EXHIBIT F

From: <apolonsky@morganlewis.com>
To: <rwebster@kinoy.rutgers.edu>
Date: 3/23/06 10:00AM
Subject: Oyster Creek

Richard,
You asked me whether there is a discrepancy between a "Fact Sheet" that discusses the upper region of the drywell at the Oyster Creek Nuclear Generating Station, and a sentence in AmerGen's December 12, 2005, Answer to your clients' Petition.

The Fact Sheet that you faxed to me states that, "[t]here is no additional corrosion at two of the four regions in the upper region of the drywell that previously experienced corrosion. Corrosion at the other two elevations in the upper region is very minor, continues to decrease and would not impact the structural integrity of the drywell through 2029." These statements are consistent with AmerGen's License Renewal Application, which states that "corrosion in the sand bed region has been arrested and no further loss of material is expected" (Application at 3.5-20), and "recent UT measurements (2004) [in the upper region] confirmed that the corrosion rate continues to decline" (Application at 3.5-21).

The sentence at issue is on page 21 of AmerGen's Answer. It states: "Based on these measurements and inspections, AmerGen concluded that corrosion of the drywell shell has been arrested, including in the sand bed region. Application at 3.5-20 to -21." Your concern with this sentence is that it suggests that corrosion in the upper region of the drywell has been arrested.

We agree with you that the sentence in the Answer could cause confusion. The word "including" should be deleted from the sentence at issue. The Board, however, did not rely on the condition of the drywell shell in the upper region when it rejected that portion of the contention. See Memorandum and Order at 33 n.27 ("We limit NIRS's contention to the sand bed region because, contrary to NIRS's assertion, AmerGen is performing, and will continue to perform during the renewal period, UT measurements at critical locations in the upper region of the drywell liner"). In addition, all parties had a copy of the Application which stated that corrosion in the upper region "continues to decline."

In any event, to ensure that there is no confusion in the record, we will be notifying the Board, the Commission, and the parties.

We thank you for bringing this issue to our attention and hope that this clarification is helpful.

Alex S. Polonsky
Morgan Lewis & Bockius LLP
1111 Pennsylvania Ave., NW
Washington, DC 20004
Direct dial: 202.739.5830
Fax: 202.739.3001

CC: <ksutton@morganlewis.com>, <dsilverman@morganlewis.com>, <Bradley.Fewell@exeloncorp.com>

EXHIBIT G

----- Original Message -----
From: Paul Gunter
To: gburl@comcast.net
Sent: Tuesday, September 06, 2005 3:44 PM
Subject: resend of request to AmerGen on Oyster Creek inspections

TO: Jhansi Kandasamy, Oyster Creek Site Regulatory Assurance Manager
FROM: Paul Gunter, Nuclear Information and Resource Service and
Edith Gbur, Jersey Shore Nuclear Watch
RE: Oyster Creek Drywell Liner Corrosion Monitoring Program

Dear Ms. Kandasamy:

Per request of Nuclear Information and Resource Service and Jersey Shore Nuclear Watch dated July 19, 2005 to New Jersey Department of Environmental Protection Commissioner Bradley Campbell and Commissioner Campbell's reply dated August 25, 2005, we are requesting information from AmerGen with regard to the Oyster Creek Drywell Liner Corrosion Monitoring Program.

Requested Documents and Related Questions:

1) Please provide the1996, 2000, 2004 inspection results of Ultrasonic Tests (UT) at all locations on the Drywell Liner including the sand bed region.
2) If no UT inspections were conducted at the sand bed region after 1994, how is the performance of the coating at the sand bed region determined to be effective in arresting corrosion in this region? Please provide the original documentation of this analysis.
3) Has the base of the Drywell Liner below the sand bed region to 0° ever been inspected by UT or any other means? If so, what were the results? If not, why not?
4) Please provide GPUN's original analysis for the revision of Technical Specification 5.2.A which reduced the Drywell Liner pressure rating from 62 psig to 44 psig and raised its temperature rating from 175 ° F to 292° F.
5) Is there any evidence of corrosive pitting in the Drywell Liner? If so, what inspection methods are being employed to assess the effects of corrosive pitting on the structural integrity of the Drywell Liner?
6) Is there any evidence of cracking in the Drywell Liner? If so, please make all Condition Reports and/or other documents regarding any cracks in the liner available for review.
7) If cracking has occurred, please provide the analysis of the root cause.
8) Have welding repairs been made to the Drywell Liner for any reason? If so, please provide the results of those repairs.

Thank you,
Paul Gunter, Director
Reactor Watchdog Project
Nuclear Information and Resource Service
1424 16th Street NW Suite 404
Washington, DC 20036
Tel. 202 328 0002

EXHIBIT H

From: pete.resler@exeloncorp.com [mailto:pete.resler@exeloncorp.com]
Sent: Monday, October 10, 2005 4:22 PM
To: gbur1@comcast.net; Paul Gunter
Cc: CommissionerCampbell@dep.state.nj.us
Subject: Response to 9/6 request for information

Ms. Gbur, Mr. Gunter:

In response to your request for information dated Sept. 6, 2005, concerning the Oyster Creek drywell liner, AmerGen will not provide proprietary business information. The results of equipment testing, analysis and other operational and regulatory documentation are available at the station to the U.S. Nuclear Regulatory Commission and the New Jersey Bureau of Nuclear Engineering for review at any time.

Much of the information you have requested is available to the public in the Oyster Creek license renewal application available on the NRC web site, as well as in a summary of this issue that was provided to the NJ BNE. I have attached that document below for your information.

In addition, the NRC approved the initial analysis and corrective actions taken after corrosion was discovered in 1980, as well as the ongoing inspection and evaluation program to ensure the corrective actions continue to be effective. Regular inspections and analyses of the drywell liner have confirmed that corrosion is managed effectively and that the drywell liner can perform its intended function.

<<drywell corrosion issue summary.doc>>

Peter C. Resler
Manager, Nuclear Communications
Exelon Corporation
610-765-5530

EXHIBIT I

From: <George.Beck@exeloncorp.com>
To: <dja1@nrc.gov>, <rkm@nrc.gov>
Date: 04/05/2006 5:02:53 PM
Subject: FW: Audit Q & A (Question Numbers AMP-141, 210, 356)

Note: As originally transmitted this email was undeliverable to the NRC; it exceeded the size limit. It is being retransmitted without the AMP-210.pdf. This file will be reconstituted and sent in smaller ".pdf"s; the first 11 pages are attached.

George

> -----Original Message-----
> From: Beck, George
> Sent: Wednesday, April 05, 2006 4:39 PM
> To: Donnie Ashley (E-mail); 'Roy Mathew (E-mail) ' (E-mail)
> Cc: Ouaou, Ahmed; Hufnagel Jr, John G; Warfel Sr, Donald B; Polaski, Frederick W
> Subject: Audit Q & A (Question Numbers AMP-141, 210, 356)
>
> Donnie/Roy,
>
> Attached are the responses to AMP-210 and AMP-356 in an updated version of the reports from the AMP/AMR Audit database. Also included is a revised version of AMP-141. These answers have been reviewed and approved by Technical Lead, Don Warfel.
>
> Regarding AMP-210, please note:
> As pointed out in our response to NRC Question AMP-210, (8a)(1), "The 0.806" minimum average thickness verbally discussed with the Staff during the AMP audit was recorded in location 19A in 1994. Additional reviews after the audit noted that lower minimum average thickness values were recorded at the same location in 1991 (0.803") and in September 1992 (0.800"). However, the three values are within the tolerance of +/- 0.010" discussed with the Staff."
>
> Regarding AMP-141, please note:
> Our response to AMP-141 has been revised to reflect additional information developed during the ongoing preparation of RAI responses.
>
> Please let John Hufnagel or me know if you have any questions.
>
> George
>
>
>
> >> <<Pages from AMP-210.pdf>>
> >> <<AMP-141.pdf>>
>
>> <<AMP-356.pdf>>
>
>

CC: <ahmed.ouaou@exeloncorp.com>, <john.hufnagel@exeloncorp.com>, <donald.warfel@exeloncorp.com>, <fred.polaski@exeloncorp.com>

Mail Envelope Properties (44343066.C5F : 19 : 7263)

Subject:	FW: Audit Q & A (Question Numbers AMP-141, 210, 356)
Creation Date:	04/05/2006 5:01:46 PM
From:	<George.Beck@exeloncorp.com>
Created By:	George.Beck@exeloncorp.com

Recipients

nrc.gov
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MESSAGE	2679	05 April, 2006 5:01:46 PM
TEXT.htm	5457	
Pages from AMP-210.pdf	64593	
AMP-141.pdf	47353	
AMP-356.pdf	71556	
Mime.822	262768	

Options

Expiration Date:	None
Priority:	Standard
Reply Requested:	No
Return Notification:	None
Concealed Subject:	No
Security:	Standard

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Item No		*Date Received:*	*Source*
AMP-210		1/24/2006	AMP Audit
Topic:		*Status:*	Open
IWE			

Document References:
B.1.27

NRC Representative Morante, Rich

AmerGen (Took Issue): Hufnagel, Joh

Question

Pages 25 through 31 of the PBD present a discussion of the OCGS operating experience.

(8a)The following statements related to drywell corrosion in the sand bed region need further explanation and clarification:
As a result of the presence of water in the sand bed region, extensive UT thickness measurements (about 1000) of the drywell shell were taken to determine if degradation was occurring. These measurements corresponded to known water leaks and indicated that wall thinning had occurred in this region.
Please explain the underlined statement. Were water leaks limited to only a portion of the circumference? Was wall thinning found only in these areas?
After sand removal, the concrete surface below the sand was found to be unfinished with improper provisions for water drainage. Corrective actions taken in this region during 1992 included; (1) cleaning of loose rust from the drywell shell, followed by application of epoxy coating and (2) removing the loose debris from the concrete floor followed by rebuilding and reshaping the floor with epoxy to allow drainage of any water that may leak into the region. UT measurements taken from the outside after cleaning verified loss of material projections that had been made based on measurements taken from the inside of the drywell. There were, however, some areas thinner than projected; but in all cases engineering analysis determined that the drywell shell thickness satisfied ASME code requirements.
Please describe the concrete surface below the sand that is discussed in paragraph above.
Please provide the following information:
(1) Identify the minimum recorded thickness in the sand bed region from the outside inspection, and the minimum recorded thickness in the sand bed region from the inside inspections. Is this consistent with previous information provided verbally? (.806 minimum)
(2) What was the projected thickness based on measurements taken from the inside?
(3) Describe the engineering analysis that determined satisfaction of ASME code requirements and identify the minimum required thickness value. Is this consistent with previous information provided verbally? (.733 minimum)
(4) Is the minimum required thickness based on stress or buckling criteria?
(5) Reconcile and compare the thickness measurements provided in (1) and (3) above with the .736 minimum corroded thickness that was used in the NUREG-1540 analysis of the degraded Oyster

Creek sand bed region.

Evaluation of UT measurements taken from inside the drywell, in the in the former sand bed region, in 1992, 1994, and 1996 confirmed that corrosion is mitigated. It is therefore concluded that corrosion in the sand bed region has been arrested and no further loss of material is expected. Monitoring of the coating in accordance with the Protective Coating Monitoring and Maintenance Program, will continue to ensure that the containment drywell shell maintains its intended function during the period of extended operation.
NUREG-1540, published in April 1996, includes the following statements related to corrosion of the Oyster Creek sand bed region: (page vii) However, to assure that these measures are effective, the licensee is required to perform periodic UT measurements. and (page 2) As assurance that the corrosion rate is slower than the rate obtained from previous measurements, GPU is committed to make UT measurements periodically. Please reconcile the aging management commitment (one-time UT inspection and monitoring of the condition of the coating) with the apparent requirement/commitment documented in NUREG-1540.

(8b)The following statement related to drywell corrosion above the sand bed region needs further explanation and clarification:

Corrective action for these regions involved providing a corrosion allowance by demonstrating, through analysis, that the original drywell design pressure was conservative. Amendment 165 to the Oyster Creek Technical Specifications reduced the drywell design pressure from 62 psig to 44 psig. The new design pressure coupled with measures to prevent water intrusion into the gap between the drywell shell and the concrete will allow the upper portion of the drywell to meet ASME code requirements.

Please describe the measures to prevent water intrusion into the gap between the drywell shell and the concrete that will allow the upper portion of the drywell to meet ASME code requirements". Are these measures to prevent water intrusion credited for LR? If not, how will ASME code requirements be met during the extended period of operation?

(8c)The following statements related to torus degradation need further explanation and clarification:
Inspection performed in 2002 found the coating to be in good condition in the vapor area of the Torus and vent header, and in fair condition in immersion. Coating deficiencies in immersion include blistering, random and mechanical damage. Blistering occurs primarily in the shell invert but was also noted on the upper shell near the water line. The fractured blisters were repaired to reestablish the protective coating barrier. This is another example of objective evidence that the Oyster Creek ASME Section XI, Subsection IWE aging management program can identify degradation and implement corrective actions to prevent the loss of the containment's intended function.
While blistering is considered a deficiency, it is significant only when it is fractured and exposes the base metal to corrosion attack. The majority of the blisters remain intact and continues to protect the base metal; consequently the corrosion rates are low. Qualitative assessment of the identified pits indicate that the measured pit depths (50 mils max) are significantly less than the criteria established

in Specification SP-1302-52-120 (141- 261 mils, depending on diameter of the pit and spacing between pits).

Please confirm or clarify (1) that only the fractured blisters found in this inspection were repaired; (2) pits were identified where the blisters were fractured; (3) pit depths were measured and found to 50 mils max; (4) the inspection Specification SP-1302-52-120 includes pit-depth acceptance criteria for rapid evaluation of observed pitting; (5) the minimum pit depth of concern is 141 mils (.141) and pits as deep as 261 mils (.261) may be acceptable.

Please also provide the following information: nominal design, as-built, and minimum measured thickness of the torus; minimum thickness required to meet ASME code acceptance criteria; the technical basis for the pitting acceptance criteria include in Specification SP-1302-52-120

Assigned To: Ouaou, Ahmed

Response:

(8a) Question: Please explain the underlined statement. Were water leaks limited to only a portion of the circumference? Was wall thinning only in these area?

Response:
This statement was not meant to indicate that water leaks were limited to only a portion of the circumference. The statement is meant to reflect the fact that water leakage was observed coming out of certain sand bed region drains and those locations were suspect of wall thinning.
No. Wall thinning was not limited to the areas where water leakage from the drains was observed. Wall thinning occurred in all areas of the sand bed region based on UT measurements and visual inspection of the area conducted after the sand was removed in 1992. However the degree of wall thinning varied from location to location. For example 60% of the measured locations in the sand bed region (bays 1, 3, 5, 7, 9, and 15) indicate that the average measured drywell shell thickness is nearly the same as the design nominal thickness and that these locations experienced negligible wall thinning; whereas bay 19A experienced approximately 30% reduction in wall thickness.

Question: Please discuss the concrete surface below the sand that is discussed in paragraph above.

Response:
The concrete surface below the sand was intended to be shaped to promote flow toward each of the five sand bed drains. However once the sand was removed it was discovered that the floor was not properly finished and shaped as required to permit proper drainage. There were low points, craters, and rough surfaces that could allow moisture to pool instead of flowing smoothly toward the drains. These concrete surfaces were refurbished to fill low areas, smooth rough surfaces, and coat these surfaces with epoxy coating to promote improved drainage. The drywell shell at juncture of the concrete floor was sealed with an elastomer to prevent water intrusion into the embedded drywell shell.

Question: Please provide the following information:

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(1) Identify the minimum recorded thickness in the sand bed region from the outside inspection, and the minimum recorded thickness in the sand bed region from the inside inspections. Is this consistent with previous information provided verbally? (.806 minimum)
(2) What was the projected thickness based on measurements taken from the inside?
(3) Describe the engineering analysis that determined satisfaction of ASME code requirements and identify the minimum required thickness value. Is this consistent with previous information provided verbally? (.733 minimum)
(4) Is the minimum required thickness based on stress or buckling criteria?
(5) Reconcile and compare the thickness measurements provided in (1) and (3) above with the .736 minimum corroded thickness that was used in the NUREG-1540 analysis of the degraded Oyster Creek sand bed region.

Response:
1. The minimum recorded thickness in the sand bed region from outside inspection is 0.618 inches. The minimum recorded thickness in the sand bed region from inside inspections is 0.603. These minimum recorded thicknesses are isolated local measurement and represent a single point UT measurement. The 0.806 inches thickness provided to the Staff verbally is an average minimum general thickness calculated based on 49 UT measurements taken in an area that is approximately 6"x 6". Thus the two local isolated minimum recorded thicknesses cannot be compared directly to the general thickness of 0.806".

The 0.806" minimum average thickness verbally discussed with the Staff during the AMP audit was recorded in location 19A in 1994. Additional reviews after the audit noted that lower minimum average thickness values were recorded at the same location in 1991 (0.803") and in September 1992 (0.800"). However, the three values are within the tolerance of +/- 0.010" discussed with the Staff.

2. The minimum projected thickness depends on whether the trended data is before or after 1992 as demonstrated by corrosion trends provided in response to NRC Question #AMP-356. For license renewal, using corrosion rate trends after 1992 is appropriate because of corrosion mitigating measures such as removal of the sand and coating of the shell. Then, using corrosion rate trends based on 1992, 1994, and 1996 UT data; and the minimum average thickness measured in 1992 (0.800"), the minimum projected average thickness through 2009 and beyond remains approximately 0.800 inches. The projected minimum thickness during and through the period of extended operation will be reevaluated after UT inspections that will be conducted prior to entering the period of extended operation, and after the periodic UT inspection every 10 years thereafter.

3.The engineering analysis that demonstrated compliance to ASME code requirements was performed in two parts, Stress and Stability Analysis with Sand, and Stress and Stability Analyses without Sand. The analyses are documented in GE Reports Index No. 9-1, 9-2, 9-3, and 9-4, were transmitted to the NRC Staff in December 1990 and in 1991 respectively. Index No. 9-3 and 9-4, were revised later to correct errors identified during an internal audit and were resubmitted to the Staff in January 1992 (see attachment 1 & 2). The analyses are briefly described below.

The drywell shell thickness in the sand bed region is based on Stability Analysis without Sand. As described in detail in attachment 1 & 2, the analysis is based on a 36-degree section model that takes advantage of symmetry of the drywell with 10 vents. The model includes the drywell shell from the base of the sand bed region to the top of elliptical head and the vent and vent header. The torus is not included in this model because the bellows provide a very flexible connection, which does not allow significant structural interaction between the drywell and the torus. The analysis conservatively assumed that the shell thickness in the entire sand bed region has been reduced uniformly to a thickness of 0.736 inches.

As discussed with the Staff during the AMP audit, the basic approach used in the buckling evaluation follows the methodology outlined in ASME Code Case N-284 revision 0 that was reconciled later with revision 1 of the Code Case. Following the procedure of this Code Case, the allowable compressive stress is evaluated in three steps. In the first step, a theoretical buckling stress is determined, and secondly modified using appropriate capacity and plasticity reduction factors. In the final step, the allowable compressive stress is obtained by dividing the buckling stress calculated in the second step by a safety factor of 2.0 for Design and Level A & B service conditions and 1.67 Level C service conditions.

Using the approach described above, the analysis shows that for the most severe design basis load combinations, the limits of ASME Section III, Subsection NE 3213.10 are fully met. For additional details refer to Attachment 1 & 2.

As described above, the buckling analysis was performed assuming a uniform general thickness of the sand bed region of 0.736 inches. However the UT measurements identified isolated, localized areas where the drywell shell thickness is less than 0.736 inches. Acceptance for these areas was based on engineering calculation C-1302-187-5320-024.

The calculation uses a Local Wall Acceptance Criteria". This criterion can be applied to small areas (less than 12" by 12"), which are less than 0.736" thick so long as the small 12" by 12" area is at least 0.536" thick. However the calculation does not provide additional criteria as to the acceptable distance between multiple small areas. For example, the minimum required linear distances between a 12" by 12" area thinner than 0.736" but thicker than 0.536" and another 12" by 12" area thinner than 0.736" but thicker than 0.536" were not provided.

The actual data for two bays (13 and 1) shows that there are more than one 12" by 12" areas thinner than 0.736" but thicker than 0.536". Also the actual data for two bays shows that there are more than one 2 ½'" diameter areas thinner than 0.736" but thicker than 0.490". Acceptance is based on the following evaluation.

The effect of these very local wall thickness areas on the buckling of the shell requires some discussion of the buckling mechanism in a shell of revolution under an applied axial and lateral pressure load.

To begin the discussion we will describe the buckling of a simply supported cylindrical shell under the influence of lateral pressure and axial load. As described in chapter 11 of the Theory of Elastic Stability, Second Edition, by Timoshenko and Gere, thin cylindrical shells buckle in lobes in both the axial and circumferential directions. These lobes are defined as half wave lengths of sinusoidal functions. The functions are governed by the radius, thickness and length of the cylinder. If we look at a specific thin walled cylindrical shell both the length and radius would be essentially constants and if the thickness was changed locally the change would have to be significant and continuous over a majority of the lobe so that the compressive stress in the lobe would exceed the critical buckling stress under the applied loads, thereby causing the shell to buckle locally. This approach can be easily extrapolated to any shell of revolution that would experience both an axial load and lateral pressure as in the case of the drywell. This local lobe buckling is demonstrated in The GE Letter Report "Sandbed Local Thinning and Raising the Fixity Height Analysis" where a 12 x 12 square inch section of the drywell sand bed region is reduced by 200 mils and a local buckle occurred in the finite element eigenvalue extraction analysis of the drywell. Therefore, to influence the buckling of a shell the very local areas of reduced thickness would have to be contiguous and of the same thickness. This is also consistent with Code Case 284 in Section –1700 which indicates that the average stress values in the shell should be used for calculating the buckling stress. Therefore, an acceptable distance between areas of reduced thickness is not required for an acceptable buckling analysis except that the area of reduced thickness is small enough not to influence a buckling lobe of the shell. The very local areas of thickness are dispersed over a wide area with varying thickness and as such will have a negligible effect on the buckling response of the drywell. In addition, these very local wall areas are centered about the vents, which significantly stiffen the shell. This stiffening effect limits the shell buckling to a point in the shell sand bed region which is located at the midpoint between two vents.

The acceptance criteria for the thickness of 0.49 inches confined to an area less than 2½ inches in diameter experiencing primary membrane + bending stresses is based on ASME B&PV Code, Section III, Subsection NE, Class MC Components, Paragraphs NE-3213.2 Gross Structural Discontinuity, NE-3213.10 Local Primary Membrane Stress, NE-3332.1 Openings not Requiring Reinforcement, NE-3332.2 Required Area of Reinforcement and NE-3335.1 Reinforcement of Multiple Openings. The use of Paragraph NE-3332.1 is limited by the requirements of Paragraphs NE-3213.2 and NE-3213.10. In particular NE-3213.10 limits the meridional distance between openings without reinforcement to 2.5 x (square root of Rt) . Also Paragraph NE-3335.1 only applies to openings in shells that are closer than two times their average diameter.

The implications of these paragraphs are that shell failures at these locations from primary stresses produced by pressure cannot occur provided openings in shells have sufficient reinforcement. The current design pressure of 44 psig for drywell requires a thickness of 0.479 inches in the sand bed region of the drywell. A review of all the UT data presented in Appendix D of the calculation indicates that all thicknesses in the drywell sand bed region exceed the required pressure thickness by a substantial margin. Therefore, the requirements for pressure reinforcement specified in the previous paragraph are not required for the very local wall thickness evaluation presented in Revision 0 of Calculation C-1302-187-5320-024.

Reviewing the stability analyses provided in both the GE Report 9-4 and the GE Letter Report Sand bed Local Thinning and Raising the Fixity Height Analysis and recognizing that the plate elements in the sand bed region of the model are 3" x 3" it is clear that the circumferential buckling lobes for the

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drywell are substantially larger than the 2 ½ inch diameter very local wall areas. This combined with the local reinforcement surrounding these local areas indicates that these areas will have no impact on the buckling margins in the shell. It is also clear from the GE Letter Report that a uniform reduction in thickness of 27% to 0.536" over a one square foot area would only create a 9.5% reduction in the load factor and theoretical buckling stress for the whole drywell resulting in the largest reduction possible. In addition, to the reported result for the 27% reduction in wall thickness, a second buckling analysis was performed for a wall thickness reduction of 13.5% over a one square foot area which only reduced the load factor and theoretical buckling stress by 3.5% for the whole drywell resulting in the largest reduction possible. To bring these results into perspective a review of the NDE reports indicate there are 20 UT measured areas in the whole sand bed region that have thicknesses less than the 0.736 inch thickness used in GE Report 9-4 which cover a conservative total area of 0.68 square feet of the drywell surface with an average thickness of 0.703" or a 4.5% reduction in wall thickness. Therefore, to effectively change the buckling margins on the drywell shell in the sand bed region a reduced thickness would have to cover approximately one square foot of shell area at a location in the shell that is most susceptible to buckling with a reduction in thickness greater than 25%. This leads to the conclusion that the buckling of the shell is unaffected by the distance between the very local wall thicknesses, in fact these local areas could be contiguous provided their total area did not exceed one square foot and their average thickness was greater than the thickness analyzed in the GE Letter Report and provided the methodology of Code Case N284 was employed to determine the allowable buckling load for the drywell. Furthermore, all of these very local wall areas are centered about the vents, which significantly stiffen the shell. This stiffing effect limits the shell buckling to a point in the shell sand bed region, which is located at the midpoint between two vents.

The minimum thickness of 0.733" is not correct. The correct minimum thickness is 0.736".

4. The minimum required thickness for the sand bed region is controlled by buckling.

5. We cannot reconcile the difference between the current (lowest measured) of 0.736" in NUREG-1540 and the minimum measured thickness of 0.806 inches we discussed with the Staff. Perhaps the value in NUREG-1540 should be labeled minimum required by the Code, as documented in several correspondences with the Staff, instead of lowest measured. In a letter dated September 15, 1995, GPU provided the Staff a table that lists sand bed region thicknesses. The table indicates that nominal thickness is 1.154". the minimum measured thickness in 1994 is 0.806", and the minimum thickness required by Code is 0.736". These thicknesses are consistent with those discussed with the Staff during the AMP/AMR audit.

Question: NUREG-1540, published in April 1996, includes the following statements related to corrosion of the Oyster Creek sand bed region: (page vii) However, to assure that these measures are effective, the licensee is required to perform periodic UT measurements. and (page 2) As assurance that the corrosion rate is slower than the rate obtained from previous measurements, GPU is committed to make UT measurements periodically. Please reconcile the aging management commitment (one-time UT inspection and monitoring of the condition of the coating) with the apparent requirement/commitment documented in NUREG-1540.Please reconcile the aging management commitment (one-time UT inspection and monitoring of the condition of the coating) with the apparent requirement/commitment documented in NUREG-1540.

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Response:

Our review of NUREG-1540, page 2 indicates that the statements appear to be based on 1991, or 1993 GPU commitment to perform periodic UT measurements. In fact UT thickness measurements were taken in the sand bed region from inside the drywell in 1992, and 1994. The trend of the UT measurements indicates that corrosion has been arrested. As results GPU informed NRC in a letter dated September 15, 1995 (ref. 2) that UT measurements will be taken one more time, in 1996, and the epoxy coating will be inspected in 1996 and, as a minimum again in 2000. The UT measurements were taken in 1996, per the commitment, and confirmed corrosion rate trend of 1992 and 1994. The results of 1992, 1994, and 1996 UT measurements were provided to the Staff during the AMP/AMR audits.

In response to GPU September 15, 1995 letter, NRC Staff found the proposed changes to sand bed region commitments (i.e. no additional UT measurements after 1996) reasonable and acceptable. This response is documented in November 1, 1995 Safety Evaluation for the Drywell Monitoring Program.

For license renewal, Oyster Creek was previously committed to perform One-Time UT inspection of the drywell shell in the sand bed region prior to entering the period of extended operation. However, in response to NRC Question #AMP-141, Oyster Creek revised the commitment to perform UT inspections periodically. The initial inspection will be conducted prior to entering the period of extended operation and additional inspections will be conducted every 10 years thereafter. The UT measurements will be taken from inside the drywell at same locations as 1996 UT campaign

(8b) Question: Please describe the measures to prevent water intrusion into the gap between the drywell shell and the concrete that will allow the upper portion of the drywell to meet ASME code requirements. Are these measures to prevent water intrusion credited for LR? If not, how will ASME code requirements be met during the extended period of operation?

Response:
The measures taken to prevent water intrusion into the gap between the drywell shell and the concrete that will allow the upper portion of the drywell to maintain the ASME code requirements are,

1. Cleared the former sand bed region drains to improve the drainage path.
2. Replaced reactor cavity steel trough drain gasket, which was found to be leaking.
3. Applied stainless steel type tape and strippable coating to the reactor cavity during refueling outages to seal identified cracks in the stainless steel liner.
4. Confirmed that the reactor cavity concrete trough drains are not clogged
5. Monitored former sand bed region drains and reactor cavity concrete trough drains for leakage during refueling outages and plant operation.

Oyster Creek is committed to implement these measures during the period of extended operation.

(8c) Please confirm or clarify (1) that only the fractured blisters found in this inspection were repaired; (2) pits were identified where the blisters were fractured; (3) pit depths were measured and found to

50 mils max; (4) the inspection Specification SP-1302-52-120 includes pit-depth acceptance criteria for rapid evaluation of observed pitting; (5) the minimum pit depth of concern is 141 mils (.141) and pits as deep as 261 mils (.261) may be acceptable.

Response:
(1) Specification SP-1302-52-120, Specification for Inspection and Localized Repair of the Torus and Vent System Coating, specifies repair requirements for coating defects exposing substrate and fractured blisters showing signs of corrosion. The repairs referred to in the inspection report included fractured blisters, as well as any mechanically damaged areas, which have exposed bare metal showing signs of corrosion. Therefore, only fractured blisters would be candidates for repair, not those blisters that remain intact. The number and location of repairs are tabulated in the final inspection report prepared by Underwater Construction Corporation.
(2) Coating deficiencies in the immersion region included blistering with minor mechanical damage. Blistering occurred primarily in the shell invert but was also noted on the upper shell near the water line. The majority of the blisters were intact. Intact blisters were examined by removing the blister cap exposing the substrate. Corrosion attack under non-fractured blisters was minimal and was generally limited to surface discoloration. Examination of the substrate revealed slight discoloration and pitting with pit depths less than 0.001. Several blistered areas included pitting corrosion where the blisters were fractured. The substrate beneath fractured blisters generally exhibited a slightly heavier magnetite oxide layer and minor pitting (less than 0.010") of the substrate.

(3) In addition to blistering, random deficiencies that exposed base metal were identified in the torus immersion region coating (e.g., minor mechanical damage) during the 19R (2002) torus coating inspections. They ranged in size from 1/16" to ½" in diameter. Pitting in these areas was qualitatively evaluated and ranged from less than 10 mils to slightly more than 40 mils in a few isolated cases. Three quantitative pit depth measurements were taken in several locations in the immersion area of Bay 1. Pit depths at these sites ranged from 0.008" to 0.042" and were judged to be representative of typical conditions found on the shell.

Prior to 2002 inspection 4 pits greater than 0.040" were identified. The pits depth are 0.058" (1 pit in 1988), 0.05" (2 pits in 1991), and 0.0685" (1 pit in 1992). The pits were evaluated against the local pit depth acceptance criteria and found to be acceptable.

(4) Specification SP-1302-52-120, Specification for Inspection and Localized Repair of the Torus and Vent System Coating, includes the pit-depth acceptance criteria for rapid evaluation of observed pitting. The acceptance criteria are supported by a calculation C-1302-187-E310-038. Locations that do not meet the pit-depth acceptance criteria are characterized based on the size of the area, center to center distance between corroded areas, the maximum pit depth and location in the Torus based on major structural features. These details are sent to Oyster Creek Engineering for evaluation.

(5) The acceptance criteria for pit depth is as follows:
-Isolated Pits of 0.125" in diameter have an allowed maximum depth of 0.261" anywhere in the shell provided the center to center distance between the subject pit and neighboring isolated pits or areas of pitting corrosion is greater than 20.0 inches. This includes old pits or old areas of pitting corrosion that have been filled and/or re-coated.

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-Multiple Pits that can be encompassed by a 2-1/2" diameter circle shall be limited to a maximum pit depth of 0.141" provided the center to center distance between the subject pitted area and neighboring isolated pits or areas of pitting corrosion is greater than 20.0 inches. This includes old pits or old areas of pitting corrosion that have been filled and/or recoated.

Question: Please also provide the following information: nominal design, as-built, and minimum measured thickness of the torus; minimum thickness required to meet ASME code acceptance criteria; the technical basis for the pitting acceptance criteria include in Specification SP-1302-52-120

Response:
Submersed area:
(a) The nominal Design thickness is 0.385 inches
(b) The as-built thickness is 0.385 inches
(c) The minimum uniform measured thickness is,
    0.343 inches - general shell
    0.345 inches - shell - ring girders
    0.345 inches - shell - saddle flange
    0.345 inches - shell - torus straps

(d) The minimum general thickness required to meet ASME Code Acceptance is 0.337 inches.

Technical basis for pitting acceptance criteria included in Specification SP-1302-52-120 is based on engineering calculation C-1302-187-E310-038. At the time of preparation of calculation C-1302-187-E310-038 in 2002 there were no published methods to calculate acceptance standards for locally thinned areas in ASME Section III or Section VIII Pressure Vessel codes. Therefore, the approach in Code Case N-597 was used as guidance in assessing locally thinned areas in the Torus. This is based on the similarity in approaches between Local Thinning Areas described in N597 and Local Primary Stress areas described in Paragraph NE3213.10 of the ASME B&PV Code Section III, particularly small areas of wall thinning which do not exceed 1.0 x (square root of Rt). In addition, the ASME B&PV Code Section III, Subsection NB, Paragraph NB-3630 allows the analysis of pipe systems in accordance with the Vessel Analysis rules described in Paragraph NB-3200 of the same Subsection as an alternate analysis approach. Therefore, the approach used in N597 for local areas of thinning was probably developed using the rules for Local Primary Membrane Stress from paragraph NB-3200 in particular Subparagraph 3213.10. The Local Primary Stress Limits in NB-3213.10 are similar to those discussed in Subsection NE, Paragraph NE-3213.10.

Since the Code Case had not yet been invoked in to the Section XI program, the calculation provided a reconciliation of the results obtained from the code case against the ASME Section III code requirements as discussed above. This reconciliation demonstrated that the approach in N597 used on a pressure vessel such as the Torus would be acceptable since the results are conservative compared to the previous work performed in MPR-953 and Lm(a) (defined in N597 Table- 3622-1) £ (Rmintmin)1/2.

Currently, the maximum pit depth measured in the Torus is a 0.0685" (measured in 1992 in bay 2). It was evaluated as acceptable using the design calculations existing at that time and was not based on

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Calculation C-1302-187-E310-038. This remains the bounding wall thickness in the Torus. The criterion developed in 2002 for local thickness acceptance provides an easier method for evaluating as-found pits. The results were shown to be conservative versus the original ASME Section III and VIII Code requirements for the Torus.

The Torus inspection program is being enhanced per IR 373695 to improve the detail of the acceptance criteria and margin management requirements using the ASME Section III criteria. The approach used in C-1302-187-E310-038 will be clarified as to how it maintains the code requirements. If Code Case N-597-1 is required to develop these criteria for future inspections, NRC review and approval will be obtained. It should also be noted that the program has established corrosion rate criteria and continues to periodically monitor to verify they remain bounded.

LRCR #: *LRA A.5 Commitment #:*

IR#:

Approvals:

Prepared By: Ouaou, Ahmed 4/ 5/2006

Reviewed By: Miller, Mark 4/ 5/2006

Approved By: Warfel, Don 4/ 5/2006

NRC Acceptance (Date):

NRC Information Request Form

Item No
AMP-356

Date Received: 2/16/2006

Source AMP Audit

Status: Open

Topic:
IWE

Document References:

NRC Representative Morante, Rich

AmerGen (Took Issue):

Question

IWE AMP
Question 4 IWE AMP Revised Feb. 17, 2006 R. Morante (AMP-356)

(1) Identify the specific locations around the circumference in the former sandbed region where UT thickness readings have been and will be taken from inside containment. Confirm that all points previously recorded will be included in future inspections.

(2) Describe the grid pattern at each location (meridional length, circumferential length, grid point spacing, total number of point readings), and graphically locate each grid pattern within the former sandbed region.

(3) For each grid location, submit a graph of remaining thickness versus time, using the UT readings since the initiation of the program (both prior to and following removal of the sand and application of the external coating).

(4) Clearly describe the methodology and acceptance criteria that is applied to each grid of point thickness readings, including both global (entire array) evaluation and local (subregion of array) evaluation.

Assigned To: Ouaou, Ahmed

Response:

Response:
1. The circumference of the drywell is divided into 10 bays, designated as Bays 1, 3, 5, 7, 9, 11,13, 15, 17, and 19. UT thickness readings have been taken in each bay at one or more locations. The specific locations around the circumference in the former sand bed region where UT thickness reading have been taken from inside containment are Bay 1D, 3D, 5D, 7D, 9A, 9D, 11A, 11C, 13A, 13C, 13D, 15A, 15D, 17A, 17D, 17/19 Frame, 19A, 19B, and 19C. For each location, UT measurements were taken centered at elevation 11'-3". These represent the locations where UT measurements were taken in 1992, 1994, and 1996.

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In addition UT measurements were taken one time inside 2 trenches excavated in drywell floor concrete. The purpose of these UT measurements is to determine the extent of corrosion in the lower portions of the sand bed region prior to removing the sand and making accessible for visual inspection.

Future UT thickness measurements will be taken at the same locations as those inspected in 1996 in accordance with Oyster Creek commitment documented in NRC Question #AMP-209.

2. For locations where the initial investigations found significant wall thinning (9D, 11A, 11C, 13A, 13D, 15D, 17A, 17D, 17/19 Frame, 19A, 19B, and 19C) the grid pattern consists of 7 x 7 grid centered at elevation 11'-3 (meridian) and centered at the centerline of the tested location within each bay, which consists of 6"x 6" square template. The grid spacing is 1" on center. There are 49 point readings. For graphical location of the grid, refer to attachment 1.

For locations where the initial investigations found no significant wall thinning (1D, 3D, 5D, 7D, 9A, 13C, and 15A) the grid pattern consists of 1 x 7 grid centered at elevation 11'-3" (meridian) on 1" centers. There are 7 point readings. For graphical location of the grid, refer to attachment 1.

3. A graph representing the remaining thickness versus time using UT reading since the initiation of the program (both prior to and following removal of the sand and application of the external coating) for location 9D, 11A, 11C, 13A, 13D,15D,17A,17D,17/19, 19A, 19B, and 19C is included in the attached graph. Other locations (i.e. 1D, 3D, 5D, 7D, 9A, 13C, and 15A) are not included because wall thinning is not significant and the trend line will be essentially a straight line.

4. The methodology and acceptance criteria that is applied to each grid of point thickness readings, including both global (entire array) evaluation and local (subregion of array) is described in engineering specification IS-328227-004 and in calculation No. C-1302-187-5300-011. These documents were submitted to the NRC in a letter dated November 26, 1990 and provided to the Staff during the AMP/AMR audit. A brief summary of the methodology and acceptance criteria is described below.

The initial locations where corrosion loss was most severe in 1986 and 1987 were selected for repeat inspection over time to measure corrosion rate. For location where the initial investigations found significant wall thinning UT inspection consists of 49 individual UT data points equally spaced over a 6"x 6" area. Each new set of 49 values was then tested for normal distribution.

The mean values of each grid were then compared to the required minimum uniform thickness criteria of 0.736. In addition each individual reading is compared to the local minimum required criteria of 0.49. The basis for the required minimum uniform thickness criteria and the local minimum required criteria is provided in response to NRC Question #AMP-210.

A decrease in the mean value over time is representative of corrosion. If corrosion does not exist, the mean value will not vary with time except for random variations in the UT measurements.

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If corrosion is continuing, the mean thickness will decrease linearly with time. Therefore the curve fit of the data is tested to determine if linear regression is appropriate, in which case the corrosion rate is equal to the slope of the line. If a slope exists, then upper and lower 95% confidence intervals of the curve fit are calculated. The lower 95% confidence interval is then projected into the future and compared to the required minimum uniform thickness criteria of 0.736.

A similar process is applied to the thinnest individual reading in each grid. The curve fit of the data is tested to determine if linear regression is appropriate. If a slope exists, then the lower 95% confidence interval is then projected into the future and compared to the required minimum local thickness criteria of .49.

LRCR #: *LRA A.5 Commitment #:*

IR#:

Approvals:

Prepared By: Ouaou, Ahmed 4/ 4/2006

Reviewed By: Getz, Stu 4/ 5/2006

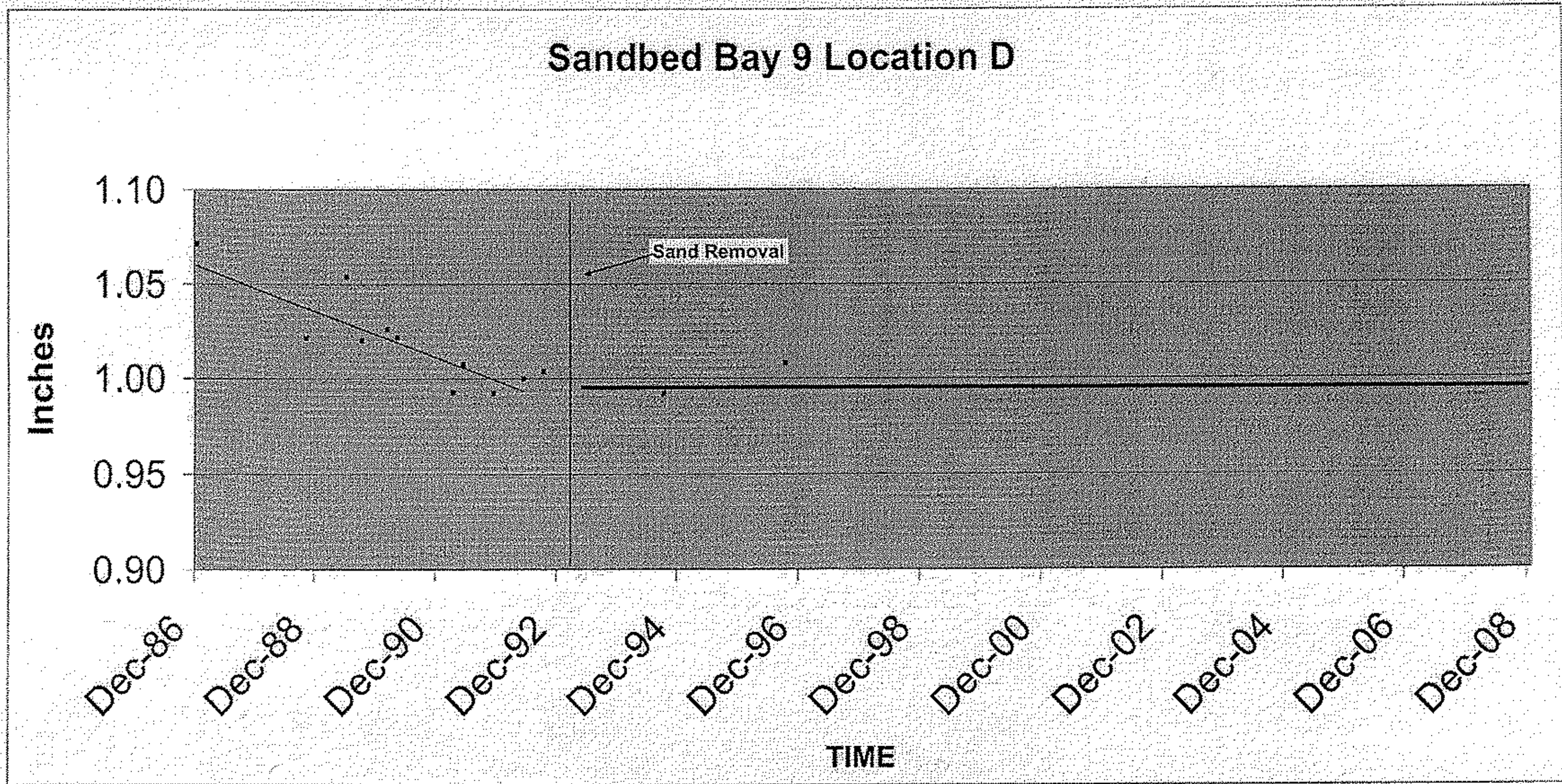
Approved By: Warfel, Don 4/ 5/2006

NRC Acceptance (Date):

Oyster Ceek Drywell Vessel Corrosion Rate Trending Program

Average Measured Thicknesses

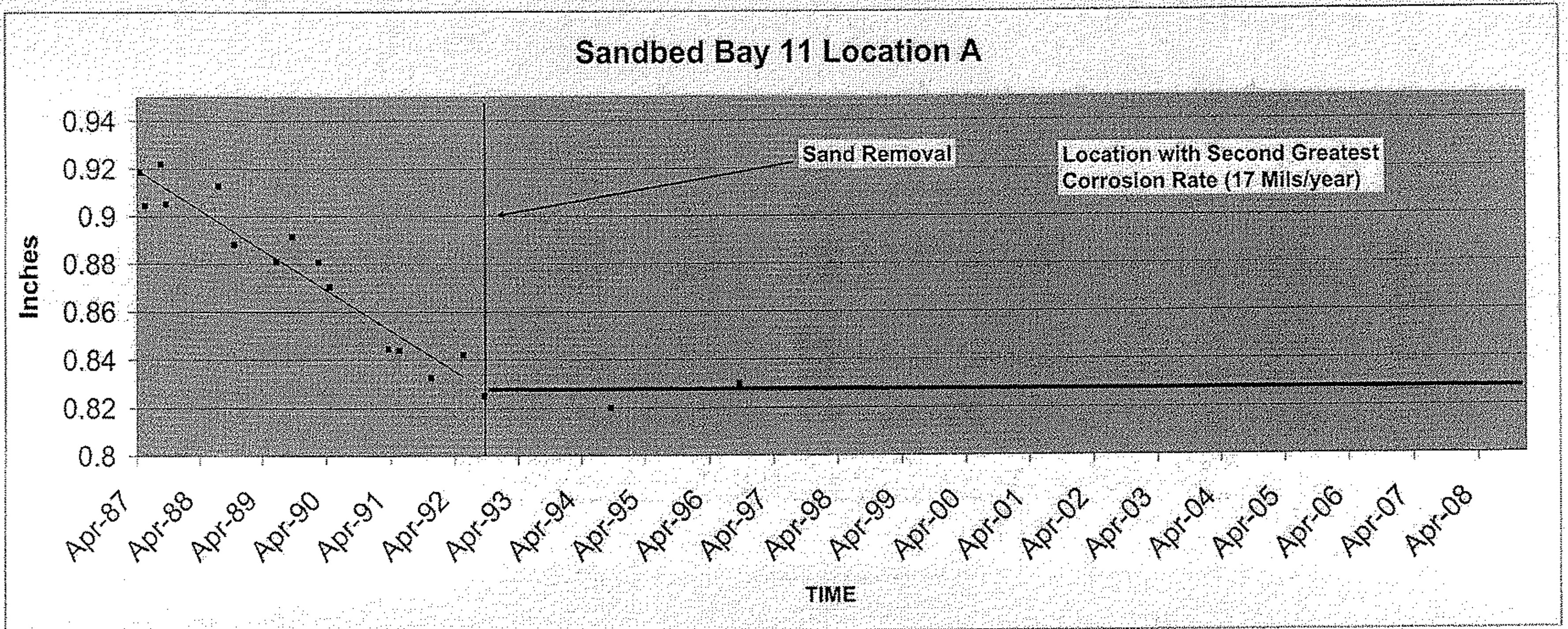
Bay	Date	Dec-86	Feb-87	Apr-87	May-87	Aug-87	Sep-87	Jul-88	Oct-88	Jun-89	Sep-89	Feb-90	Apr-90	Mar-91	May-91	Nov-91	May-92	Sep-92	Sep-94	Sep-96
1D									1.115										1.101	1.151
3D									1.178										1.184	1.181
5D									1.174										1.168	1.173
7D									1.135										1.136	1.138
9A									1.155										1.157	1.155
9D		1.072							1.021	1.054	1.020	1.026	1.022	0.893	1.008	0.992	1.000	1.004	0.992	1.008
11A				0.919	0.905	0.922	0.905	0.913	0.888	0.881	0.892	0.881	0.870	0.845	0.844	0.833	0.842	0.825	0.820	0.830
11C	Bottom				0.917	0.954	0.916	0.906	0.891	0.877	0.891	0.870	0.865	0.858	0.863	0.856	0.882	0.859	0.850	0.883
	Top				1.046	1.109	1.079	1.045	1.009	1.016	1.005	0.952	0.977	0.982	1.018	0.964	1.010	0.970	0.984	1.042
13A		0.919							0.905	0.883	0.883	0.862	0.853	0.855	0.853	0.849	0.865	0.858	0.828	0.843
13C	Bottom													0.909	0.901	0.900	0.931	0.906	0.895	0.933
	Top													1.072	1.049	1.048	1.088	1.055	1.037	1.059
13D									0.982				0.932					1.001	0.959	0.990
15A									1.120										1.114	1.127
15D		1.089							1.059	1.060	1.061	1.059	1.057	1.060	1.050	1.042	1.065	1.058	1.053	1.066
17A	Bottom	0.999							0.957	0.965	0.955	0.954	0.951	0.935	0.942	0.933	0.948	0.941	0.934	0.997
	Top	0.999							1.133	1.130	1.131	1.128	1.128	1.131	1.129	1.123	1.125	1.125	1.129	1.144
17D			0.922		0.895	0.891	0.895	0.878	0.862	0.857	0.847	0.836	0.829	0.825	0.829	0.822	0.823	0.817	0.810	0.845
17/19	Bottom								1.004	0.999	0.955	1.010	1.006	0.987	0.982	0.971	0.990	0.989	0.975	0.991
	Top								0.982	1.019	1.131	0.990	0.986	0.975	0.969	0.954	0.972	0.976	0.963	0.967
19A			0.884		0.873	0.859	0.858	0.849	0.837	0.829	0.825	0.840	0.808	0.817	0.803	0.803	0.809	0.806	0.806	0.815
19B					0.898	0.892	0.888	0.864	0.857	0.826	0.845	0.812	0.837	0.853	0.844	0.846	0.847	0.840	0.824	0.837
19C					0.901	0.888	0.888	0.873	0.856	0.845	0.845	0.831	0.825	0.843	0.823	0.822	0.832	0.819	0.830	0.848



Based on Calculation C-1302-187-5300-021

Slope	Best Est	Date	Average Since 1992	Original Nominal Thickness	Minimum Uniform Required Thickness
-0.0125	0.9932	05/01/92	1.00012	1.154"	0.736"

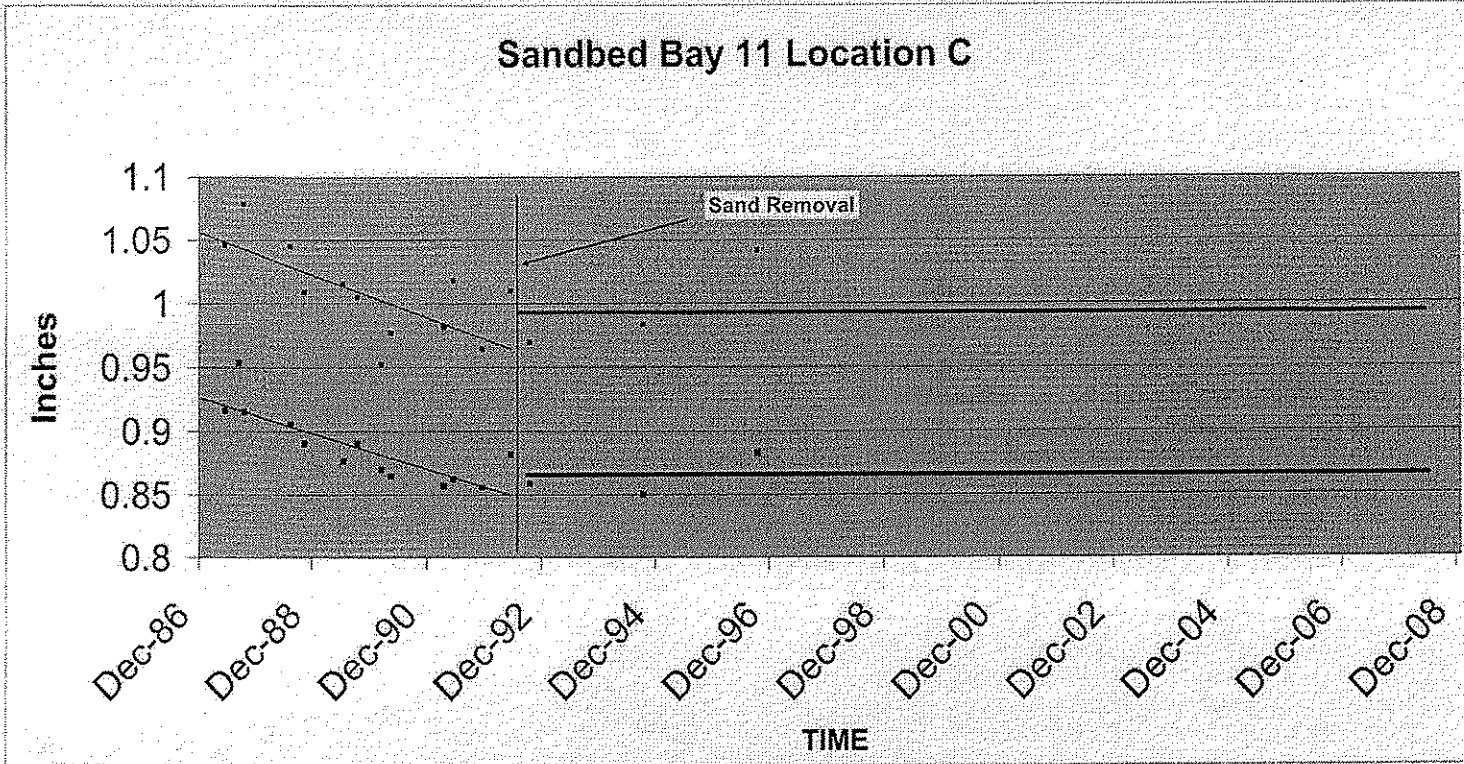
Dates	Dec-86	Feb-87	May-87	Aug-87	Sep-87	Jul-88	Oct-88	Jun-89	Sep-89	Feb-90	Apr-90	Mar-91	May-91	Nov-91	May-92	Sep-92	Sep-94	Sep-96
9D	1.0715						1.0214	1.0540	1.0200	1.0260	1.0217	0.9926	1.0075	0.9924	1.0000	1.0036	0.9920	1.0080



Based on Calculation C-1302-187-5300-021

Slope	Best Est.	Date	Average Since 1992	Original Nominal Thickness	Minimum Unform Required Thickness
-0.0171	0.83311	05/01/92	0.8251	1.154"	0.736"

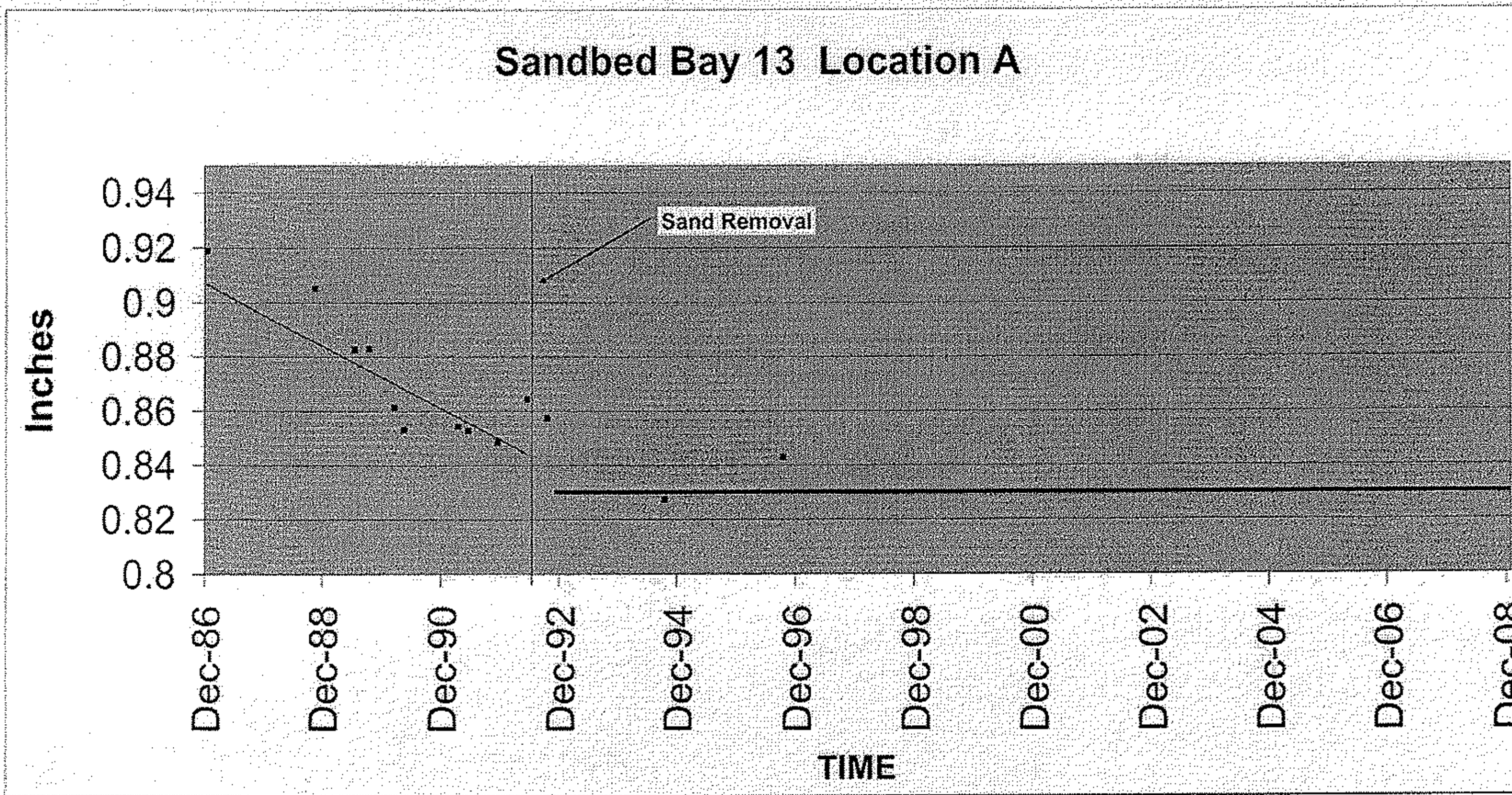
Dates	Dec-86	Feb-87	Apr-87	May-87	Aug-87	Sep-87	Jul-88	Oct-88	Jun-89	Sep-89	Feb-90	Apr-90	Mar-91	May-91	Nov-91	May-92	Sep-92	Sep-94	Sep-96
11A			0.9187	0.90464	0.92209	0.9052	0.913	0.8882	0.881	0.8916	0.8808	0.8704	0.8446	0.844	0.8326	0.842	0.8252	0.82	0.83



Based on Calculation C-1302-187-5300-021

Slope	Slope	Best Est. Low	Best Est. High	Date	Average Since 1992	Average Since 1992	Original Nominal Thickness	Minimum Uniform Required Thickness
-0.0143	-0.0171	0.8498	0.9642	05/01/92	0.8641	0.9984	1.154"	0.736"

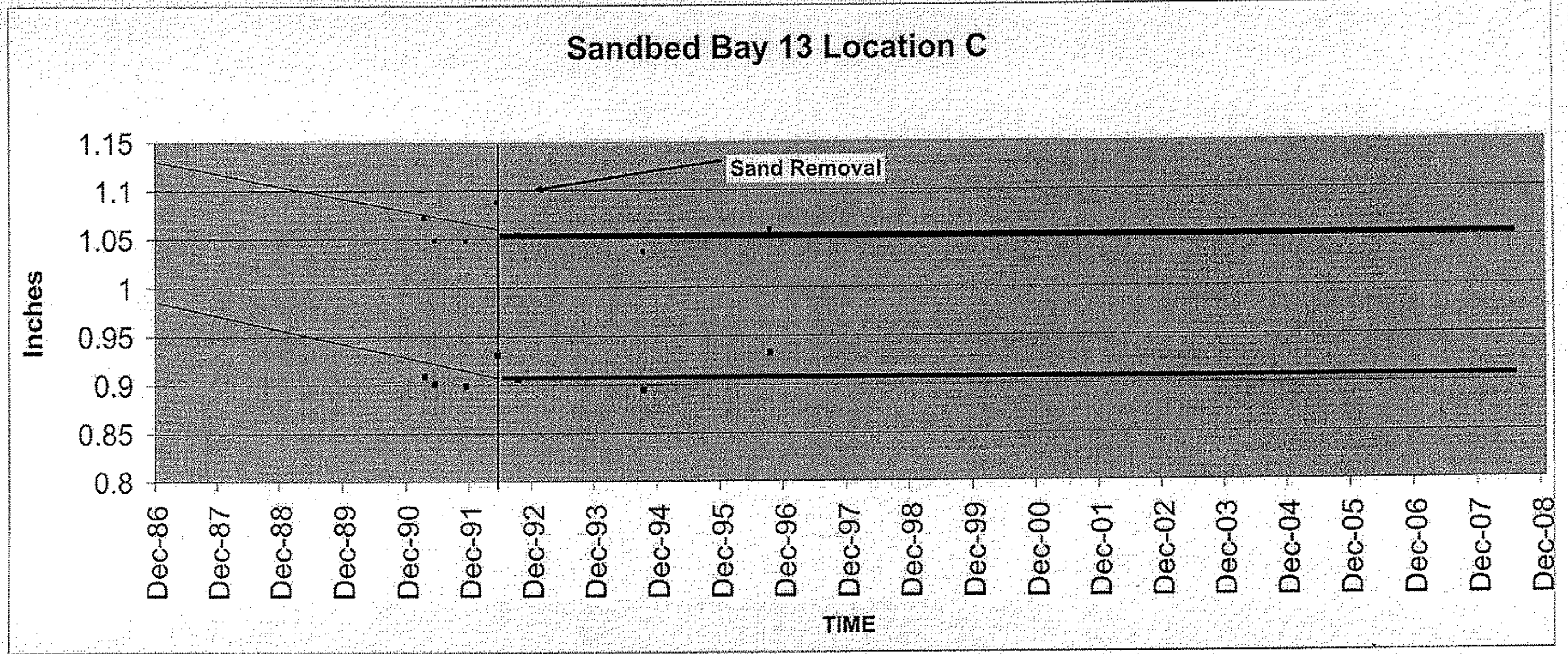
Dates	11C Bottom	11C Top
Dec-86		
Feb-87		
Apr-87		
May-87	0.91679	1.046
Aug-87	0.95364	1.1086
Sep-87	0.91571	1.0791
Jul-88	0.9061	1.0454
Oct-88	0.8907	1.0089
Jun-89	0.8768	1.0158
Sep-89	0.8907	1.005
Feb-90	0.8703	0.9522
Apr-90	0.865	0.977
Mar-91	0.8575	0.9817
May-91	0.8626	1.018
Nov-91	0.8563	0.9643
May-92	0.882	1.01
Sep-92	0.8591	0.9697
Sep-94	0.8503	0.9836
Sep-96	0.883	1.0418



Based on Calculation C-1302-187-5300-021

Slope	Best Est.	Date	Average Since 1992	Original Nominal Thickness	Minimum Unform Required Thickness
-0.012	0.8442	05/01/92	0.8386	1.154"	0.736"

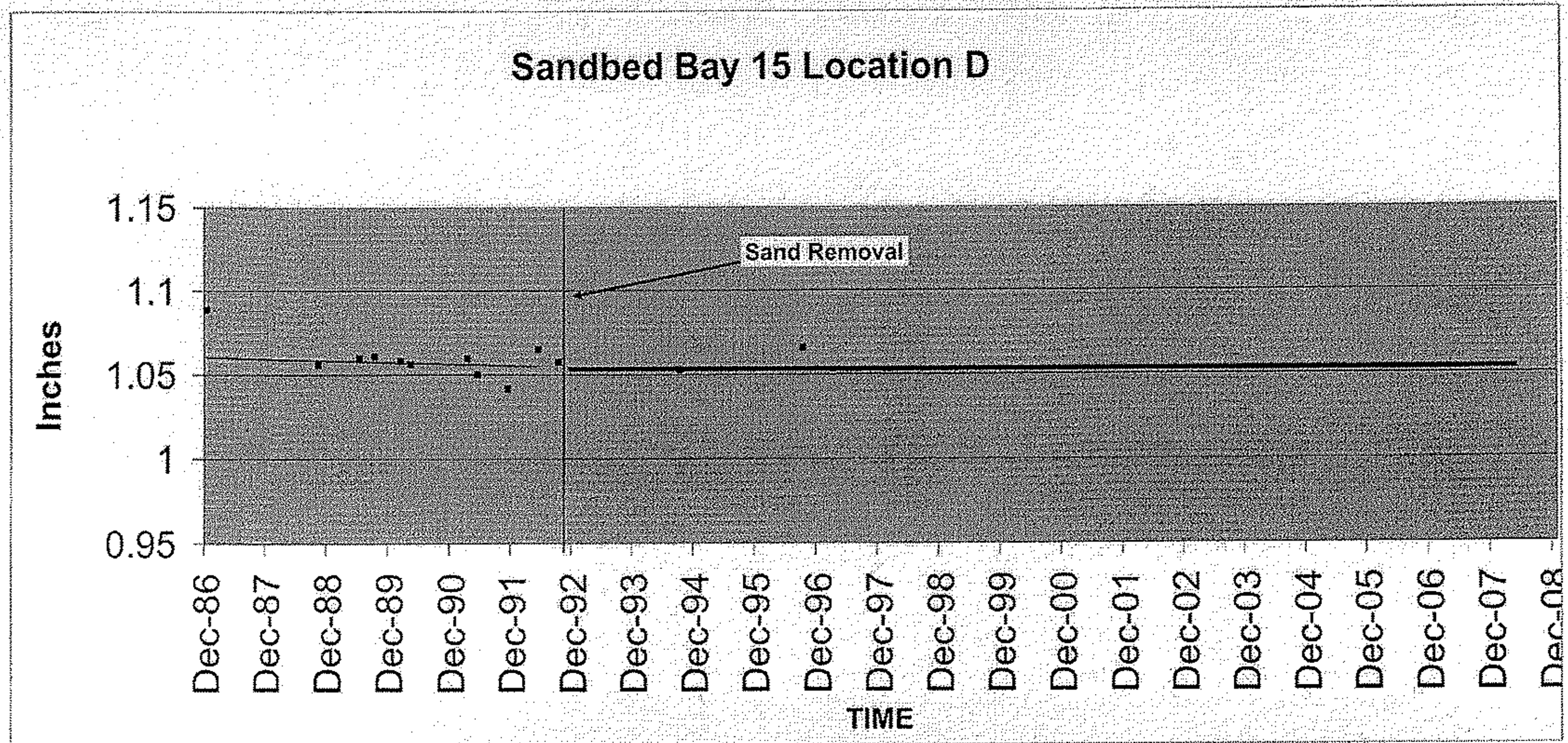
Dates	Dec-86	Feb-87	Apr-87	May-87	Aug-87	Sep-87	Jul-88	Oct-88	Jun-89	Sep-89	Feb-90	Apr-90	Mar-91	May-91	Nov-91	May-92	Sep-92	Sep-94	Sep-96
	0.91908							0.9053	0.8828	0.883	0.8615	0.8531	0.8545	0.8529	0.8486	0.8645	0.8576	0.8275	0.843



Based on Calculation C-1302-187-5300-021

Slope	Slope	Best Est. Low	Best Est. Hig	Date	Average Since 1992	Average Since 1992	Original Nominal Thickness	Minimum Unform Required Thickness
-0.013	-0.0146	0.9073	1.06	05/01/92	1.0505	0.9114	1.154"	0.736"

Dates	Dec-86	Feb-87	Apr-87	May-87	Aug-87	Sep-87	Jul-88	Oct-88	Jun-89	Sep-89	Feb-90	Apr-90	Mar-91	May-91	Nov-91	May-92	Sep-92	Sep-94	Sep-96
13C Bottom													0.9094	0.9013	0.8996	0.9305	0.906	0.8953	0.933
13C Top													1.0722	1.0488	1.0479	1.0882	1.0546	1.037	1.0593

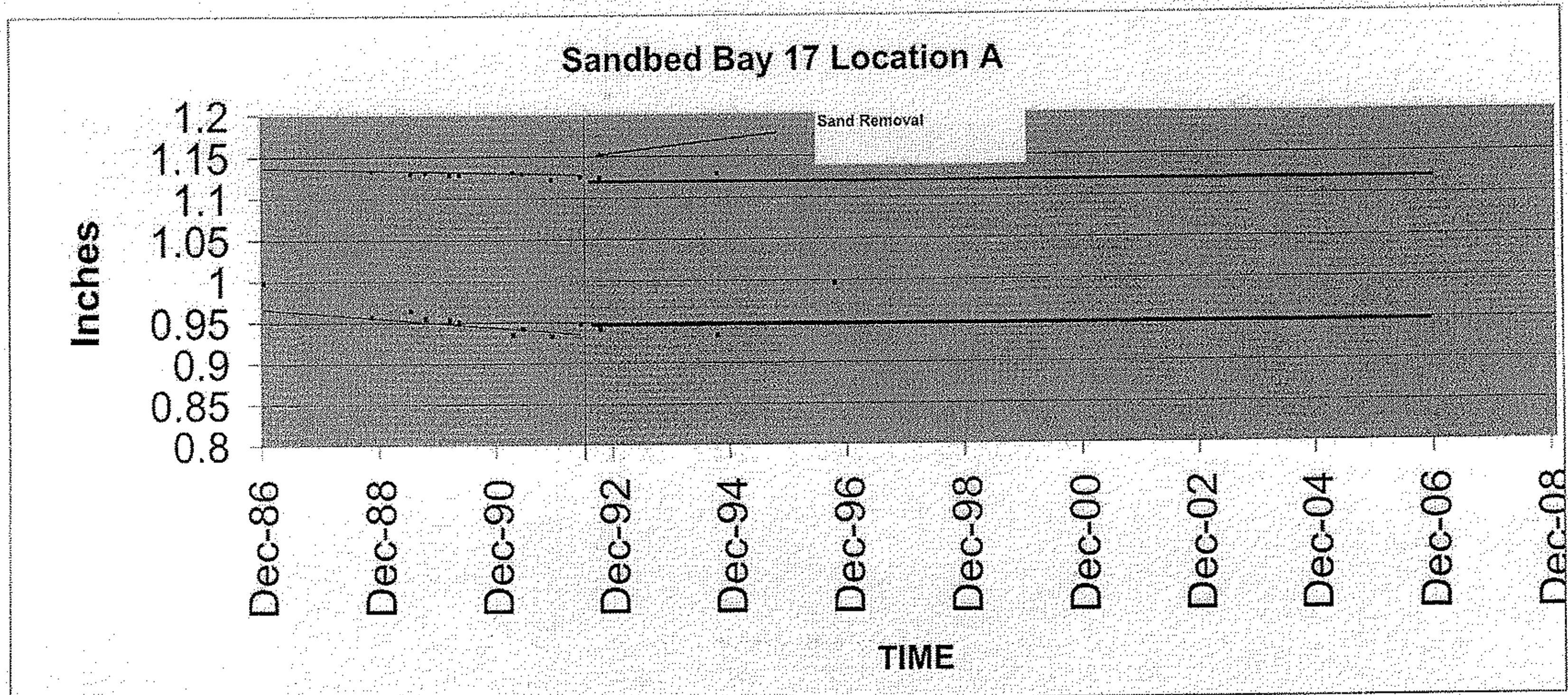


Based on Calculation C-1302-187-5300-021

Slope	Best Est.	Date	Average Since 1992	Original Nominal Thickness	Minimum Unform Required Thickness
-0.001	1.055	05/01/92	1.0588	1.154"	0.736"

Dates	Dec-86	Feb-87	Apr-87	May-87	Aug-87	Sep-87	Jul-88	Oct-88	Jun-89	Sep-89	Feb-90	Apr-90	Mar-91	May-91	Nov-91	May-92	Sep-92	Sep-94	Sep-96
	1.089							1.056	1.06	1.0609	1.0586	1.0565	1.0598	1.0502	1.0417	1.0652	1.0577	1.053	1.066

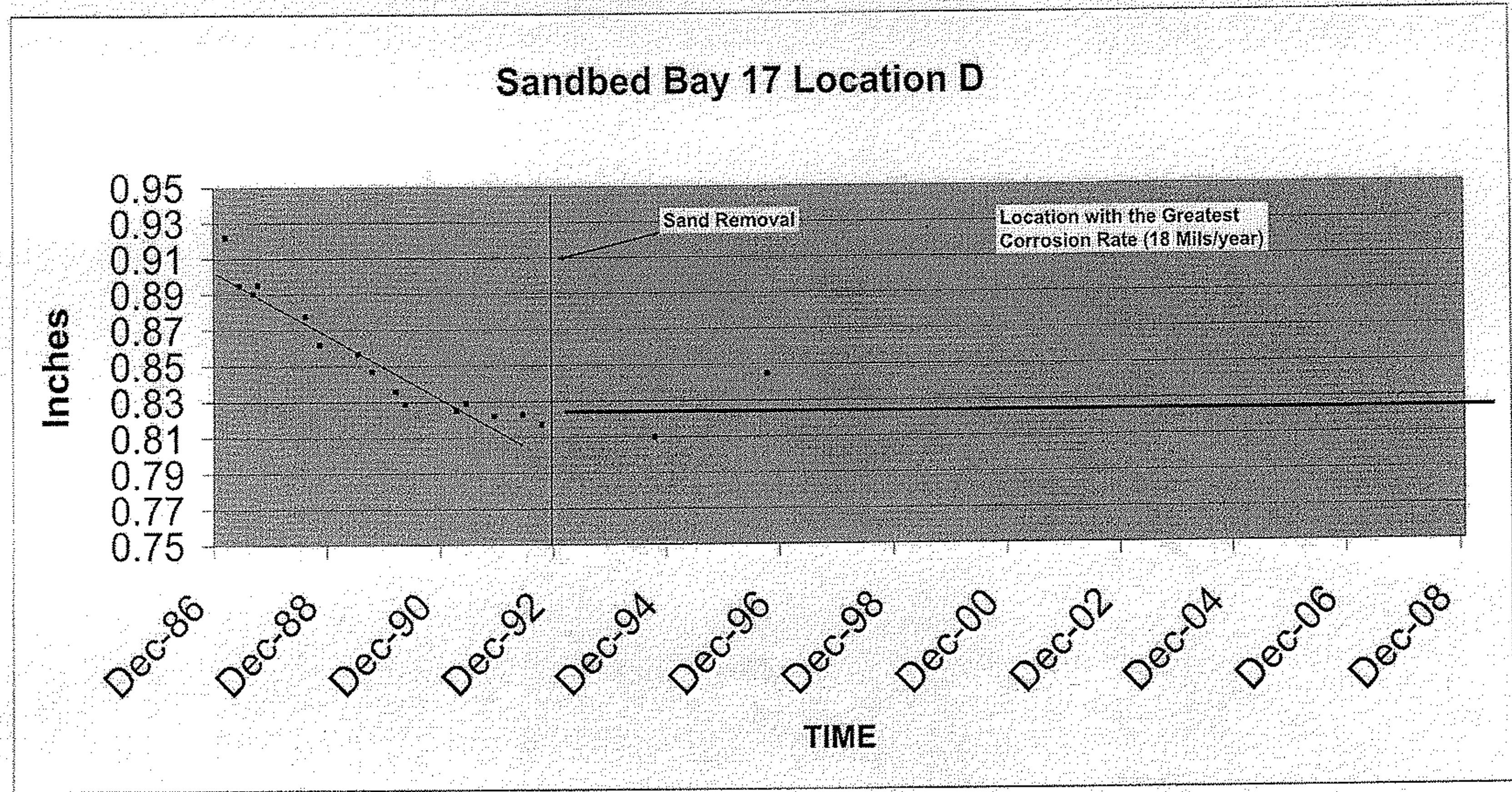
Sandbed Bay 17 Location A



Based on Calculation C-1302-187-5300-021

Slope	Slope	Best Est. Low	Best Est. High	Date	Average Since 1992	Average Since 1992	Original Nominal Thickness	Minimum Uniform Required Thickness
-0.0058	-0.0017	0.9352	1.1278	05/01/92	1.1326	0.9573	1.154"	0.736"

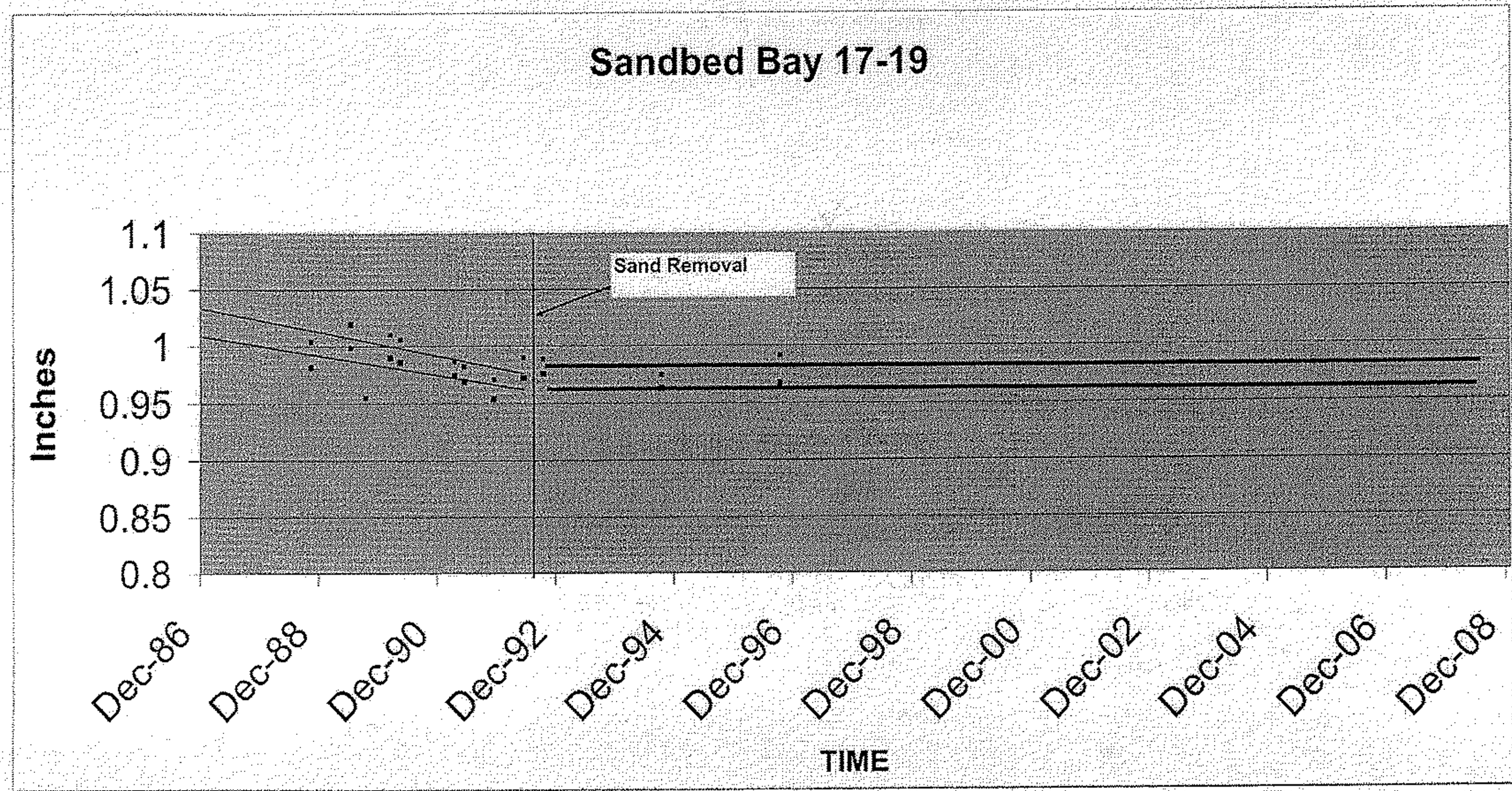
Dates	Dec-86	Feb-87	Apr-87	May-87	Aug-87	Sep-87	Jul-88	Oct-88	Jun-89	Sep-89	Feb-90	Apr-90	Mar-91	May-91	Nov-91	May-92	Sep-92	Sep-94	Sep-96
17A bottom	0.999							0.9574	0.9645	0.9552	0.9536	0.9508	0.9347	0.9424	0.9328	0.9481	0.9413	0.9338	0.9969
17A Top	0.999							1.1331	1.13	1.1308	1.128	1.1263	1.1309	1.1293	1.1226	1.1254	1.1248	1.1289	1.1441



Based on Calculation C-1302-187-5300-021

Slope	Best Est.	Date	Average Since 1992	Original Nominal Thickness	Minimum Uniform Required Thickness
-0.018	0.8057	05/01/92	0.8239	1.154"	0.736"

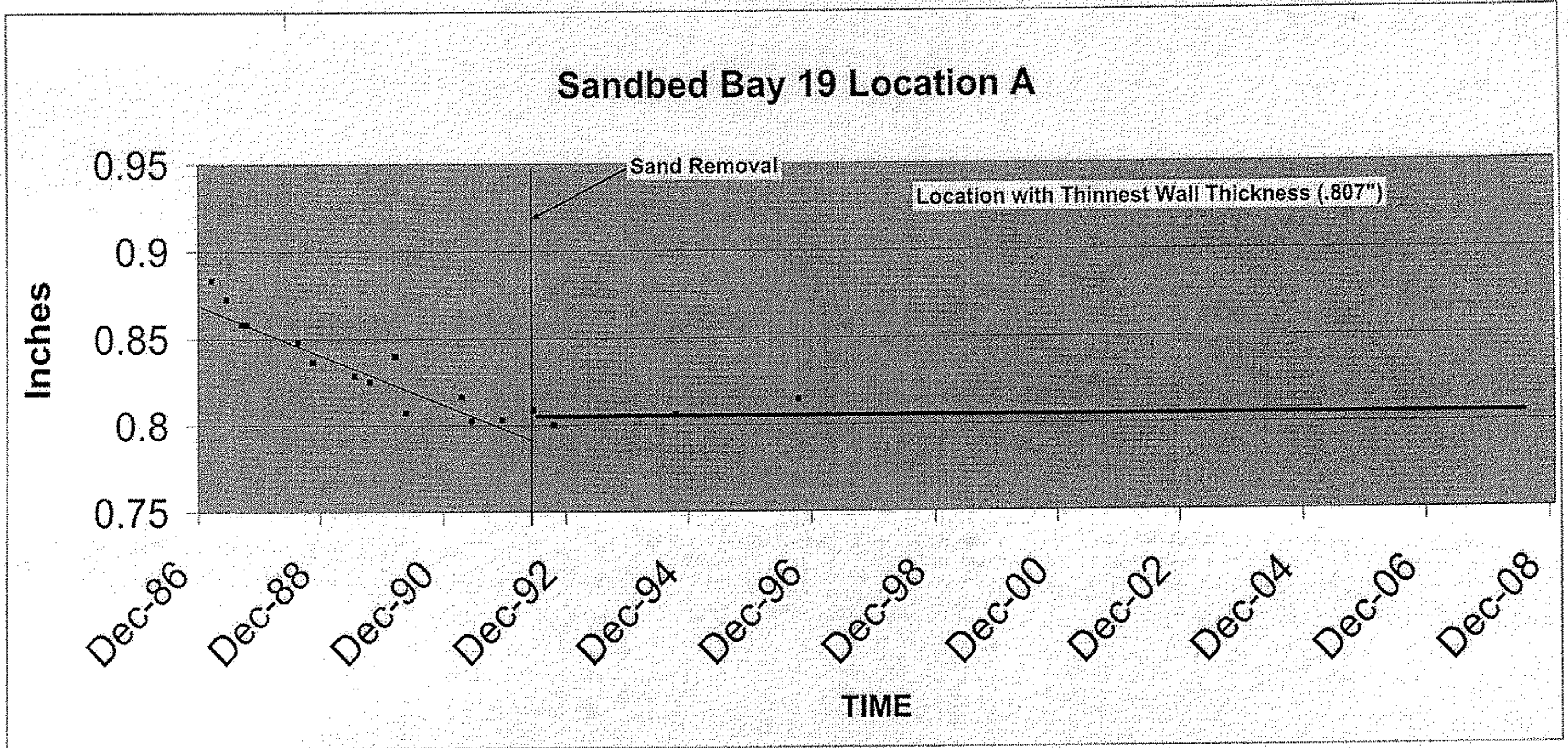
Dates	Dec-86	Feb-87	Apr-87	May-87	Aug-87	Sep-87	Jul-88	Oct-88	Jun-89	Sep-89	Feb-90	Apr-90	Mar-91	May-91	Nov-91	May-92	Sep-92	Sep-94	Sep-96
17D		0.92217		0.89507	0.89069	0.89528	0.8779	0.8622	0.8568	0.8471	0.8358	0.829	0.8253	0.8291	0.8222	0.823	0.8172	0.81	0.845



Based on Calculation C-1302-187-5300-021

Slope	Slope	Best Est. Low	Best Est. High	Date	Average Since 1992	Average Since 1992	Original Nominal Thickness	Minimum Unform Required Thickness
-0.0087	-0.0107	0.9621	0.9761	05/01/92	0.9871	0.9689	1.154"	0.736"

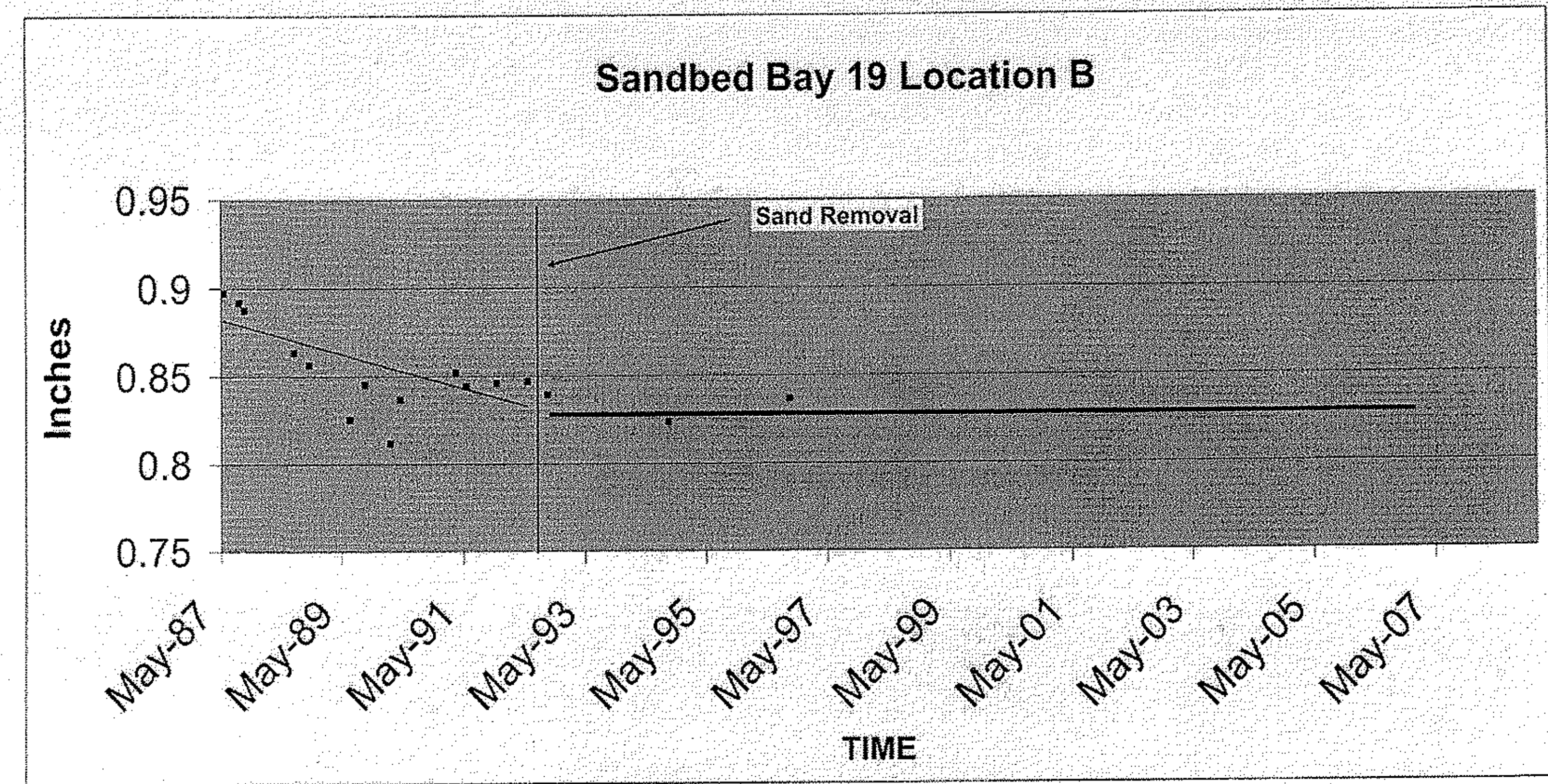
Dates	Dec-86	Feb-87	Apr-87	May-87	Aug-87	Sep-87	Jul-88	Oct-88	Jun-89	Sep-89	Feb-90	Apr-90	Mar-91	May-91	Nov-91	May-92	Sep-92	Sep-94	Sep-96
17/19 Top								0.9817	1.0191	1.1308	0.9898	0.986	0.9746	0.9693	0.9542	0.9722	0.976	0.963	0.9674
17/19 Bottom								1.0038	0.9988	0.9552	1.01	1.0057	0.987	0.9824	0.9711	0.99	0.9887	0.9748	0.9914



Based on Calculation C-1302-187-5300-021

Slope	Best Est.	Date	Average Since 1992	Original Nominal Thickness	Minimum Unform Required Thickness
-0.015	0.7911	05/01/92	0.8071	1.154"	0.736"

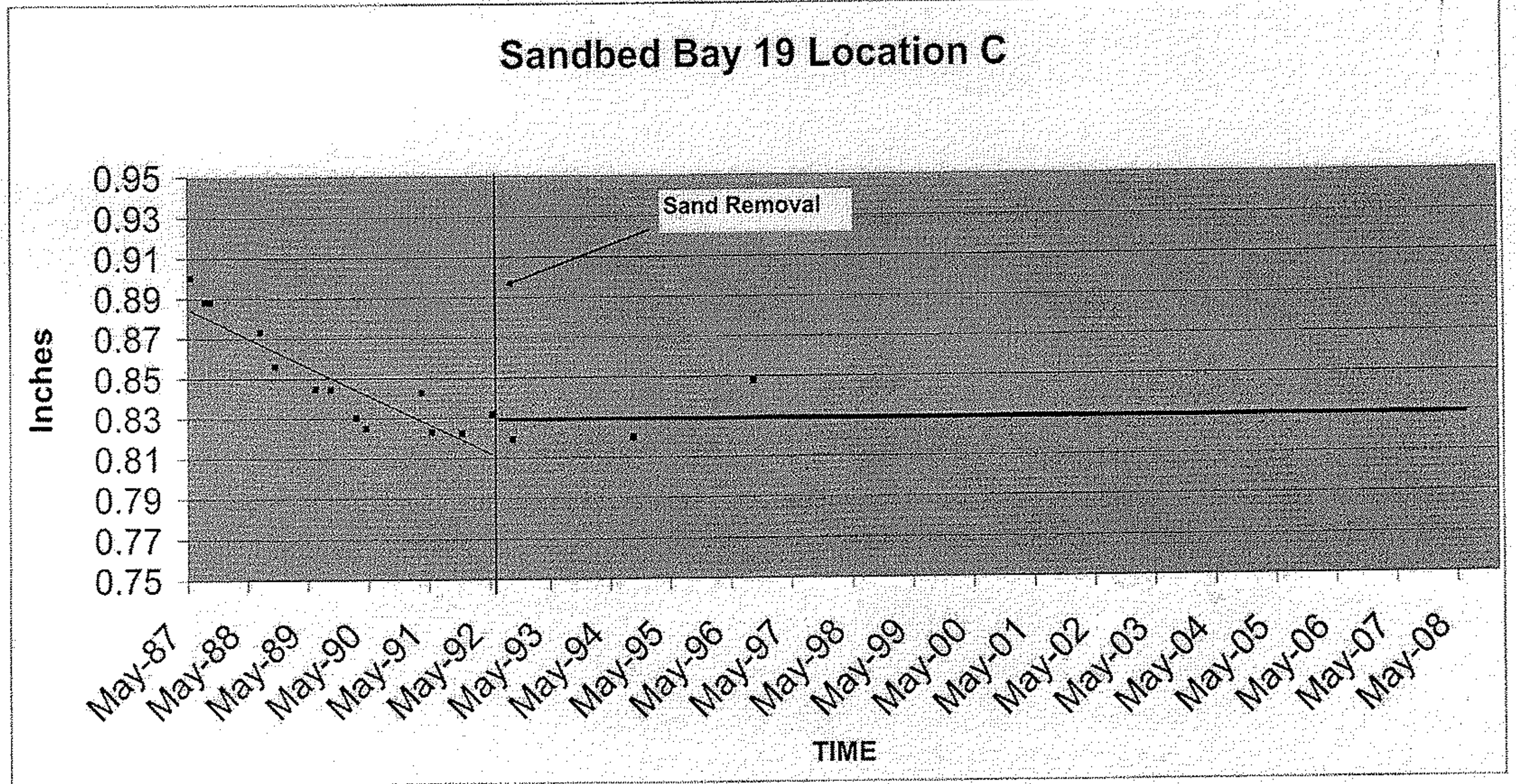
Dates	Dec-86	Feb-87	Apr-87	May-87	Aug-87	Sep-87	Jul-88	Oct-88	Jun-89	Sep-89	Feb-90	Apr-90	Mar-91	May-91	Nov-91	May-92	Sep-92	Sep-94	Sep-96
19A		0.88364		0.87293	0.8586	0.85829	0.8486	0.8369	0.8288	0.8254	0.8399	0.8076	0.8167	0.8028	0.8032	0.8091	0.8002	0.806	0.815



Based on Calculation C-1302-187-5300-021

Slope	Best Est.	Date	Average Since 1992	Original Nominal Thickness	Minimum Unform Required Thickness
-0.0099	0.8330	05/01/92	0.8337	1.154"	0.736"

Dates	Dec-86	Feb-87	Apr-87	May-87	Aug-87	Sep-87	Jul-88	Oct-88	Jun-89	Sep-89	Feb-90	Apr-90	Mar-91	May-91	Nov-91	May-92	Sep-92	Sep-94	Sep-96
19B				0.89763	0.89221	0.8876	0.864	0.8565	0.8256	0.84549	0.812	0.8369	0.8525	0.8444	0.8463	0.8472	0.8396	0.824	0.837



Based on Calculation C-1302-187-5300-021

Slope	Best Est.	Date	Average Since 1992	Original Nominal Thickness	Minimum Unform Required Thickness
-0.015	0.8117	05/01/92	0.829	1.154"	0.736"

Dates	Dec-86	Feb-87	Apr-87	May-87	Aug-87	Sep-87	Jul-88	Oct-88	Jun-89	Sep-89	Feb-90	Apr-90	Mar-91	May-91	Nov-91	May-92	Sep-92	Sep-94	Sep-96
19C				0.90051	0.88816	0.88831	0.8735	0.8563	0.845	0.8447	0.8305	0.8251	0.8428	0.8232	0.8223	0.8319	0.8192	0.82	0.848

NRC Information Request Form

Item No AMP-141	***Date Received:*** 10/ 6/2005 ***Source*** AMP Audit
Topic: IWE	***Status:*** Open

Document References:
B.1.27

NRC Representative Morante, Rich

AmerGen (Took Issue): Hufnagel, Joh

Question

AMP B.1.27 IWE

a. Visual inspection of the coatings in the former sandbed region of the drywell is currently conducted under the applicant's protective coatings monitoring and maintenance program; only this AMP is credited for managing loss of material due to corrosion for license renewal. Visual inspection of the containment shell conducted in accordance with the requirements of IWE is typically credited to manage loss of material due to corrosion.

The applicant is requested to provide its technical basis for not also crediting its IWE program for managing loss of material due to corrosion in the former sandbed region of the drywell.

B. During discussions with the applicant's staff on 10/04/05 about augmented inspection conducted under IWE, the applicant presented tabulated inspection results obtained from the mid 1980s to the present, to monitor the remaining drywell wall thickness in the cylindrical and spherical regions where significant corrosion of the outside surface was previously detected.

The applicant is requested to provide (1) a copy of these tabulated inspection results, (2) a list of the nominal design thicknesses in each region of the drywell, (3) a list of the minimum required thicknesses in each region of the drywell, and (4) a list of the projected remaining wall thicknesses in each region of the drywell in the year 2029.

AMP B.1.27 IWE Question on Remaining Wall Thickness in the Former Sandbed Region of the Drywell

c. During discussions with the applicant's staff on 10/05/05, the applicant described the history and resolution of corrosion in the sandbed region. After discovery, thickness measurements were taken from 1986 through 1992, to monitor the progression of wall loss. Remedial actions were completed in early 1993. At that time, the remaining wall thickness exceeded the minimum required thickness. The applicant concluded that it had completely corrected the conditions which led to the corrosion, and terminated its program to monitor the remaining wall thickness. At that time, the remaining years of operation was expected to be no more than 16 years (end of the current license term).

NRC Information Request Form

The applicant's aging management commitment for license renewals is limited to periodic inspection of the coating that was applied to the exterior surface of the drywell as part of the remedial actions. The applicant has not made a license renewal commitment to measure wall thickness in the sandbed region in order to confirm the effectiveness of the remedial actions taken.

Assigned To: Ouaou, Ahmed

Response:

a) Visual inspection of the containment drywell shell, conducted in accordance with ASME Section XI, Subsection IWE, is credited for aging management of accessible areas of the containment drywell shell. Typically this inspection is for internal surfaces of the drywell. The exterior surfaces of the drywell shell in the sand bed region for Mark I containment is considered inaccessible by ASME Section XI, Subsection IWE, thus visual inspection is not possible for a typical Mark I containment including Oyster Creek before the sand was removed from the sand bed region in 1992. After removal of the sand, an epoxy coating was applied to the exterior surfaces of the drywell shell in the sand bed region. The region was made accessible during refueling outages for periodic inspection of the coating. Subsequently Oyster Creek performed periodic visual inspection of the coating in accordance with an NRC current licensing basis commitment. This commitment was implemented prior to implementation of ASME Section XI, Subsection IWE. As a result inspection of the coating was conducted in accordance with the Protective Coating Monitoring and Maintenance Program. Our evaluation of this aging management program concluded the program is adequate to manage aging of the drywell shell in the sand bed region during the period of extended operation consistent with the current licensing basis commitment, and that inclusion of the coating inspection under IWE is not required. However we are amending this position and will commit to monitor the protective coating in the exterior surfaces of the drywell in the sand bed region in accordance with the requirements of ASME Section XI, Subsection IWE during the period of extended operation. For details related to implementation of this commitment, refer to the response to NRC AMP Question #188.

b) A tabulation of ultrasonic testing (UT) thickness measurement results in monitored areas of the drywell spherical region above the sand bed region and in the cylindrical region is included in ASME Section XI, Subsection IWE Program Basis Document (PBD-AMP-B.1.27) Notebook. The tabulation contains information requested by the Staff and is available for review during AMP audit. The tabulation is also provided in Table -1, and Table-2 below.

c) In December 1992, with approval from the NRC a protective epoxy coating was applied to the outside surface of the drywell shell in the sand bed region to prevent additional corrosion in that area. UT thickness measurements taken in 1992, and in 1994, in the sand bed region from inside the drywell confirmed that the corrosion in the sand bed region has been arrested. Periodic inspection of the coating indicates that the coating in that region is performing satisfactorily with no signs of deterioration such as blisters, flakes, or discoloration, etc. Additional UT measurements, taken in 1996 from inside the drywell in the sand bed region showed no ongoing corrosion and provided objective evidence that corrosion has been arrested.

NRC Information Request Form

As a result of these UT measurements and the observed condition of the coating, we concluded that corrosion has been arrested and monitoring of the protective coating alone, without additional UT measurements, will adequately manage loss of material in the drywell shell in the sand bed region. However to provide additional assurance that the protective coating is providing adequate protection to ensure drywell integrity, Oyster Creek will perform periodic confirmatory UT inspections of the drywell shell in the sand bed region. The initial UT measurements will be taken prior to entering the period of extended operation and then every 10 years thereafter. The UT measurements will be taken from inside the drywell at the same locations where the UT measurements were taken in 1996. This revises the license renewal commitment communicated to the NRC in a letter from C. N. Swenson Site Vice President, Oyster Creek Generating Station to U. S. Nuclear Regulatory Commission, "Additional Commitments Associated with Application for renewed Operating License - Oyster Creek Generating Station", dated 12/9/2005. This letter commits to one-time inspection to be conducted prior to entering the period of extended operation. The revised commitment will be to conduct UT measurements on a frequency of 10 years, with the first inspection to occur prior to entering the period of extended operation.

This response was revised to incorporate additional commitments on UT examinations for the sand bed region discussed with NRC Audit team on 1/26/2006.
This response was revised to reference response to NRC Question #AMP-188 and RAI 4.7.2-1(d). AMO 4/1/2006.
The response was revised to add Table-1, and Table-2, and delete reference to RAI 4.7.2-1(d) AMO 4/5/2006.

LRCR #: 229 *LRA A.5 Commitment #:* 27

IR#:

Approvals:

Prepared By: Ouaou, Ahmed 4/ 5/2006

Reviewed By: Getz, Stu 4/ 5/2006

Approved By: Warfel, Don 4/ 5/2006

NRC Acceptance (Date):

Table –1. UT Thickness measurements for the Upper Region of the Drywell Shell

Monitored Elevation	Location	Minimum Required Thickness, inches [5]	Average Measured Thickness [1,2,3], inches											Projected Lower 95% Confidence Thickness in 2029
			1987	1988	1989	1990	1991	1992	1993 [3]	1994	1996	2000	2004	
Elevation 50' 2"		0.541"												
	Bay 5–D12					0.743 0.745 0.746	0.742 0.745 0.748	0.747 0.747		0.741	0.748	0.741	0.743	No Ongoing Corrosion
	Bay 5– 5H					0.761 0.761	0.755 0.758 0.760	0.758 0.758		0.754	0.757	0.754	0.756	0.7384
	Bay 5– 5L					0.706 0.703	0.703 0.705 0.706	0.703 0.707		0.702	0.705	0.706	0.701	No Ongoing Corrosion
	Bay 13–31H					0.762 0.779	0.760 0.758 0.765	0.765 0.763		0.759	0.766	0.762	0.758	No Ongoing Corrosion
	Bay 13–31L					0.687 0.684	0.689 0.678 0.688	0.685 0.688		0.683	0.690	0.682	0.693	No Ongoing Corrosion
	Bay 15–23H					0.758 0.764	0.762 0.762 0.765	0.767 0.763		0.758	0.760	0.758	0.757	0.738
	Bay 15–23L					0.726 0.728	0.726 0.729 0.725	0.726 0.724		0.728	0.724	0.729	0.727	No Ongoing Corrosion
Elevation 51' 10"		0.541"												

Table –1. UT Thickness measurements for the Upper Region of the Drywell Shell

Monitored Elevation	Location	Minimum Required Thickness, inches [5]	Average Measured Thickness [1,2,4], inches											Projected Lower 95% Confidence Thickness in 2029
			1987	1988	1989	1990	1991	1992	1993 [3]	1994	1996	2000	2004	
	Bay 13–32H					0.716	0.715 0.715 0.719	0.717 0.717		0.714	0.715	0.715	0.713	No Ongoing Corrosion
	Bay 13–32L					0.686	0.683 0.683 0.682	0.683 0.676		0.680	0.684	0.679	0.687	No Ongoing Corrosion
Elevation 60' 10"		0.518"												
	Bay 1–5-22								0.693	0.711	0.692	0.689	0.689	No Ongoing Corrosion
Elevation 87' 5"		0.452"												
	Bay 9–20		0.619	0.622 0.620	0.619	0.620	0.614 0.612	0.629 0.614		0.613	0.613	0.604	0.612	0.604.
	Bay 13–28		0.643	0.641 0.642	0.645	0.643	0.635 0.629	0.641 0.637		0.640	0.636	0.635	0.640	No Ongoing Corrosion
	Bay 15–31		0.638	0.636 0.636	0.638	0.642	0.628 0.627	0.631 0.630		0.633	0.632	0.628	0.630	0.615

Notes:

1. The average thickness is based on 49 Ultrasonic Testing (UT) measurements performed at each location
2. Multiple inspections were performed in the years 1988, 1990, 1991, and 1992.
3. The 1993 elevation 60' 10" Bay 5-22 inspection was performed on January 6, 1993. All other locations were inspected in December 1992.
4. Accuracy of Ultrasonic Testing Equipment is plus or minus 0.010 inches.
5. Reference SE-000243-002.

Table –1. UT Thickness measurements for the Upper Region of the Drywell Shell

Conclusion:

Summary of Corrosion Rates of UT measurements taken through year 2004

- There is no ongoing corrosion at two elevations (51' 10" and 60' 10")
- Based on statistical analysis, one location at elevation 50' 2" is undergoing a minor corrosion rate of 0.0003 inches per year,
- Based on statistical analysis, two locations at elevation 87' 5" are undergoing minor corrosion rates of 0.0005 and 0.00075 inches per year

Table –2 UT Thickness measurements for the Sand Bed Region of the Drywell Shell

Location Bay	Sub Location	Dec 1986	Feb 1987	Apr 1987	May 1987	Aug 1987	Sep 1987	Jul 1988	Oct 1988	Jun 1989	Sep 1989	Feb 1990	Apr 1990	Mar 1991	May 1991	Nov 1991	May 1992	Sep 1992	Sep 1994	Sep 1996
1D									1.115										1.101	1.1514
3D									1.178										1.184	1.181
5D									1.174										1.168	1.173
7D									1.135										1.136	1.138
9A									1.155										1.157	1.155
9D		1.072							1.021	1.054	1.020	1.026	1.022	0.993	1.008	0.992	1.000	1.004	0.992	1.008
11A				0.919	0.905	0.922	0.905	0.913	0.888	0.881	0.892	0.881	0.870	0.845	0.844	0.833	0.842	0.825	0.820	0.830
11C	Bottom				0.917	0.954	0.916	0.906	0.891	0.877	0.891	0.870	0.865	0.858	0.863	0.856	0.882	0.859	0.850	0.883
	Top				1.046	1.109	1.079	1.045	1.009	1.016	1.005	0.952	0.977	0.982	1.018	0.964	1.010	0.970	0.984	1.042
13A		0.919							0.905	0.883	0.883	0.862	0.853	0.855	0.853	0.849	0.865	0.858	0.828	0.843
13C	Bottom													0.909	0.901	0.900	0.931	0.906	0.895	0.933
	Top													1.072	1.049	1.048	1.088	1.055	1.037	1.059
13D									0.962				0.932					1.001	0.959	0.990
15A									1.120										1.114	1.127
15D		1.089							1.056	1.060	1.061	1.059	1.057	1.060	1.050	1.042	1.065	1.058	1.053	1.066
17A	Bottom	0.999							0.957	0.965	0.955	0.954	0.951	0.935	0.942	0.933	0.948	0.941	0.934	0.997
	Top	0.999							1.133	1.130	1.131	1.128	1.128	1.131	1.129	1.123	1.125	1.125	1.129	1.144
17D			0.922		0.895	0.891	0.895	0.878	0.862	0.857	0.847	0.836	0.829	0.825	0.829	0.822	0.823	0.817	0.810	0.845
17/19	Top								0.982	1.019	1.131	0.990	0.986	0.975	0.969	0.954	0.972	0.976	0.963	0.967
	Bottom								1.004	0.999	0.955	1.010	1.006	0.987	0.982	0.971	0.990	0.989	0.975	0.991
19A			0.884		0.873	0.859	0.858	0.849	0.837	0.829	0.825	0.840	0.808	0.817	0.803	0.803	0.809	0.800	0.806	0.815
19B					0.898	0.892	0.888	0.864	0.857	0.826	0.845	0.812	0.837	0.853	0.844	0.846	0.847	0.840	0.824	0.837
19C					0.901	0.888	0.888	0.873	0.856	0.845	0.845	0.831	0.825	0.843	0.823	0.822	0.832	0.819	0.820	0.848

EXHIBIT J

CORRO-CONSULTA
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MEMORANDUM

MAY 3, 2006

To: Richard Webster, Esq.
Rutgers Environmental Law Clinic
123 Washington Street
Newark, NJ 07102-5695

Paul Gunter
Reactor Watchdog Project
Nuclear Information and Resource Service
Washington, DC 10036

Subject: **Oyster Creek Dry Well Corrosion**
Comments regarding "Audit Q&A (Question Numbers AMP –141, 210, 356) dated 4/5/06, Ref. ML060960563

I. Summary

The referenced document, which is attached, makes statements with regards to future corrosion and the presence or absence of a need for future inspections of the damaged areas in the sandbed region. After having examined the data presented we find that the conclusions are at odds with good or best corrosion practice. In general, predicting future corrosion based on data collected over the past is perhaps judicious only if a) the corrosion (or deterioration) mechanism is well known, and b) if assurances can be given that the circumstances (environment) under which prior corrosion occurred can and will be maintained in the future. Neither of these two imperatives can be assured at this time. Furthermore, it would seem that for meaningful predictions, the data set, on the basis of which extrapolations are carried out, needs to be consistent. I have found serious concerns in this respect as well. I, therefore, believe that the proposed long inspection intervals are totally unjustified.

II. Details

Following the removal of the sand bed in September of 1992 two sets of UT measurements were recorded in September 1994 and September of 1996. For twelve locations where initial investigations found significant wall thinning, these measurements were made over a 6" x 6" grid at 1 inch spacings, hence 49 individual UT measurements were gathered over an individual grid.[1] Identical grids were

[1] For seven other areas only seven readings were taken at one inch intervals.

surveyed in each of the bays in the sandbed area, and in some instances more than one grid was surveyed in the same bay. The clusters of individual UT measurements were evaluated by calculating the average of the 49 measurements, and naming the resulting averages "average minimum general thickness (AMGT)". The terminology "average minimum general thickness" is misleading because another shape or size of grid could have resulted in lower results.

Presumably the identical grid locations, which were surveyed in September of 1992, were again surveyed in September of 1994 and September of 1996. This generated three data points (AMGT) in each of the various bay locations as a function of time. Plots were then generated to show the AMGT as a function of time. These plots are shown in the Figure 1 below for 8 different bay locations, correlating the measurements as a function of time. Exelon drew straight horizontal lines through these points with extrapolation over the next 10 years to demonstrate the absence of continued corrosion.

This procedure and the subsequent evaluation and interpretation of the data warrant a number of comments.

First, it is noted that with unfailing regularity (even in the cases not shown in Figure 1 below) the AMGT for each grid decreases from 1992 to 1994, ,but then *increases* in 1996. This is of course physically impossible; metal simply does not spontaneously get thicker. Furthermore, statistically, UT measurements in general are accurate within a standard deviation of 2% of wall thickness (modern methodologies can do better). Assuming for a moment that all 49 measurements within a grid measured the same wall thickness, then the average would have had a standard deviation of 2/7 % or about 0.3% of wall thickness which corresponds to +/- 2.5 mils or 95% confidence limits of +/- 5 mils. Since however, not all measurements within the grid were of the same wall thickness, the variability of individual measurements within the grid must have been larger than the standard deviation derived from instrument capability. Exelon assures us that "the tolerance is or the order of +/- 10 mils. As it turns out, the variability of the AMGT for the three measurements for a particular grid expressed as a standard deviation in almost all cases exceeds the +/- 10 mils by a large margin. **I interpret this as a systematic error in the UT methodology employed.**

Moreover, the repeated trends over time observed in Fig. 1 below from high to low to high values underlines the unreliability to these UT measurements. Clearly, the methodology chosen by Exelon to monitor corrosion in the sandbed area is difficult in practice for a number of reasons. **However, these difficulties do not excuse the improper extrapolation of sporadic data obtained over a four-year period to another 10 years and the untenable and irresponsible conclusion that corrosion in the sandbed area therefore is under control.**

There is no doubt that the application of the epoxy coating has slowed the corrosion over the four years from 1992 to 1996. However, no assurances have been given that in the following years the environmental conditions in the sand bed area remained the same. Neither have there been any assertions that the epoxy coating did not deteriorate over time. Nor does one have any information about the rate of deterioration. It is well known that coatings (generally speaking) perform well for a period of time, but then deteriorate rapidly. In light of such knowledge, coupled with the reliability of the UT data submitted by Exelon, we find an inspection schedule stretched out over 10 years as proposed in ref. document totally irresponsible.

Dr. Rudolf H. Hausler

Figure 1

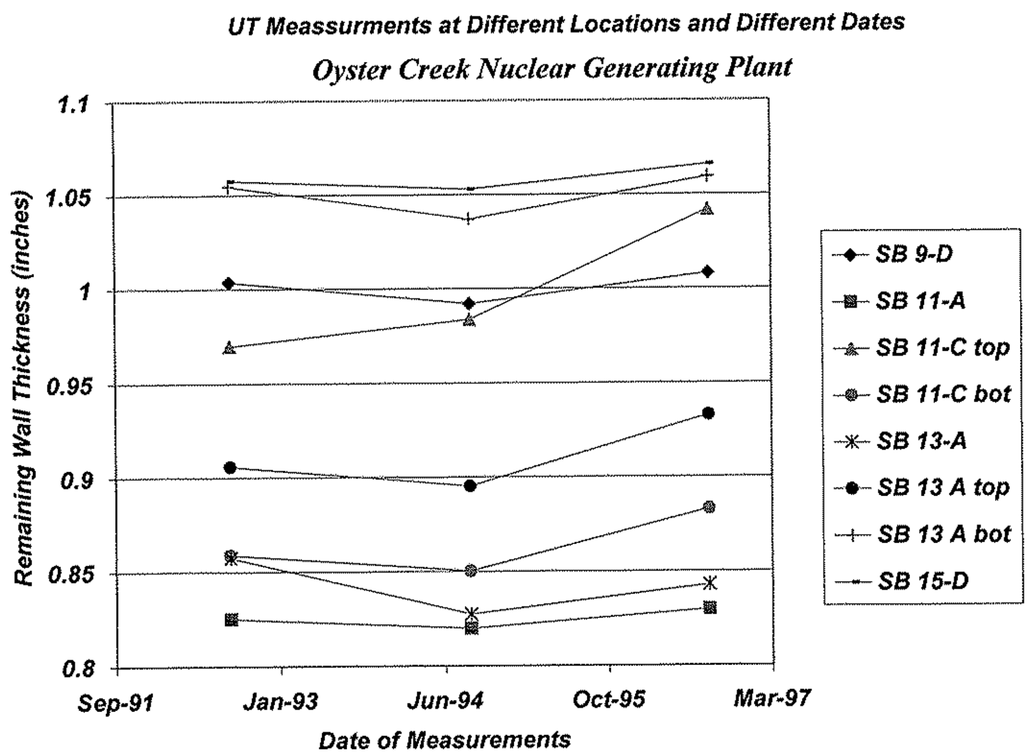
UT Meassurments at Different Locations and Different Dates
Oyster Creek Nuclear Generating Plant
Remaining Wall Thickness (inches)
1.1
1.05
1
0.95
0.9
0.85
0.8
Sep-91
Jan-93
Jun-94
Oct-95
Mar-97
Date of Measurements
SB 9-D
SB 11-A
SB 11-C top
SB 11-C bot
SB 13-A
SB 13 A top
SB 13 A bot
SB 15-D

EXHIBIT K

Christian Science Monitor

April 19, 2006, Wednesday

SECTION: USA; Pg. 3

LENGTH: 987 words

HEADLINE: Should oldest US nuke plant stay on line?

BYLINE: Alexandra Marks Staff writer of The Christian Science Monitor

DATELINE: NEW YORK

HIGHLIGHT:

New Jersey says the plant is too vulnerable to terrorist attack to have its license renewed.

BODY:

In what could be a precedent-setting case, **New Jersey** and a coalition of citizens are fighting renewal of the license for the nation's oldest operating **nuclear** power plant.

Their concern: The structural design of the 1960s-era **Oyster Creek nuclear** generating station is a security risk because, among other things, it stores highly radioactive spent fuel rods above ground. They argue that makes it vulnerable in the event of a terrorist attack from the air.

Their contention, if proved, could lead the **Nuclear** Regulatory Commission to deny for the first time a **nuclear** generating station's request for a license renewal after its original 40-year license expires. It could also set a new standard for the NRC, which currently does not take terrorism into account when it decides whether to renew a **nuclear** plant's license.

In fact, the NRC recently ruled the "possibility of a terrorist attack ... is speculative" and therefore "beyond the scope" of relicensing proceedings.

The state of **New Jersey** is appealing that ruling, arguing that the threat of terrorism is not speculative at all but a danger that must be addressed. Terrorism experts agree.

"From a policy perspective, it's absolutely critical that the relicensing procedures take into account the vulnerability from man-made attacks," says Michael Greenberger, director of the University of Maryland's Center for Health and Homeland Security in Baltimore. "It's the height of folly ... for the [NRC] to say that it's not going to consider seriously the vulnerability of the oldest plants when everybody knows these facilities are high-level targets."

Oyster Creek is located in the densely populated Jersey Shore, a fast-growing area in the most densely populated state. That's one of the things that prompted Janet Tauro to join the fight to close the plant when its license expires in 2009.

"It's an obsolete design," she says. "There are almost 3,000 pounds of highly radioactive rods stored 70 feet in the air in a cooling pool of water protected only by a thin metal roof. It's way too vulnerable."

The owners of **Oyster Creek,** who have applied for a license renewal to operate another 20 years, deny the plant is obsolete and note the metal roof above the spent fuel rods is "a heavily reinforced steel structure."

"**Oyster Creek** is required to meet every single safety standard and regulation as every plant, no matter what the age," says **Oyster Creek** spokesman Pete Resler. "The station has been continually upgraded: We put in the most modern safety systems and equipment."

The clash hints at the challenge of addressing electricity needs as well as environmental concerns about greenhouse gases, which **nuclear** power plants don't emit. It also shows the challenges faced in this post-9/11 world by the NRC, which has recently come under fire from some members of Congress for what they see as not taking the threat of terrorism seriously enough.

NRC officials say they do take the threat extremely seriously and since 9/11 have taken "numerous steps" to ensure all plants are secure. It's something that is dealt with on a daily basis, they say, not in the context of whether a plant is too old to operate safely - which is what the relicensing procedure is designed to address.

"The fact remains that security at a **nuclear** power plant is independent of the length of its license. It doesn't matter if a plant operates for five years, 15, or 20: It will have to meet the security requirements that are placed upon it by the NRC," says Scott Brunell, an NRC spokesman. "To attempt to address security for a plant that is seeking relicensing is an attempt to judge a plant on a snapshot that is not going to apply in the future one way or another."

The State of **New Jersey** sees things very differently. It argues that **Oyster Creek's** age and design are the very things that present serious security risks, and that those issues can best be addressed during the relicensing process. In its appeal of the NRC ruling, **New Jersey's** attorney general calls the design "comparatively unreliable and vulnerable." The appeal also argues that a terrorist attack is not just speculative and that the NRC's own actions prove that.

"There would be no need for the Commission to require extensive steps to guard against terrorist attack if the chances of an attack were only speculative," the appeal states.

The NRC has yet to rule on the appeal. In the meantime, a coalition of citizens' groups is lending its support to the state's stand.

"Security's not just a day-to-day concern. In this case, it is a structural issue as well," says Richard Webster, an attorney at the Rutgers Environmental Law Clinic in Newark, which represents the citizens' coalition. "The structure of the plant doesn't protect against this type of attack. If it was being built from scratch today, it could be designed to protect against one."

Oyster Creek officials disagree, saying their plant can sustain a direct hit by an aircraft.

"We're certainly able to defend the facility," says Mr. Resler. "The Electric Power Research Institute [a nonprofit company backed by the power industry] also did a study and found that even if such an event did occur, which is an extremely remote possibility, that there would not be a catastrophic release of radioactivity. These structures are designed for safety with multiple barriers to protect the fuel."

But Ms. Tauro is not convinced. She points to a recent study by the National Academy of Sciences' National Research Council, done at the request of Congress. It found that "successful terrorist attacks on spent fuel pools [at some **nuclear** power plants,] though difficult, are possible" and that "a propagating fire in a pool could release large amounts of radioactive materials."

"**Oyster Creek** is within 10 minutes of seven airports, both local and major," she says. "This plant should be retired. Its time has come."

LOAD-DATE: April 19, 2006

EXHIBIT L

Safety and Security of Commercial Spent Nuclear Fuel Storage
Public Report, National Academy of Sciences, April 2005

Report's Highlights

Prepared by the Nuclear Security Coalition, April 2005

The U.S. Congress asked the National Academies to analyze the safety and security of commercial spent nuclear storage in the United States. The highlights of the report are as follows. The full report is on their website http://www.nap.edu/books/0309096472/html/

VULNERABILITY TO ATTACK & CONSEQUENCES

1. Spent fuel pools are necessary at all operating nuclear reactors to store recently discharged fuel. Freshly discharged spent fuel generates too much decay heat to be passively air cooled in casks. This fuel must be stored in a pool of cold water for at least one year before being moved to dry casks.

2. Terrorist attacks on spent fuel pools are possible ---a credible threat.

"Finding 2A: Spent fuel storage facilities cannot be dismissed as targets for such attacks because it is not possible to predict the behavior and motivations of terrorists, and because of the attractiveness of spent fuel as a terrorist target given the well known public dread of radiation…The committee judges that attacks by knowledgeable terrorists with access to appropriate technical means are possible."[1]

"Terrorists view nuclear power plant facilities as desirable targets because of the large inventories of radionuclides they contain. The committee believes that knowledgeable terrorists might choose to attack spent fuel pools because: (1) at U.S. commercial nuclear power plants, these pools are less well protected structurally than reactor cores; (2) they typically contain inventories of medium – and long-lived radionuclides that are several times greater than those in individual reactor cores[2]*."*

"A loss-of-pool-coolant event resulting from damage or collapse of the pool could have severe consequences. Severe damage of the pool wall could potentially result from several types of terrorist attacks, for instance: (1) Attacks with large civilian aircraft; (2) Attacks with high-energy weapon; Attacks with explosive charges[3]*."*

3. If a terrorist attack on the spent fuel pool leads to a zirconium cladding fire, it could result in large amounts of radioactive material spreading hundreds of miles.

"Finding 3B –… a terrorist attack that partially or completely drained a spent fuel pool could lead to a propagating zirconium cladding fire and the release of large quantities of radioactive materials to the environment. Details are provided in the committee's classified report[4]*."*

"Such (zirconium cladding) fires would create thermal plumes that could potentially transport radioactive aerosols hundreds of miles downwind under appropriate atmospheric conditions.[5]*"*

"The excess cancer estimates …to between 2,000 and 6,000 cancer deaths"[6]

GE BOILING WATER MARK I & MARK II UNITS –MORE VULNERABLE

4. Vulnerability to attack differs according to the plant's design; GE Boiling Water Mark I and Mark II Units are more vulnerable to attack.

"The potential vulnerabilities of spent fuel pools to terrorist attack are plant specific ... there are substantial differences in the designs of spent fuel pool that make them more or less vulnerable to certain types of attack.[7] "

"The spent fuel pool, (GE Mark I BWR reactors) is located in the reactor building well above ground level. Most designs have thin steel superstructures. The superstructures and pools were not, however, specifically designed to resist terrorist attacks[8]. "

"The vulnerability of a spent fuel pool to terrorist attack depends in part on its location with respect to ground level as well as its construction. Pools are potentially susceptible to attacks from above or the sides depending on their elevation with respect to grade and the presence of surrounding shielding structures[9]. "

"Mark I and II BWR plants are located above grade and are shielded by at least one exterior building wall. Some pools are also shielded by the reactor buildings. Some pools are also shielded by "significant" surrounding structures, and some have supplemental floor and column supports. "[10]

NAS RECOMMENDS TAKING THE FOLLOWING STEPS IMMEDIATELY

5. NAS recommends taking the following steps immediately to reduce the likelihood of zirconium cladding fire: rearranging spent fuel assemblies in the pool; limiting offloads of full reactor cores; and installing water-spray systems capable of operating when high radiation limits worker access and the pool or overlying building are damaged.

"Finding 3C: It appears to be feasible to reduce the likelihood of a zirconium fire following the loss-of-pool-coolant event using readily implemented measures. The following measures appear to have particular merit:

- *<u>Reconfiguring the spent fuel pools</u> (i.e. redistribution of high decay heat assemblies so that they are surrounded by low decay-heat assemblies) to more evenly distribute decay heat loads...The potential for zirconium cladding fires can be reduced substantially by surrounding freshly discharged fuel assemblies with older spent fuel assemblies in "checkerboard" fashion. The analyses suggest that such arrangements might even be more effective for reducing the potential for zirconium cladding fires than removing this older spent fuel from the pools. However, these advantages have not been unequivocally demonstrated by modeling and experiments.*
- *<u>Limiting the frequency of offloads of full reactor cores</u> into the spent fuel pools, requiring longer shut downs of the reactor before any fuel is offloaded, and providing enhanced security when such offloads must be made.*
- *Development of a redundant and diverse response system to mitigate loss-of-pool-coolant events. Any <u>mitigation system, such as a spray cooling system</u>, must be capable of operation even when the pool is drained (which would result in high radiation fields and limit worker access to the pool) and the pool or overlying building, including equipment attached to the roof or walls, are severely damaged.[11] "*

"The (spray cooling system) second measure...may not be needed at all plants, particularly in plants in which the spent fuel pools are located below grade or are protected from external line-of-sight attacks by exterior walls and other structures.[12] "

NAS discusses conditions that would make it difficult to take some of these mitigating steps; hence multiple, redundant and diverse measures required. Page 55 –

"Of course, damage to the pool and high radiation fields could make it difficult to take some of these mitigative measures. Multiple redundant and diverse measures may be required so that more than one remedy is available to mitigate loss-of-pool-coolant event, especially when access to the pool is limited by damage or high radiation fields.[13]"

DRY CASK STORAGE

6. Dry cask storage has inherent security advantages over spent fuel storage, but it can only be used to store older fuel –fuel that has cooled 3-5 years in the pool before placed in dry casks licensed today.

"Safety and Security Advantages of Dry Cask Storage Versus Wet Pool Storage -Finding 4D: Dry cask storage for older, cooler spent fuel has two inherent advantages over pool storage: (1) It is a passive system that relies on natural air circulation for cooling; and (2) it divides the inventory of that spent fuel among a large number of discrete, robust containers. These factors make it more difficult to attack a large amount of spent fuel at one time and also reduce the consequences of such attacks[14]."

7. There are no large security differences among different storage designs; all designs are more secure than pools but still vulnerable.

Findings 4A: "All storage casks designs are vulnerable to some types of terrorist attacks, but the quantity of radioactive material releases predicted from such attacks is relatively small. These releases are not easily dispersed in the environment[15]."

8. Additional steps may be taken to make dry casks less vulnerable

"Finding 4B: Additional steps can be taken to make dry casks less vulnerable to potential terrorist attacks.[16]"

"Recommendation: "The Nuclear Regulatory Commission should consider using the results of the vulnerability analyses for possible upgrades of requirements in 10 CFR 72 for dry casks, specifically to improve their resistance to terrorist attacks.[17]"

"In the committee's opinion, there are several, relatively simple steps that could be taken to reduce the likelihood of releases of radioactive material from dry casks in the event of a terrorist attack.

- *Additional surveillance could be added to dry cask storage facilities to detect and thwart ground attacks.*
- *Certain types of cask systems could be protected against aircraft strikes by partial earthen berms. Such berms also would deflect the blasts from vehicle bombs.*
- *Visual barriers could be placed around storage pads to prevent targeting of individual casks by aircraft or standoff weapons. These would have to be designed so that they would not trap jet fuel in the event of an aircraft attack.*
- *The spacing of vertical casks on the storage pads can be changed, or spacers (shims) can be placed between the casks, to reduce the likelihood of cask-to-cask interactions in the event of an aircraft attack.*
- *Relatively minor changes in the design of newly manufactured casks could be made to improve their resistance to certain types of attack scenarios.[18]"*

9. Based on plant-specific vulnerability analyses, NRC might determine moving spent fuel from pools into dry cask storage to reduce risk. However, NAS was not asked by Congress to recommend whether transfer of spent fuel rods from pools to a system of dry casks should be accelerated; therefore they could not provide specific recommendations on this issue.

"Finding 4E: Depending on the outcome of plant-specific vulnerability analyses described in the committee's classified report, the Nuclear Regulatory Commission might determine that earlier movements of spent fuel from pools into dry cask storage would be prudent to reduce the potential consequences of terrorist attacks on pools at some commercial nuclear plants. The statement of task directs the committee to examine the risks of spent fuel storage options and alternatives for decision makers; not to recommend whether any spent fuel should be transferred from pool storage to cask storage. In fact, there may be some commercial plants that, because of pool designs or fuel loadings, may require some removal of spent fuel from their pools. If there is a need to remove spent fuel from the pools it should become clearer once the vulnerability and consequence analyses described in the classified report are completed[19]."

NAS CALLS FOR MORE ANALYSES

10. NAS expresses concern over NRC's slow pace analyses.

"...the Nuclear Regulatory Commission's analyses of spent fuel storage vulnerabilities have not yet been completed and actions to reduce vulnerabilities ...have not yet been taken. Moreover, some important additional analyses remain to be done. The slow pace in completing this work is of concern given the enormous consequences as described elsewhere in this report[20]"

11. Additional <u>independent</u> analyses are needed to understand more fully the vulnerabilities and consequences of events that could lead to propagating zirconium fires.

"Finding 2C Recommendation: "Although the committee did not specifically investigate the effectiveness and adequacy of improved surveillance and security measures for protecting stored spent fuel, an assessment of current measures should be performed by an independent organization[21]."

12. Because vulnerability is <u>plant specific</u>, the committee recommended that plant-by-plant vulnerability analyses be performed.

"Finding 3 D: The potential vulnerabilities of spent fuel pools to terrorist attacks are plant-design specific. Therefore specific vulnerabilities can only be understood by examining the characteristics of spent fuel storage at each plant. As described in the classified report, there are substantial differences in the designs of PWR and BWR spent fuel pools. PWR pools tend to be located near or below grade, whereas BWR pools typically are located well above grade but are protected by exterior walls and other structures. In addition, there are plant-specific differences among BWRs and PWRs that would increase or decrease the vulnerabilities of the pools to various kinds of terrorist attacks, making generic conclusions difficult.[22]"

13. The report calls on the <u>NRC</u> to conduct additional analyses to obtain better understanding of potential risks to ensure operators take prompt and effective measures to reduce consequences of attacks.

"The analyses carried out for the Nuclear Regulatory Commission (described in the committee's classified report) do not consider maximum credible scenarios... To be judged a "credible" scenario, the terrorist must be able to carry it out as designed.[23]"

"Finding 3E: The NRC and independent analysts have made progress in understanding some vulnerability of spent fuel pools to certain terrorist attacks and the consequences of such attacks for releases of radioactivity to the environment. However, additional work on specific issues is needed urgently...The work to date ...has not been sufficient to adequately understand the vulnerabilities and consequences."

"Recommendation: The Nuclear Regulatory Commission should undertake best -estimate analyses to more fully understand the vulnerabilities and consequences of loss-of-pool-coolant events that could lead to a zirconium cladding fire. Based on these analyses, the Commission should take appropriate actions to address any significant vulnerabilities identified. The committee provides details on additional analyses that should be carried out in the classified report.[24] *"*

Consequence analyses should address the following questions

"The consequence analyses should address the following questions:

- *To what extent would such attacks damage the spent fuel in the pool, and what would be the thermal consequences of such damage?*
- *Is it feasible to reconfigure the spent fuel within pools to prevent zirconium cladding fires given the actual characteristics (i.e., heat generation) of spent fuel assemblies in the pool, even if the fuel were damaged in an attack? Is there enough space in the pools at all commercial reactor sites to implement fuel reconfiguration?*
- *In the event of a localized zirconium cladding fire, will such rearrangement prevent its spread to the rest of the pool?*
- *How much spray cooling is needed to prevent zirconium cladding fires and prevent propagation of such fires? Which of the different options for providing spray cooling are effective under attack and accident conditions?"*[25]

Analyses must be performed that accounts for the full range of variation in spent fuel pool designs

"Sensitivity analyses should be undertaken to account for the full range in variation in spent fuel pool designs (e.g., rack designs, capacities, spent fuel burn-ups and ages) at U.S. commercial nuclear power plants.[26] *"*

NAS - SECURITY NEGATIVELY IMPACTED BY NRC'S SECRECY

14. Current classification and security practices are impeding sharing valuable information between the NRC, <u>nuclear reactor operators, and system vendors</u>, negatively impacting constructive feedback and ultimately security.

"Finding 5A: Security restrictions on sharing of information and analyses are hindering progress in addressing potential vulnerabilities of spent fuel storage to terrorist attacks."

Recommendation: "The Nuclear Regulatory Commission should improve the sharing of pertinent information on vulnerability and consequence analyses on spent fuel storage with nuclear power plant operators and dry cask storage system vendors on a timely basis.[27] *"*

15. <u>More constructive interaction with the public and independent analysts</u> would increase confidence in the NRC and industry and reduce vulnerability.

"The committee also believes that the public is an important audience for the work being carried out to assess and mitigate vulnerabilities to spent fuel storage facilities. While it is inappropriate to share all information publicly, more constructive interaction with the public

and independent analysts could improve the work being carried out, and also increase confidence in the nuclear regulatory Commission and industry decisions and actions to reduce the vulnerability of spent fuel storage to terrorist threats[28]"

16. On several important questions, NAS was unable to obtain enough information from the NRC to assess effectiveness security at commercial nuclear reactors. Therefore the committee recommended assessment of security measures is undertaken by an organization independent of the NRC and industry.

"The Commission staff declined to provide a formal briefing to the committee on the DBT for radiological sabotage, asserting that the committee did not have a need to know this information.[29]"

NRC & NUCLEAR INDUSTRY SAY THEY ARE NOT RESPONSIBLE FOR SECURITY AGAINST TERRRORIST ATTACKS – CITIZENS ASK, WHO IS TAKING RESPONSIBILITY?

"To the committee's knowledge, there are currently no requirements in place to defend against the kinds of larger-scale; premeditated, skillful attacks that were carried out on September 11, 2001, whether or not a commercial aircraft is involved. Staff from the NRC and representatives from the nuclear industry repeatedly told the committee that they view detecting, preventing, and thwarting such attacks as the federal government's responsibility[30]*."*

End Notes

[1] 'Safety and Security of Commercial Spent Fuel Storage," National Research Council of the National Academy of Sciences, Public Version, April, 2005, p. 4
[2] NAS Ibid, p. 36
[3] NAS, Ibid, p. 49
[4] NAS, Ibid, p. 6
[5] NAS, Ibid, p.50
[6] NAS, Ibid, p.45
[7] NAS, Ibid, p. 6
[8] NAS, Ibid, p. 41
[9] NAS, Ibid, p. 43
[10] NAS, Ibid, p. 43
[11] NAS, Ibid, p.6, 57
[12] NAS, Ibid, p. 59
[13] NAS, Ibid, p. 55
[14] NAS, Ibid, p. 8
[15] NAS, Ibid, p.7
[16] NAS, Ibid, p. 7
[17] NAS, Ibid, p. 7
[18] NAS, Ibid, p. 68
[19] NAS, Ibid, p. 8
[20] NAS, Ibid, p. 75
[21] NAS Ibid, p. 5, 36
[22] NAS, Ibid, p. 6, 58
[23] NAS, Ibid, p. 28
[24] NAS, Ibid, p. 6, 58
[25] NAS, Ibid, p. 59
[26] NAS, Ibid, p. 59
[27] NAS, Ibid, p. 8
[28] NAS, Ibid, p. 9
[29] NAS, Ibid, p. 31
[30] NAS, Ibid, p. 47

EXHIBIT M

Powering Our Community for Today and Tomorrow

Used Fuel Security Fact Sheet

Independent Used Fuel Storage Installation (ISFSI)

Spent nuclear fuel decays over time and is less radioactive than the fuel in the reactor. The Used Fuel can be safely cooled by air circulating through their storage vaults. The radioactive material is contained in ceramic fuel pellets, which have a melting point of 5,000 degrees, and are inside metal rods, with a melting point of 3,000 degrees. The fuel assemblies are placed in thick stainless steel, leak-tight canisters that are welded shut.

The loaded storage canister is transported in a thick steel cask that is tightly sealed and then placed into a reinforced concrete storage vault. The vault walls are approximately three feet thick and are designed to withstand natural or man-made events, including, but not limited to, earthquakes, hurricanes, tornados and tornado generated projectiles, and floods.

The ISFSI facility at Oyster Creek is protected from sabotage and intrusion with measures that are equivalent to those for the plant itself. This "defense in depth" system makes dry fuel storage a safe and secure way to temporarily store spent fuel.

All project planning, construction, and transportation related to Used Fuel management is designed to account for and protect against multiple disaster-level events as well as radiological sabotage. An analysis post 9/11 demonstrates the ISFSI can withstand the impact of a large commercial aircraft without breaching the canister barrier.

The ISFSI would not be an attractive target for terrorists. The vaults are low structures that take up very little space. The high level of security now in effect in the airspace, waterways and ground surrounding all nuclear plants today make it even more secure.

The storage area is designed to hold Used Fuel from Oyster Creek Generating Station alone. NRC regulations and township guidelines prohibit any other generating station from storing fuel at the site.

Congress has mandated that the Department of Energy find a national Used Fuel repository. AmerGen's goal is for Oyster Creek's waste to go to Yucca Mountain when it is approved and constructed by the federal government. The draft schedule has the Yucca Mountain repository receiving Used Fuel in 2010 at the earliest.

Used Fuel Pool

The Used Fuel pool is accessed from the 119-foot elevation at the top of the Reactor Building at Oyster Creek. The walls around this elevation are insulated metal siding mounted to steel framing. It is misleading to consider this as a "thin steel structure" as the steel framing is of substantial I-Beam construction that supports the reactor building gantry crane.

The pool extends down through the reactor building enclosed in massive concrete walls integrated with the reactor containment structure and supported on concrete columns. The pool is approximately 40 feet deep providing in excess of 25 feet of water above the top of the stored fuel.

The crash of an airplane into the reactor building 119-foot elevation would involve damage to the reactor building gantry crane and equipment, both from impact and fire, but it is highly unlikely the crash would cause any significant damage to the fuel stored in the Used Fuel pool. The steel framing, the pool's massive concrete structure and supporting columns would protect the pool from impact damage and the contained water would provide protection to the fuel from impact and fire effects.

oystercreek
Clean, Safe, Reliable Energy For The Future

EXHIBIT N

Division of Environmental Safety and Health
Radiation Protection and Release Prevention Element
PO Box 415
Trenton, NJ 08625-0415
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July 30, 2004

Mr. Hubert Miller
Regional Administrator
U.S. Nuclear Regulatory Commission
475 Allendale Rd.
King of Prussia, PA 19406-1415

Subject: Effects of Aircraft Impact on Spent Fuel Pools in New Jersey

Dear Mr. Miller:

Since the September 11, 2001 tragedy, nuclear power generation facilities have been the subject of numerous evaluations related to the prevention of and emergency response to possible terrorist actions, including the use of aircraft as a destructive device. The State of New Jersey through our Radiation Protection and Release Prevention Element – Bureau of Nuclear Engineering (BNE) has been studying developments in this area.

Recently, two technical studies related to the effects of aircraft impact on Spent Fuel Pools have been performed by private parties and were reviewed by the NRC. These two studies were the Nuclear Energy Institute (NEI)/Electric Power Research Institute (EPRI) Study: "Deterring Terrorism: Aircraft Crash Impact Analyses Demonstrate Nuclear Power Plant's Structural Strength," issued March 2003 (hereafter referred to as the NEI Study) and the paper, "Reducing the Hazards from Stored Spent Power-Reactor Fuel in the United States," April 21, 2003, Robert Alvarez, et al., published in Science and Global Security, Spring 2003 (hereafter referred to as the Alvarez Paper).

NEI considers the details of their study, submitted to the NRC for review, to be security sensitive. Accordingly, New Jersey did not have access to the complete report and could not conduct a detailed independent review as to the study's validity for nuclear facilities located in the state. However, NEI has made public sufficient information to conclude that the study was limited to the evaluation of the impact of a Boeing 767-400 airplane into containment buildings, used fuel storage pools, used fuel "Dry" storage facilities and used fuel transportation containers. The NEI Study does not appear to have taken into account the thermal and structural consequences and collateral damage of the explosion and resulting fire that would also occur from the impact of a commercial aircraft. In addition, it appears that the structural models used to evaluate impact damage were based on "representative" (not site-specific), structures, which were considered by NEI to be typical to those that exist across the nuclear power industry.

The Alvarez Paper was available to New Jersey as was the NRC staff's review and comments. This paper focused on the potential generic vulnerabilities of spent fuel pools to terrorist attack. The paper also details the possible public safety and environmental consequences should such attacks successfully occur. Included in this paper were conservative estimates of the radiological release should a spent fuel zircaloy cladding fire occur due to a significant breach of a spent fuel pool. The paper states, "The long-term land-contamination consequences of such an event could be significantly worse than those from Chernobyl". The paper further states (in reference to Chernobyl), "The total area of this radiation-control zone is huge: 10,000 km^2, equal to half the area of the State of New Jersey. During the following decade, the population of this area declined by almost half because of migration to areas of lower contamination".

As you are aware, New Jersey is the home to four operating nuclear power reactors located at two separate generating sites. Three power reactors, "Hope Creek", "Salem Unit 1" and "Salem 2", are located on the Delaware River at the PSE&G Artificial Island Site and the fourth reactor, "Oyster Creek", is located near Barnegat Bay and the Atlantic coastline at the AmerGen Oyster Creek Site.

New Jersey is especially concerned about the vulnerability of the Oyster Creek spent fuel pool (OCSFP) to a terrorist attack using a commercial aircraft. This concern is based, in part, on the structural design of the superstructure of the building which encloses the OCSFP (metal siding, concrete roof panels, high collapse potential for this scenario), the location of the pool in the building (high elevation, near an outside wall, pool surface open to superstructure), the relatively unimpeded flight path to the fuel pool location (located on an open coastal plane with minimal surrounding obstructions to fuel pool wall), and, most importantly, the lack of a comprehensive site-specific evaluation for this terrorist aircraft impact scenario which addresses the collective consequences of impact and resulting explosion, fire (including thermal gradients through fuel pool concrete), and probable structural collapse on the OCSFP and fuel assemblies.

Additionally, the site-specific radiological release (including a timeline for the expected release) resulting from this terrorist aircraft impact scenario needs to be quantified by the NRC and provided to New Jersey for emergency planning preparation to insure that the safety of the residents of New Jersey and first responders can be maintained. New Jersey requests this information be provided expediently.

Since New Jersey is not aware of any site-specific evaluation of the OCSFP that addresses these issues, it is requesting that the USNRC provide detailed technical assurance documenting the basis that the above mentioned concerns have been rigorously addressed and that the safety of residents of New Jersey and the environment can be maintained should a 9-11 style terrorist attack occur at Oyster Creek.

New Jersey is also requesting that some provision be made so that authorized representatives of the State of New Jersey, Bureau of Nuclear Engineering, can be granted access to review any and all documentation which is used by the NRC as the basis for concluding that terrorist threats to nuclear power facilities do not represent a risk to New Jersey residents.

If you need additional information, please contact Mr. Kent Tosch, Manager of the Bureau of Nuclear Engineering, at (609) 984-7701.

Sincerely,

Jill Lipoti, Ph.D., Assistant Director
Radiation Protection Program and Release Prevention

EXHIBIT O

A paper by Alvarez et al. (2003a; see also Thompson, 2003) took the analyses in NUREG-1738 to their logical ends in light of the September 11, 2001, terrorist attacks: Namely, what would happen if there were a loss-of-pool-coolant event that drained the spent fuel pool? Such an event was not considered in NUREG-1738, but the analytical results in that study were presented in a manner that made such an analysis possible.

Alvarez and his co-authors concluded that such an event would lead to the rapid heat-up of spent fuel in a dense-packed pool to temperatures at which the zirconium alloy cladding would catch fire and release many of the fuel's fission products, particularly cesium-137. They suggested that the fire could spread to the older spent fuel, resulting in long-term contamination consequences that were worse than those from the Chernobyl accident. Citing two reports by Brookhaven National Laboratory (BNL, 1987, 1997), they estimated that between 10 and 100 percent of the cesium-137 could be mobilized in the plume from the burning spent fuel pool, which could cause tens of thousands of excess cancer deaths, loss of tens of thousands of square kilometers of land, and economic losses in the hundreds of billions of dollars. The excess cancer estimates were revised downward to between 2000 and 6000 cancer deaths in a subsequent paper (Beyea et al., 2004) that more accurately accounted for average population densities around U.S. power plants.

Alvarez and his co-authors recommended that spent fuel be transferred to dry storage within five years of discharge from the reactor. They noted that this would reduce the radioactive inventories in spent fuel pools and allow the remaining fuel to be returned to open-rack storage to allow for more effective coolant circulation, should a loss-of-pool-coolant event occur. The authors also discussed other compensatory measures that could be taken to reduce the consequences of such events.

The Alvarez et al. (2003a) paper received extensive attention and comments, including a comment from the Nuclear Regulatory Commission staff (USNRC, 2003a; see Alvarez et al., 2003b, for a response). None of the commentators challenged the main conclusion of the Alvarez et al. (2003a) paper that a severe loss-of-pool-coolant accident might lead to a spent fuel fire in a dense-packed pool. Rather, the commentators challenged the likelihood that such an event could occur through accident or sabotage, the assumptions used to calculate the offsite consequences of such an event, and the cost-effectiveness of the authors' proposal to move spent fuel into dry cask storage. One commentator summarized these differences in a single sentence (Benjamin, 2003, p. 53): "In a nutshell, [Alvarez et al.] correctly identify a problem that needs to be addressed, but they do not adequately demonstrate that the proposed solution is cost-effective or that it is optimal."

The Nuclear Regulatory Commission staff provided a briefing to the committee that provides a further critique of the Alvarez et al. (2003a) analysis that goes beyond the USNRC (2003a) paper. Commission staff told the committee that the NUREG-1738 analyses attempted to provide a bounding analysis of current and conceivable future spent fuel pools at plants undergoing decommissioning and therefore relied on conservative assumptions. The analysis assumed, for example, that the pool contained an equivalent of three-and-one-half reactor cores of spent fuel, including the core from the most recent reactor cycle. The staff also asserted that NUREG-1738 did not provide a realistic analysis of consequences. Commission staff concluded that "the risks and potential societal cost of [a] terrorist attack on spent fuel pools do not justify the complex and costly measures proposed in Alvarez et al. (2003) to move and store 1/3 of spent fuel pools [sic] inventory in dry storage casks."[8]

The committee provides a discussion of the Alvarez et al. (2003a) analysis in its classified report. The committee judges that some of their release estimates should not be dismissed.

The 2003 Nuclear Regulatory Commission (USNRC, 2003b) staff publication NUREG-0933, *A Prioritization of Generic Safety Issues*,[9] discusses beyond-design-basis accidents in spent fuel pools. The study draws some of the same consequence conclusions as the Alvarez et al. (2003a) paper. It notes that in a dense-packed pool, a zirconium cladding fire "would probably spread to most or all of the spent fuel pool" (p. 1). This could drive what the report refers to as "borderline aged fuel" into a molten condition leading to the release of fission products comparable to molten fuel in a reactor core.

The NUREG-0933 report (USNRC, 2003b) summarizes technical analyses of the frequencies of severe accidents for three BWR scenarios. The report concludes that the greatest risk is from a beyond-design-basis seismic event. While the consequences of such accidents are considerable, the report concludes that their frequencies are no greater than would be expected for reactor core damage accidents due to seismic events beyond the design basis safe shutdown earthquake.

An analysis of spent fuel operating experience by the Nuclear Regulatory Commission staff (USNRC, 1997) showed that several accidental partial-loss-of-pool-coolant events have occurred as a result of human error. Two of these involved the loss of more than 5 feet of water from the pool, but none had serious consequences. Nevertheless, Commission staff suggested that plant-specific analyses and corrective actions should be taken to reduce the potential for such events in the future.

It is important to recognize that with the exception of the Alvarez et al. (2003a) paper, all of the previous U.S. work reviewed by the committee has focused on safety risks, not security risks. The Nuclear Regulatory Commission analyses of spent fuel storage vulnerabilities were not completed by the time the committee finalized its information gathering for this report, but the committee did receive briefings on this work. In addition, analyses have been undertaken of external impacts on power plant structures by aircraft for the few commercial power plants that are located close enough to airports to consider hardening of the plant design to resist accidental aircraft crashes. These analyses were done as part of the plants' licensing safety analyses. The committee did not look further into these few plants because the aircraft considered were smaller and the impact velocities considered were much lower than those that might be brought to bear in a well-planned terrorist attack.

The committee did learn about work to assess the risks of spent fuel storage to terrorist attacks in Germany (see Appendix C for a description). However, the details of this work are classified by the German government and therefore are unavailable to the

[8] The quote is from a PowerPoint presentation made by Nuclear Regulatory Commission staff to the committee at one of its meetings.
[9] NUREG-0933 is a historical record that provides a yearly update of generic safety issues. It does not provide any additional technical analysis of these issues.