for sea salt constituents.
- This suggests to focus on lower temperature canisters for evaluating SCC susceptibility.

Canister Temperatures

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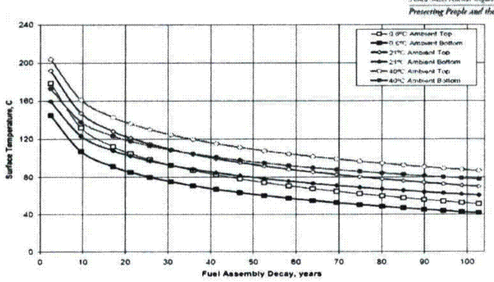

Canister temperatures could reach 60ºC or lower within the timeframe expected for canister service.

Humidity Conditions at ISFSIs

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M = morning avg. RH, A = afternoon avg. RH

Month	Kure Beach, NC		Diablo Canyon		Humboldt Bay		Pilgrim/ Seabrook		Hope Creek		Turkey Point		St. Lucie	
	M	A	M	A	M	A	M	A	M	A	M	A	M	A
Jan.	81	56	77	58	86	66	68	58	76	59	84	59	88	60
Feb.	79	53	83	61	84	64	68	57	78	56	83	57	88	56
March	82	52	86	63	81	62	69	57	76	54	82	56	86	55
April	81	48	89	60	81	59	69	55	77	51	79	64	84	56
May	84	55	92	60	83	59	71	59	79	58	80	58	84	56
June	85	59	91	60	84	58	73	58	81	56	84	66	80	59
July	87	63	95	60	86	59	74	56	83	57	83	63	82	59
August	90	64	93	62	86	61	77	59	87	58	85	66	85	60
Sept.	90	62	92	62	83	58	79	60	88	58	87	66	84	60
Oct.	88	56	86	61	82	59	77	58	88	56	86	62	81	58
Nov.	85	53	81	61	84	63	74	59	84	57	85	62	81	58
Dec.	82	56	84	58	85	67	70	58	79	59	84	60	81	56
Ave.	85	56	86	61	84	61	72	58	82	56	83	61	84	56

At coastal ISFSIs, ambient RH may be relatively high, but conditions are uncertain at canister surface.

Previous NRC-Sponsored Work

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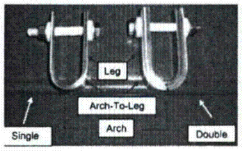
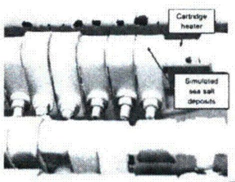


Cycle Number	Chamber Cycle	Cycle Time, min	Cycle Description
1	Salt fog	5	Deposit salt on the specimens
2	Ambient	60	
3	Salt fog	5	
4	Ambient	60	
5	Salt fog	5	
6	Ambient	60	
7	Salt fog	5	
8	Ambient	60	
9	Dry	100	Low relative humidity
10	Increase humidity	125	Increase relative humidity in chamber
11	High humidity	55	Highest relative humidity
12	Dry	180	Low relative humidity



Exposed stainless steel U-bend specimens to salt fog conditions at 43, 85, and 120°C to evaluate susceptibility for cracking. Specimens had high surface salt concentration.

NUREG/CR-7030, "Atmospheric Stress Corrosion Cracking Susceptibility of Welded and Unwelded 304, 304L, and 316L Austenitic Stainless Steels Commonly Used for Dry Cask Storage Containers Exposed to Marine Environments." (2010)

Previous NRC-Sponsored Work

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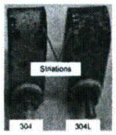
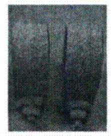
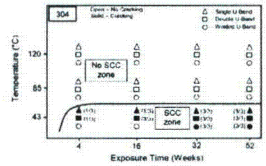
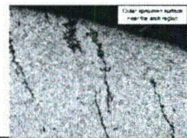
Specimens examined after 4, 16, 32, and 52 weeks exposure. Only specimens tested at 43°C showed cracking.



Exposed 52 weeks, 43°C


Exposed 52 weeks, 120°C





Previous NRC-Sponsored Work

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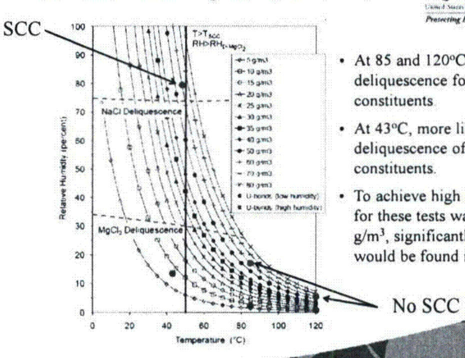

- At 85 and 120°C, limited deliquescence for sea salt constituents.
- At 43°C, more likely deliquescence of sea salt constituents.
- To achieve high RH, AH used for these tests was about 60 g/m^3, significantly higher than would be found in nature.

Objectives for Current Research Program at CNWRA

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- Investigate the minimum amount of salt on the surface needed to support the initiation of SCC for realistic temperature and humidity conditions
  - → Potential regulatory use: help determine significance of salt deposits found in canister inspections.
- Investigate deliquescence behavior of sea salt and its constituent pure salts at a range of elevated temperatures
  - → Potential regulatory use: help determine age or temperature of canisters where deliquescence could occur, thereby narrowing focus for inspection and monitoring.

Minimum Salt Concentration – Experimental Concept

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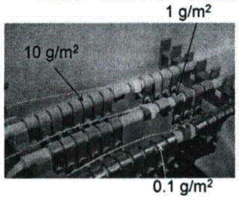
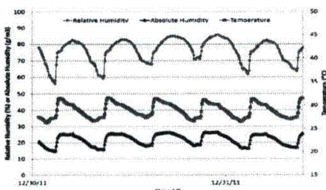
- Deposit known quantities of sea salt onto U-bend specimens, including as-received, sensitized, and welded specimens.
- Expose specimens to temperature and humidity conditions within natural bounds. Observe for evidence of cracking.



Specimens coated with 0.1, 1, or 10 g/m² salt



For specimens at 35 and 45°C, cycle between AH of 15 to 30 g/m³

Minimum Salt Concentration – Experimental Concept

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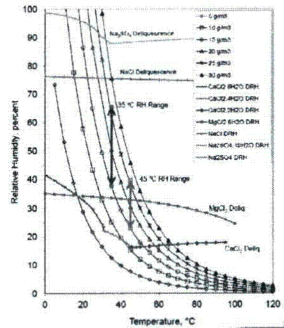

- Calculated DRH for pure salts indicates that at 35°C, humidity cycle will be above DRH for $MgCl_2$ and $CaCl_2$ but below NaCl.
- At 45°C, humidity cycle should cross DRH line for $MgCl_2$.

Minimum Salt Concentration – Initial Observations

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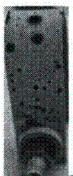
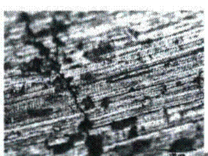
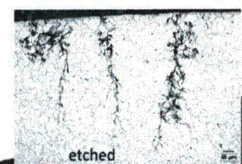
- Within about 3 months, all as-received and sensitized specimens coated with 1 or 10 g/m² salt cracked. Primarily, this is intergranular cracking.
- Some specimens with 0.1 g/m² salt at 35°C have minor pitting.
- Cracking occurred earlier and is more extensive for specimens at 35°C compared to 45°C.
- No welded specimens have cracked yet. There may be some interdendritic attack.







Elevated Temperature Deliquescence – Experimental Concept

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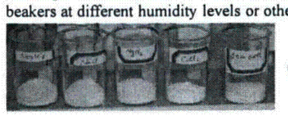
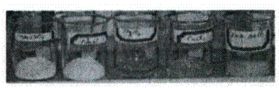
- Determine DRH for sea salt and its pure salt constituents at temperatures in the range of 60 to 80°C. DRH could be measured by observing salts in beakers at different humidity levels or other analytical methods.



60°C, 10%RH



60°C, 50%RH

- Expose U-bend specimens to different humidity levels at the elevated temperatures to determine whether SCC could occur in realistic environmental conditions.

Elevated Temperature Deliquescence – Initial Observations

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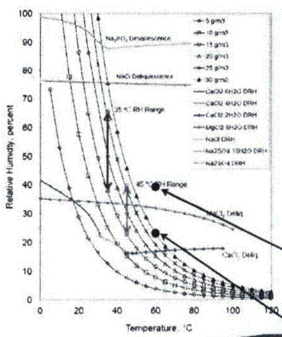

- At 60°C, measured DRH for $CaCl_2$ is about 20%, $MgCl_2$ about 35%, and sea salt about 55%. NaCl and Na_2SO_4 did not deliquesce up to 55% RH. These are generally consistent with calculated values.
- U-bend specimens with 10 g/m² salt at 60°C and AH = 30 g/m³ (RH = 22%) did not crack after 1 month.
- Similar specimens cracked within 1 week at RH 40%.

Ongoing or Future Work

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- Minimum salt concentration:
  - Testing at higher temperatures than 35 and 45°C
  - Testing at salt concentration between 0.1 and 1 g/m²
  - Testing on flat, non U-bend specimens
- Elevated temperature deliquescence:
  - Additional SCC tests at 60°C at RH between 22 and 44%.
  - Testing at 80°C at various humidity levels.
- Effects of non-coastal atmospheric species
- Current contract goes through September, 2012.

Collaborative Research

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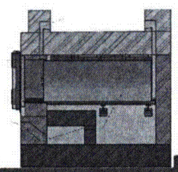
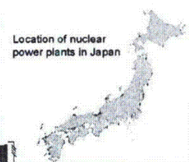
- Addressing the range of questions for this issue is beyond the scope of NRC and will require collaboration.
  - NRC is engaged in EPRI Extended Storage Collaboration Program with domestic and international industry and vendor representatives.
  - Seeking to leverage interactions with Japan, CRIEPI
- Industry is currently focused on identifying what is on the canisters.
  - Pilot inspection planned for this summer at Calvert Cliffs.
  - Identify range of examination methods, such as visual, surface swipe





Mitigation Approaches

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- New canisters
  - Corrosion resistant alloys
  - Stress mitigation
    - Peening
    - Burnishing
  - Coatings
- Canisters already in service
  - Enhanced inspection and monitoring
  - Washing

Summary

- NRC is assessing the technical bases for dry storage of SNF beyond 120 years.
- Stainless steel storage canisters may be susceptible to SCC in locations with chloride-rich atmosphere, such as coastal areas.
- Lower temperature canisters are more likely to be susceptible for SCC.
- Ongoing research is investigating the minimum chloride concentration for SCC initiation in realistic environmental conditions as well as elevated temperature deliquescence behavior.
- There is limited available information concerning the condition of canisters in the field, but pilot inspection programs are under development.

4/15/13

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U.S. NRC Sponsored Research on Stress Corrosion Cracking Susceptibility of Dry Storage Canister Materials in Marine Environments

[1]G. Oberson, [1]D. Dunn, [2]X. He,
[2]T. Mintz, [2]R. Pabalan, [2]L. Miller

[1]U.S. Nuclear Regulatory Commission, Washington DC
[2]Center for Nuclear Waste Regulatory Analyses, San Antonio, TX

WM2013 Symposia 2013 Waste Management Symposium, February 24-28, 2013, Phoenix, AZ

U.S.NRC

Outline

- Background and motivation
- Status of current work
- Path forward
- Summary

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BACKGROUND AND MOTIVATION

U.S.NRC

Stress Corrosion Cracking (SCC)

- NRC identified SCC of dry storage canisters exposed to chlorides as a high priority information need.
  - Many canisters are fabricated from austenitic stainless steel.
  - Airborne salts or particulates could migrate through vents in the vault or overpack and be deposited on the canister.
- Sources of chlorides may include marine environments, condensed cooling tower water, salted roads.
- Operational experience at plants indicates events where SCC of stainless steel components was attributed to atmospheric chloride exposure.

E120

U.S.NRC — Previous Work

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- NUREG/CR-7030 – NRC study published in 2010
- Japanese studies – primarily by Central Research Institute for the Electric Power Industry (CRIEPI)
- NRC and CRIEPI testing indicates that austenitic stainless steel is susceptible to SCC when exposed to chloride salts in certain conditions of salt quantity, temperature, humidity, and stress level.
- Limited data, differing test methodologies and interpretation of results hinder absolute determinations of SCC susceptibility.

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U.S.NRC — Current Project

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- Further NRC research was warranted to identify conditions for SCC susceptibility
  - Temperature
  - Humidity
  - Salt quantity
  - Stress level
- Scope of tasks
  - Deliquescence testing
  - SCC testing at absolute humidity (AH) less than 30 g/m^3
  - SCC testing at elevated temperatures
  - SCC testing at high relative humidity (RH) conditions
  - SCC testing at different stress/strain levels

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STATUS OF CURRENT WORK

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U.S.NRC — Deliquescence Testing

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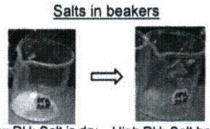
- Background: Deliquescence of dry salt on canisters in conditions above a certain RH may introduce moisture to support SCC
- Test objective: Identify the deliquescence RH (DRH) for sea salt at various temperatures
- Test methodologies:

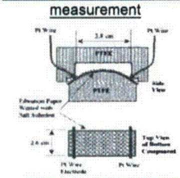
Salts in beakers



Low RH: Salt is dry High RH: Salt has deliquesced

Observe beakers for moisture absorption

Conductivity cell impedance measurement



Record impedance drop as function of RH

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U.S.NRC Deliquescence Testing

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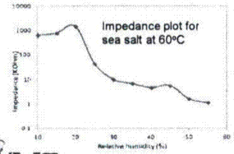
- Beaker test results: DRH based on visual observation of moisture absorption

		Temperature (°C)		
		45	60	80
Species	Sea salt (ASTM D1141-98)	43%	35%	25%
	$CaCl_2$	28%	20%	20%
	$MgCl_2$	37%	35%	25%
	NaCl	None up to 59%	None up to 60%	None up to 59%
	Na_2SO_4	None up to 59%	None up to 60%	None up to 59%

- Good agreement between beaker and impedance cell results



- DRH for sea salt is close to that of $MgCl_2$
- DRH decreases somewhat with increasing temperature

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U.S.NRC SCC Testing at AH Less Than 30 g/m³

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- Background: AH near canister surface is unknown but 30 g/m³ may represent a maximum for certain ambient atmospheric conditions.
- Test objectives:
  - Identify whether SCC can initiate at AH less than 30 g/m³
  - Investigate effects of surface salt concentration and material condition on SCC susceptibility
- Test methodology:
  - Deposited 0.1, 1, or 10 g/m² of sea salt on ASTM G30 U-bend specimens
  - Specimens were Type 304 in as-received, sensitized, or as-welded with Type 308
  - Specimens were exposed to cyclic AH between about 15 and 30 g/m³ at various temperatures

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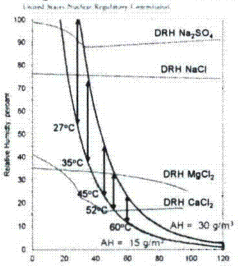
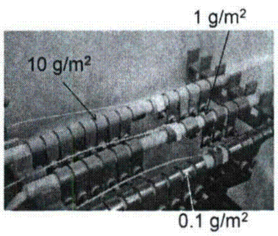
U.S.NRC Test Setup

United States Nuclear Regulatory Commission



Test temperatures (arrows indicating span of humidity cycle) and lines showing calculated DRH for sea salt constituents



U-bend specimens in environmental test chamber

- Specimen temperatures controlled by heating elements
- Specimens exposed for up to 1 year

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U.S.NRC Test Results

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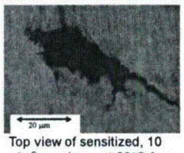
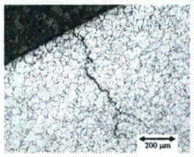
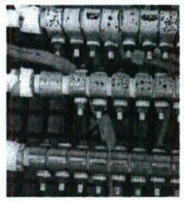
Temperature (°C)	SCC Observed?	Lowest salt concentration at which SCC was observed
27	No	N/A
35	Yes	0.1
45	Yes	0.1
52	Yes	1
60	Yes	10



Top view of sensitized, 10 g/m² specimen at 60°C for 6.5 months



Cross section of sensitized, 0.1 g/m² specimen at 45°C after 4 months



Specimens at 10 g/m² (top), 1 g/m² (middle), and 0.1 g/m² (bottom)

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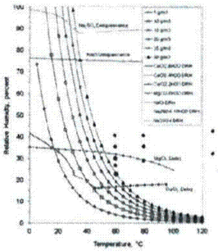
U.S.NRC — SCC Testing at Elevated Temperatures

United States Nuclear Regulatory Commission
Protecting People and the Environment

- Background: Japanese have reported SCC initiation at temperature up to 80°C at 15% RH. NUREG/CR-7030 testing showed cracking only at 43°C, not at 85 or 120°C.
- Test objective: Identify whether SCC can initiate at temperatures in the range of 60 to 80°C
- Test methodology:
  - Deposited 10 g/m² of sea salt on U-bend specimens
  - Exposed specimens to static RH at 60 and 80°C
  - Constant exposures at AH between about 40 and 25 g/m³



- SCC observed
- No SCC observed

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U.S.NRC — Test Results

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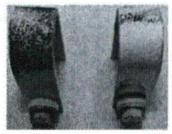
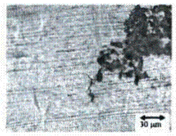
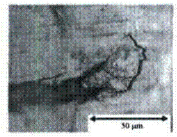
- SCC initiation was observed at 60°C as low as 25% RH and at 80°C as low as 28% RH.
- AH for tests was above 30 g/m³ but results indicate that SCC can initiate by salt deliquescence if RH is sufficient for deliquescence.



Specimens after 8 weeks at 80°C and 28% RH



Micrograph of specimen at 80°C and 28% RH



Micrograph of specimen at 60°C and 25% RH

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U.S.NRC — SCC Testing at High RH

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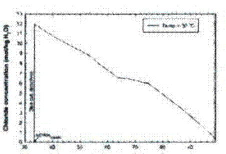
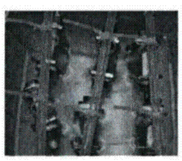
- Background: Equilibrium chloride concentration in solution decreases with increasing RH. Dilution of chlorides at high RH could reduce SCC susceptibility.
- Test objective: Determine whether SCC can initiate in conditions of high RH
- Test methodologies:
  - Immersed U-bend specimens in prepared saturated solutions for 30°C and 90% RH
  - Deposited 10 g/m² of sea salt on U-bend specimens for exposure at 30°C and 90% RH



Calculated chloride concentration in saturated sea salt solution as function of RH at 30°C



U-bend specimens immersed in solution

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U.S.NRC — Test Results

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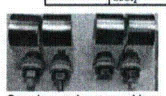
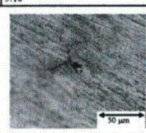
- For specimens with deposited salt, salt quickly deliquesced and ran off sides of specimens with no SCC observed.
- For immersed specimens, pitting and SCC were observed within 5 weeks.

Chloride and Salt Concentrations in Saturated Solutions at 30°C and 90% RH

		Chloride Concentration (mol/kg H_2O)	Salt Concentration (g/kg H_2O)
Solution	Sea salt	2.71	203
	NaCl	2.79	163
	$MgCl_2$	3.01	306
	$CaCl_2$	3.16	232



Specimens immersed in sea salt after 5 weeks, as received (L); sensitized (R)



Cracking on surface of specimen immersed in sea salt

Cracking on surface of specimen immersed in $MgCl_2$

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U.S.NRC — C-Ring SCC Testing

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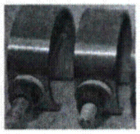
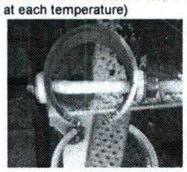
- Background: U-bend specimens represent a highly strained state, 13-14% at the apex. These may not be representative of canister conditions.
- Test objective: Use C-ring testing to control applied stress/strain and investigate effect on SCC initiation.
- Test methodologies:
  - Specimens fabricated following ASTM G38-01 and deposited with 1 or 10 g/m^2 of salt
  - Specimens were strained to slightly above yield stress (~0.3% strain) or 1.5% strain, as measured by strain gage.
  - Specimens were exposed at conditions of 35°C and 72% RH, 45°C and 44% RH, and 52°C and 32% RH (AH ~ 30 g/m^3 at each temperature)





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U.S.NRC — Test Results

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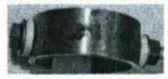
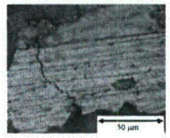
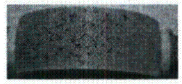
- Pitting was observed on most specimens and cracking on some specimens
- Further specimen examination is in progress to confirm crack initiation

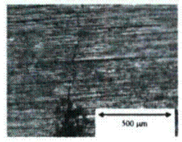
35°C, yield stress, 10 g/m^2 salt





52°C, yield stress, 10 g/m^2 salt





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PATH FORWARD

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U.S.NRC — Schedule for Completion of Test Program

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- Program began in October 2011
- Experimental work is complete
- NUREG/CR report expected to be published in Summer of 2013
- No additional laboratory testing by NRC is currently planned

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Related Activities

- Engagement with industry and other stakeholders
  - EPRI Extended Storage Collaboration Program (ESCP)
  - NEI Regulatory Issue Resolution Protocol (RIRP)
- Other NRC projects
  - Modeling of weld residual stresses
  - Functional monitoring of structures systems and components
  - Canister inspection techniques

Summary

- Test program was proposed to identify conditions where austenitic stainless steel could be susceptible to atmospheric chloride induced SCC.
- Test results indicate that in certain conditions, SCC can initiate for:
  - Surface salt concentrations as low as 0.1 g/m^2
  - AH less than 30 g/m^3 for temperatures between 35 and 60°C
  - Temperature up to 80°C at RH above about 28%
  - Stress level as low as material yield stress for temperatures between 35 and 52°C and AH less than 30 g/m^3
- Test results will be used to support engagement with industry and other stakeholders and help determine where further information is needed.