UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

Before the Atomic Safety and Licensing Board

In the Matter of	)	
		Docket No. 50-346-LR
First Energy Nuclear Operating Company	)	
(Davis-Besse Nuclear Power Station, Unit 1)		August 17, 2012
.	)	

*   *   *   *   *

INTERVENORS' FIFTH MOTION TO AMEND AND/OR SUPPLEMENT PROPOSED CONTENTION NO. 5 (SHIELD BUILDING CRACKING)

APPENDIX IX: NRC FOIA RESPONSES (B-51 THROUGH B-53) ; TURKEY POINT EVENT REPORT; NRC INFORMATION NOTICE 2010-12: CONTAINMENT LINER CORROSION

S

Davis-Besse Root Cause Review – Status Call 1-19-2012

Please describe the work completed and still ongoing/planned for each of your subcontractors:

PII-Core Bore Sample Testing (C), Finite Element Model Development (Jan 30), FMEA Report - TBD
MPR- Initial Evaluation of failure mechanisms and third party review of PII report.
Vatic Associates- Initial Evaluation of failure mechanisms and third party review of PII report.

Status of Your Team Work Products

Event & Causal Factor Chart with Barrier Analysis- NA
PII Fault Trees- NA

FMEA- (Worksheet for each failure mode or group of modes? – Yes separate failure modes and effects analysis worksheet expected for each item in RC FMA matrix. Preliminary version mid-week (Jan 25th)

Finite Element Model of the Containment Shield Building (Abaqus Code)- (Jan 30th)

- ABAQUS software controlled by licensee's contractor?
- Model "benchmarked" for what the evaluation is attempting to determine?
  - Model results compared against test data?, or
  - Model results compared against a known solution?
- ABAQUS software errors that could affect analysis results, how are they evaluated and controlled?

No information on these questions other than model developed and applied to Crystal River containment cracking issue. Updated to reflect DB Shield building including specific mechanical properties of DB SB materials (concrete/rebar). Also, believe that PII will run ABAQUS against another software model to validate results. Licensee expects this information to be included in final PII report.

Root Cause Report-TBD, draft report will not be ready before end of month.

Explain Purpose of Purdue Univ Testing and Why not Used by RCT- To evaluate the excess capacity or the bond strength of Lap splice configuration for DB SB rebar configurations. Dr. Susan of Purdue (concrete expert) and Dr. Darwin Kansas St Univ (Bond strength expert) will collaborate under Bechtel direction to perform test. Licensee will fabricate a beam section with concrete and rebar that matches 28 day strength requirements for DB SB. Beam will be placed in bending until failure by laminar cracking. Then additional tensile capacity of rebar w cracked concrete will be measured. Suspect 30-50% capacity with this configuration. Result will be used to support a use-as-is disposition for the existing concrete cracking configuration and will not be needed for root cause efforts. NRC will be informed of test schedule.

B/51

Questions Related to the NRC request for information.

Item #11- Core Bore Sample Plans

Core Bore Data at 4 Locations Needed to Support/Refute Groups of Critical Failure Modes:

- Explain Each of the 4 Groups of Failure Mechanisms- **early work not related to PII failure modes.**

- Were Core samples taken as planned including location and size for core samples? **Yes.**

- How many were harvested at each location to support these tests? **One or more.**

Davis-Besse Shield Building Concrete Bore Samples Test Schedule:

- Were the three core samples -4" diameter at Photometrics and the three 4" diameter at Univ of Colorado (+two 3" dia samples) sample selected per document above? **Yes.**

- How was the location for harvesting the Three 2" dia core bores samples sent to Photometrics facility determined? **No specific logic, used 2" samples from cracked and uncracked locations for carbonation examinations.**

- Were any of the 2" bore samples needed to confirm Group 1 and 2 failure modes? If so, why is this acceptable? **No, just to confirm extent of cracking**

- Typo after discussion of samples at Un of Colorado? **Yes.**

Item #15 – Schedule for completion of RCR

- You indicated freeze thaw test not complete by need date to end RCR due to equipment failure. What failed? **Lost power to test rig**. Is freeze/thaw still a viable potential cause? **Yes**, Why is this test information not needed by RCT? **We have data from original construction freeze/thaw tests.**

-Stated that freeze/thaw testing had been done on the first pour of the shield building. What testing was done and did this testing include both the type 1 and type 2 cement used in construction of the SB? **Freeze thaw done early on for only the type 2 cement because of time of year below grade portion of SB was poured (winter).**

How sensitive is your analytical model to obtaining accurate material properties in this area? **Don't know yet if model can predict freeze thaw.**

Other

Current Leading Potential Causes For Cracking? **At least 8 and could be combination of several.**

Do you still believe that a Root Cause for SB cracking can be identified? **Yes, but verdict is still out since this is not straightforward.**

52

Murphy, Martin

From: Hiland, Patrick MRR
Sent: Thursday, January 19, 2012 8:42 AM
To: Murphy, Martin
Subject: RE: Without the Root Cause

On target. Thanks.

From: Murphy, Martin MRR
Sent: Thursday, January 19, 2012 8:00 AM
To: Hiland, Patrick
Cc: Hoang, Dan; Manoly, Kamal; Jessup, William
Subject: FW: Without the Root Cause

Pat,

See attached for the reasoning why it was acceptable for startup without the RCA. I think Dan and Kamal did a nice job getting this to a level where the general public will comprehend the reasoning.

Hope it hits the mark. If you additional changes let me know.

Marty

From: Hoang, Dan MRR
Sent: Wednesday, January 18, 2012 2:28 PM
To: Murphy, Martin
Cc: Manoly, Kamal; Billy Jessup
Subject: Without the Root Cause

Marty,

Attached is the latest (Kamal's reviewed and input), Hope it will do

Thanks,

Dan

B/52

53

Davis-Besse Root Cause Review – Status Call 1-26-2012

Davis-Besse – Key staff on call (John Hook and Kevin Browning & others)
RIII- Mel Holmberg, Atif Shaikh, Elba Sanchez-Santiago, James Neurauter, Dan Kimble
NRR- Samuel CuadradoDeJesus and several NRR technical staff

Status of Testing Core Samples

Any plans to salvage freeze/thaw test data? Not initially, since it will not be completed until after mid-February (too late for RC schedule). There will be a corrective action to evaluate results to determine if FE model needs to be updated.

What testing was done on core samples to obtain measured data on the rebar/concrete bond strength for the shield building? None. Have core sample from SB at PII which "nicked" a portion of the rebar and photographs from construction of the access opening which will suffice to evaluate this issue.

Did your vendor request test samples to rule out bond/adhesion issues? No.

Any other core sample tests needed for root cause? No.

Status of Your Team/Contractor Work Products

PII- FMEA-. Preliminary version was due (Jan 25th) did you receive it? Yes. Portions of the FMEA were received and reviewed by FENOC (those which eliminated the less likely causes).

Finite Element Model of the Containment Shield Building (Abaqus Code)- (Jan 30th)? Yes, we are still on track to receive this.

Any more information on Benchmarking ABAQUS software? (e.g. run ABAQUS against another software model to validate results). Plan to run Davis-Besse model and compare with Crystal River containment model results but both are ABAQUS models. No other benchmarking plans and results of this comparison are expected to be documented by the vendor in the RCR.

B/53

Root Cause Report- Explain internal review process once team lead approves it and best guess timeline for these reviews. MPR staff and Vatic Associates will review draft PII report and root cause team lead will incorporate/resolve third party review comments. Once team lead and site managers approve draft report it will go to the Corrective Action Review Board (CARB) and then site VP for final review and approval.

Will you meet February 28th issue and submission date identified in the NRC CAL? Yes.

It is our understanding that the results of the vendor shield building FE modeling done in support of your root cause effort will not be used to validate or be referenced in support of site analysis/calculations that confirm the operability or functionality of the shield building (with cracks). Is our understanding correct? Yes. Because this FE model will not be considered or used in a design calculation it does not need to be under an Appendix B QA program.

Further Explanation (John Hook- Root Cause Team Sponsor) - The FE model will provide stress numbers in the SB at specific locations but actual number is not as important as magnitude of result. For example if model predicts 1500 psi tensile load at a known crack location, this is well above concrete tensile capacity so it validates basis for cracking. In fact, one of the more likely causes at this point is the FE model predictions of stresses in the SB based upon the blizzard of 1978. This blizzard produced sustained winds above 80mph (gusts over 100mph) and had very cold temperatures (high thermal stress with plant on-line). The preliminary FE modeling suggests that these loads combined with the lack of radial hooks in the shoulder regions (areas with extensive cracking) combined to allow tensile stresses well above concrete tensile capacity. Normal winter wind and thermal loads do not approach the magnitude of stress developed during this storm.

Will the final root cause result be used in an updated OE? Yes, the corrective action exists to update the OE with cause of SB cracking.

Will root cause report results be used to validate the adequacy of site programs for managing the aging effects of safety related structures? Yes. The site did not develop the FE or root cause under Appendix B controls but intends to use the result to ensure that they have an adequate structures monitoring program for license renewal aging management.

<u>Status of Purdue Univ Testing (not Used by RCT)</u>-

Background: To evaluate the excess capacity or the bond strength of Lap splice configuration for DB SB rebar configurations. Dr. Sozen of Purdue (concrete expert) and Dr. Darwin Kansas St Univ (Bond strength expert) will collaborate under Bechtel direction to perform test. Licensee will fabricate a beam section with concrete and rebar that matches 28 day strength requirements for DB SB. Beam will be placed in bending until failure by laminar cracking. Then additional tensile capacity of rebar w cracked concrete will be measured. Suspect 30-50%

capacity with this configuration. Result will be used to support a use-as-is disposition for the existing concrete cracking configuration and will not be needed for root cause efforts.

Is this testing going to be conducted under a vendor (Bechtel) or site QA approved Appendix B program? Undecided at this point. Schedule for this testing? Not yet developed.

Power Reactor	Event Number: 46362
Facility: TURKEY POINT Region: 2 State: FL Unit: [3] [ ] [ ] RX Type: [3] W-3-LP,[4] W-3-LP NRC Notified By: KEITH MAESTAS HQ OPS Officer: VINCE KLCO	Notification Date: 10/25/2010 Notification Time: 22:30 [ET] Event Date: 10/25/2010 Event Time: 22:00 [EDT] Last Update Date: 10/25/2010
Emergency Class: NON EMERGENCY 10 CFR Section: 50.72(b)(3)(ii)(A) - DEGRADED CONDITION	Person (Organization): RANDY MUSSER (R2DO)

Unit	SCRAM Code	RX CRIT	Initial PWR	Initial RX Mode	Current PWR	Current RX Mode
3	N	N	0	Cold Shutdown	0	Cold Shutdown

Event Text

NTAINMENT LINER CORROSION DEGRADATION

his report is made in accordance with 10 CFR 50.72(b)(3)(ii) as a condition resulting in a serious degradation of the Containment liner. During a visual inspection on /22/10, corrosion was found in the containment sump liner on Unit 3, A more detailed inspection was performed on 10/24/10 after the corroded area was cleaned. aluation of the inspection results has determined that the corrosion is greater than allowed by the ASME Code, including through wall areas, and requires repair. At this me, it is not known whether any leakage caused by the through-wall condition would have resulted in exceeding the containment allowed leakage limit. Unit 3 is currentl Mode 5 preparing to return to service following refueling. The corroded areas will be repaired prior to entering Mode 4."

e licensee notified the NRC Resident Inspector

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
OFFICE OF NEW REACTORS
WASHINGTON, DC 20555-0001

June 18, 2010

NRC INFORMATION NOTICE 2010-12: CONTAINMENT LINER CORROSION

ADDRESSEES

All holders of an operating license or construction permit for a nuclear power reactor issued under Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities," except those who have permanently ceased operations and have certified that fuel has been permanently removed from the reactor vessel.

All holders of or applicants for a standard design certification, standard design approval, manufacturing license, or combined license issued under 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants."

PURPOSE

The U.S. Nuclear Regulatory Commission (NRC) is issuing this information notice (IN) to inform addressees of recent issues involving corrosion of the steel reactor containment building liner. The NRC expects recipients to review the information for applicability to their facilities and to consider actions, as appropriate, to avoid similar problems. The suggestions contained in this IN are not NRC requirements; therefore, no specific action or written response is required.

DESCRIPTION OF CIRCUMSTANCES

Beaver Valley Power Station

On April 23, 2009, during a refueling outage at Beaver Valley Power Station, Unit 1, the licensee performed a visual examination of the interior reactor containment building steel liner in accordance with Subsection IWE, "General Visual Examination," of American Society of Mechanical Engineers (ASME) Section XI of the Boiler and Pressure Vessel Code, "Rules for Inservice Inspection of Nuclear Power Plant Components." At a containment elevation of 746 feet, the licensee identified an area approximately 3 inches in diameter that exhibited blistered paint. The paint blister was intact at the time of discovery. Collapse of the blister during further inspection revealed a protruding rust product underneath. The licensee then cleaned this area to allow further evaluation. The cleaning activity uncovered a rectangular area of approximately 1 inch (horizontal) x 3/8 inch (vertical) that penetrated through the entire liner plate thickness. Ultrasonic testing (UT) of the surrounding area showed liner thinning within an area of approximately 10 square inches. The licensee removed the corroded section of the liner and discovered a partially decomposed piece of wood approximately 2 inches x 4 inches x 6 inches embedded in the concrete behind the section of the liner. The wood was left behind as

ML100640449

a result of inadequate housekeeping and quality assurance practices during the original construction of the containment wall in the early 1970s.

The licensee determined that the cause of the through-wall liner corrosion was a pitting-type corrosion (rust) originating from the concrete side caused by foreign material (wood) that was in contact with the containment carbon steel liner. Licensee corrective actions included removing the embedded wood, grouting the concrete area that was displaced by the wooden debris, and welding a new section of steel plate to replace the previously removed portion of the liner. The licensee also scheduled an examination of the containment liner during the next refueling outage at Beaver Valley Power Station Units, 1 and 2 to visually inspect 100 percent of the accessible liner area. In addition, the licensee stated in its license renewal submittals that it would perform: (a) supplemental volumetric examinations of 1-square-foot samples in at least 75 random locations of each unit's containment liner in order to statistically determine whether the containment liner is unacceptably degraded by corrosion originating from the concrete side; and (b) supplemental volumetric examinations of a minimum of eight one-foot square locations in the accessible areas of liner plate at locations that operating experience shows are susceptible to localized pitting corrosion.

Additional information is available in Beaver Valley Licensee Event Report 50-334/2009-003-00, dated June 18, 2009, and Beaver Valley Power Station, Unit 1, NRC Routine Inspection Report 05000334/2009006, dated July 6, 2009, which can be found on the NRC's public Web site under Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML091740056 and ML091870328, respectively.

Brunswick Steam Electric Plant

During a refueling outage in 2008 at Brunswick Steam Electric Plant, Unit 1, the licensee performed a VT-1 visual inspection of the primary containment penetration sleeve for the personnel air lock and found two bulged areas. The discovery of thinned areas on the bulges led the licensee to perform UT examinations of the entire Unit 1 personnel air lock penetration sleeve. These additional UT inspections identified many discrete locations that were below the minimum wall thickness established by the design-basis containment liner specification.

During construction, the outside diameter of the sleeve was wrapped with two layers of 1/4 inch felt and the felt was covered with a layer of 60 mil ethylene propylene film. The felt was intended to permit the sleeve to expand when subjected to thermal loading. The licensee's evaluation determined that the bulges were caused by corrosion product buildup between the sleeve and the concrete backing. This corrosion was caused by the felt that wrapped the outside of the containment penetration sleeve; which became wet during the original construction.

Samples of the degraded areas of the sleeve and felt wrapping were sent to the licensee's center for evaluation of potential ongoing corrosion mechanisms. These evaluations identified that the pitting and corrosion on the concrete side of the sleeve were caused by under-deposit corrosion.

Licensee corrective actions include installing a new concentric sleeve inside the personnel air lock penetration to repair the existing containment penetration sleeve. Following this planned modification, the new sleeve will become the primary containment liner for this penetration.

Salem Nuclear Generating Station

In October 2009, at Salem Nuclear Generating Station Unit 2, the licensee inspected the containment moisture barrier (the silicone RTV [room temperature vulcanizing] seal between the concrete floor and containment liner) and found heavy corrosion on the containment liner within 6 inches of the concrete floor. This area of the containment liner was considered inaccessible because it was normally covered by an insulation package that consisted of a layer of sheet metal, a layer of plastic sheeting, and a layer of insulation. The licensee had not inspected the containment liner areas covered by this insulation because ASME Code Section XI allowed an exemption for inaccessible areas. In response to this discovery, and as a conservative approach to the license renewal process, the licensee decided to enhance inspections of the containment liner above the moisture barrier within about 6 inches of the concrete floor and to randomly inspect several other areas that were covered by the insulation package. To perform the inspections, the licensee removed that portion of the insulation package that extended below the lower leak detection channel for the entire containment liner circumference, and cut through and removed the insulation package for four other randomly selected areas. Licensee inspections in these four areas identified some corrosion but subsequent ultrasonic measurements did not indicate significant wall loss.

To evaluate the effect of the identified general corrosion on the safety function of the containment boundary and to meet the expanded inspection requirements of ASME Code Section XI, the licensee performed ultrasonic testing of 440 locations on the bottom 6 inches of the cylindrical portion of the containment liner. Based on the results of the measurements at these locations, the licensee determined that the liner remained operable because the lowest thickness measured was 0.677 inches, which was above the design-required minimum wall thickness of 0.43 inches. The actual safety significance of this general corrosion was minor because there was significant design margin for the liner in this area.

The licensee reviewed the circumstances that led to the identified areas of heavy corrosion on the liner and determined that previous containment liner inspections were not performed adequately. Specifically, examinations should have identified evidence of corrosion (rust on floor) and prompted removal of lagging to determine the source of the corrosion products.

The licensee determined that the source of the moisture that caused the liner corrosion at the joint between the containment liner and concrete floor was service water leakage from the containment fan coil units and associated piping. Licensee corrective actions included conducting frequent containment walk-downs to identify, isolate, and repair any identified service water leaks; verifying that the leakage from existing service water leaks did not reach the containment liner; and, until the base of the containment liner is re-coated during a future refueling outage, revising procedures to ensure liner inspections were performed when containment service water leaks were identified. In addition, the licensee stated in its license renewal submittals that it would perform supplemental and augmented examinations of the liner plates at random and non-random locations.

BACKGROUND

Related NRC communications include the following:

- IN 2004-09, “Corrosion of Steel Containment and Containment Liner,” dated April 27, 2004 (ADAMS Accession No. ML041170030)

- IN 1997-10, “Liner Plate Corrosion in Concrete Containments,” dated March 13, 1997 (ADAMS Accession No. ML031050365)

DISCUSSION

This IN provides examples of containment liner degradation caused by corrosion. Concrete reactor containments are typically lined with a carbon steel liner to ensure a high degree of leak tightness during operating and accident conditions. The reactor containment is required to be operable as specified in plant technical specifications to limit the leakage of fission product radioactivity from the containment to the environment. The regulations at 10 CFR 50.55a, “Codes and Standards,” require the use of Subsection IWE of ASME Section XI to perform inservice inspections of containment components. The required inservice inspections include periodic visual examinations and limited volumetric examinations using ultrasonic thickness measurements. The containment components include the steel containment liner and integral attachments for the concrete containment, containment personnel airlock and equipment hatch, penetration sleeves, moisture barriers, and pressure-retaining bolting. The NRC also requires licensees to perform leak rate testing of the containment pressure-retaining components and isolation valves according to 10 CFR Part 50, Appendix J, “Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors,” as specified in plant technical specifications. This operating experience highlights the importance of good quality assurance, housekeeping and high quality construction practices during construction operations in accordance with 10 CFR Part 50, Appendix B, “Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants.”

Operating experience shows that containment liner corrosion is often the result of liner plates being in contact with objects and materials that are lodged between or embedded in the containment concrete. Liner locations that are in contact with objects made of an organic material are susceptible to accelerated corrosion because organic materials can trap water that combined with oxygen will promote carbon steel corrosion. Organic materials can also cause a localized low pH area when they decompose. Organic materials located inside containment can come in contact with the containment liner and cause accelerated corrosion. However, corrosion that originates between the liner plate and concrete is a greater concern because visual examinations typically identify the corrosion only after it has significantly degraded the liner. In some cases, licensees identified such corroded areas by performing ultrasonic examination of suspect areas (e.g., areas of obvious bulging, hollow sound).

The objects and materials that caused liner corrosion that licensees have found lodged between or embedded in the containment concrete include both foreign material (e.g., wooden pieces, workers’ gloves, wire brush handles) and material that was deliberately installed as part of the design such as the felt material described in the above example at Brunswick Steam Electric

Plant, Unit 1. Although there is no regulatory requirement to do so, one or more licensees have chosen to review design documents to identify locations where organic material was intentionally installed between the liner or penetration sleeve and schedule additional examinations of these areas to monitor for liner material loss.

GENERIC IMPLICATIONS

In response to the above through-wall corrosion at Beaver Valley Power Station, Unit 1, the NRC Office of Nuclear Reactor Regulation requested the NRC Office of Regulatory Research to begin an assessment to better understand the possible mechanisms responsible for through-wall corrosion of containment liners. The NRC staff has also engaged committee members for ASME Section XI to devise a formal tracking mechanism to monitor industry experience and events involving containment liner corrosion. Subsection IWE of ASME Section XI could then be updated using insights from these events.

CONTACT

This IN requires no specific action or written response. Please direct any questions about this matter to the technical contacts listed below or to the appropriate Office of Nuclear Reactor Regulation (NRR) project manager.

/RA/

Timothy McGinty, Director
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

/RA by MShuaibi for/

Glenn Tracy, Director
Division of Construction Inspection and
Operational Programs
Office of New Reactors

Technical Contacts:

Mike Brown, NRR
301-415-4096
E-mail: Michael.Brown@nrc.gov

William Jessup, NRR
301-415-2972
E-mail: William.Jessup@nrc.gov

Paul Klein, NRR
301-415-4030
E-mail: Paul.Klein@nrc.gov

Note: NRC generic communications may be found on the NRC public Web site, http://www.nrc.gov, under Electronic Reading Room/Document Collections.

Plant, Unit 1. Although there is no regulatory requirement to do so, one or more licensees have chosen to review design documents to identify locations where organic material was intentionally installed between the liner or penetration sleeve and schedule additional examinations of these areas to monitor for liner material loss.

GENERIC IMPLICATIONS

In response to the above through-wall corrosion at Beaver Valley Power Station, Unit 1, the NRC Office of Nuclear Reactor Regulation requested the NRC Office of Regulatory Research to begin an assessment to better understand the possible mechanisms responsible for through-wall corrosion of containment liners. The NRC staff has also engaged committee members for ASME Section XI to devise a formal tracking mechanism to monitor industry experience and events involving containment liner corrosion. Subsection IWE of ASME Section XI could then be updated using insights from these events.

CONTACT

This IN requires no specific action or written response. Please direct any questions about this matter to the technical contacts listed below or to the appropriate Office of Nuclear Reactor Regulation (NRR) project manager.

/RA/

Timothy McGinty, Director
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

/RA by MShuaibi for/

Glenn Tracy, Director
Division of Construction Inspection and
Operational Programs
Office of New Reactors

Technical Contacts: Mike Brown, NRR
301-415-4096
E-mail: Michael.Brown@nrc.gov

William Jessup, NRR
301-415-2972
E-mail: William.Jessup@nrc.gov

Paul Klein, NRR
301-415-4030
E-mail: Paul.Klein@nrc.gov

Note: NRC generic communications may be found on the NRC public Web site, http://www.nrc.gov, under Electronic Reading Room/Document Collections.

ADAMS Accession No.: ML100640449 TAC ME3101

OFFICE	IOEB:DIRS	SCVB:DSS	Tech Editor	BC:SCVB:DSS	D:DSS
NAME	MBrown	JBettle	KAzariah-Kribbs	RDennig	WRuland
DATE	03/30/10	05/24/10 e-mail	05/21/10 e-mail	05/24/10 e-mail	5/27/10 e-mail
OFFICE	BC:EMCB:DE	BC:CSGB:DCI	BC:SPCV:DSRA	BC:RASB:DLR	BC:IOEB:DIRS
NAME	MKhanna	RTaylor	JMcKirgan	RAuluck	JThorp
DATE	5/25/2010 e-mail	5/25/10 e-mail	5/27/10 e-mail	6/9/10 e-mail	5/25/10 e-mail
OFFICE	LA:PGCB:NRR	PM:PGCB:NRR	(A)BC:PGCB:NRR	D:DCIP:NRO	D:DPR:NRR
NAME	CHawes	DBeaulieu	TAlexion	GTracy(MShuaibi for)	TMcGinty
OFFICE	6/10/10	5/27/10	6/10/10	7/17/10	6/18/10

OFFICIAL RECORD COPY