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

November 16, 2006

MEMORANDUM TO: ACRS Members

FROM: Cayetano Santos Jr., Team Leader
Technical Support Staff, ACRS

Cayetano Santos Jr.

SUBJECT: CERTIFICATION OF THE MINUTES OF THE ACRS SUBCOMMITTEE MEETING ON THE OYSTER CREEK GENERATING STATION LICENSE RENEWAL APPLICATION, OCTOBER 3, 2006 - ROCKVILLE, MARYLAND

The minutes of the subject meeting were certified on November 12, 2006, as the official record of the proceedings of that meeting. A copy of the certified minutes is attached.

Attachment: As stated

cc w/o Attachment: J. Larkins
M. Snodderly
S. Duraiswamy
M. Junge

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

MEMORANDUM TO: Cayetano Santos Jr., Team Leader
Technical Support Staff, ACRS

FROM: Otto Maynard, Chairman
ACRS Plant License Renewal Subcommittee

SUBJECT: CERTIFICATION OF THE MINUTES OF THE ACRS SUBCOMMITTEE MEETING ON THE OYSTER CREEK GENERATING STATION LICENSE RENEWAL APPLICATION, OCTOBER 3, 2006 - ROCKVILLE, MARYLAND

I hereby certify, to the best of my knowledge and belief, that the minutes of the subject meeting on October 3, 2006, are an accurate record of the proceedings for that meeting.

Otto Maynard 11/12/06

Otto Maynard, Date
Plant License Renewal Subcommittee Chairman

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

October 31, 2006

MEMORANDUM TO: Otto Maynard, Chairman
ACRS Plant License Renewal Subcommittee

FROM: Cayetano Santos Jr., Team Leader Cayetano Santos Jr.
Technical Support Staff, ACRS

SUBJECT: WORKING COPY OF THE MINUTES OF THE ACRS SUBCOMMITTEE MEETING ON THE OYSTER CREEK GENERATING STATION LICENSE RENEWAL APPLICATION, OCTOBER 3, 2006 - ROCKVILLE, MARYLAND

A working copy of the minutes for the subject meeting is attached for your review. Please review and comment on them at your earliest convenience. If you are satisfied with these minutes please sign, date, and return the attached certification letter.

Attachments: Certification Letter
Minutes (DRAFT)

cc w/o Attachment: J. Larkins
M. Snodderly
S. Duraiswamy
M. Junge

DRAFT Issued 10/31/06
CERTIFIED by O. Maynard on 11/12/06

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
MINUTES OF ACRS PLANT LICENSE RENEWAL SUBCOMMITTEE MEETING
OYSTER CREEK GENERATING STATION
OCTOBER 3, 2006
ROCKVILLE, MARYLAND

On October 3, 2006, the Plant License Renewal Subcommittee held a meeting in Room T2B3, 11545 Rockville Pike, Rockville, Maryland, to review the License Renewal Application (LRA) for the Oyster Creek Generating Station (OCGS) and the associated draft Safety Evaluation Report (SER) with Open Items.

The meeting was open to the public. Mr. Paul Gunter of the Nuclear Information Resource Service and Mr. Richard Webster of the Rutgers Environmental Law Clinic made oral statements following the formal presentations by AmerGen and the NRC staff. Mr. Bill Herring submitted written comments in an email dated October 3, 2006. Mr. Cayetano Santos was the Designated Federal Official for this meeting. The meeting convened at 1:30 pm and adjourned at 6:10 pm on October 3, 2006.

ATTENDEES:

ACRS MEMBERS/STAFF

Otto Maynard, Chairman
John Sieber, Member
Graham Wallis, Member
Said Abdel-Khalik, Member
Cayetano Santos Jr., ACRS Staff
William Shack, Member
Mario Bonaca, Member
Joseph Armijo, Member
John Barton, Consultant
Michael Junge, ACRS Staff

NRC STAFF/PRESENTERS

D. Ashley, NRR
L. Lois, NRR
H. Ashar, NRR
D. Shum, NRR
J. Zimmerman, NRR
D. Hoang, NRR
D. Reddy, NRR
R. De La Garza, NRR
L. Lund, NRR
R. Mathew, NRR
P. Lougheed, Region III
D. Merzke, NRR
J. Eargle, NRR
K. Hsu, NRR
V. Rodriguez, NRR
G. Cheruvenki, NRR
D. Coe, OCM
S. Tingen, NRR
J. Davis, NRR
J. Ayala, NRR
J. Fair, NRR
S. Burnell, OPA
J. Lamb, OEDO
S. Ali, RES
P. Buckberg, NRR
L. Tran, NRR
N. Dudley, NRR
M. Morgan, NRR

K. Chang, NRR
M. Mitchell, NRR
W. Bateman, NRR
C. Sydnor, NRR
C. Ng, NRR
D. Nguyen, NRR
R. Li, RES
M. Modes, Region I
J. Rajan, NRR
T. Le, NRR
S. Arora, NRR
H. Graves, RES
R. Sun, NRR
A. Pal, NRR
J. Canady, NRR

OTHER ATTENDEES

J. O'Rourke, Exelon
D. Benson, The Press of Atlantic City
T. Quintenz, AmerGen
J. Kandasamy, Exelon
J. Hufnagel, Exelon
C. Wilson, AmerGen
K. Muggleson, Exelon
T. Trettel, AmerGen
G. Harttraft, AmerGen
D. Warfel, Exelon
L. Corsi, Exelon
T. Mscisz, Exelon
A. Ouaou, Exelon
H. Ray, Exelon
S. Schwartz, Exelon
T. Schuster, Exelon
P. Tamburno, AmerGen
R. Benson, AmerGen
J. Petti, Sandia
G. Ritz, First Energy
C. Marks, ISL
R. Rucker, First Energy
R. Webster, Rutgers
J. Laird, Exelon
M. Gallagher, Exelon
A. Polonsky, Morgan Lewis
P. Cowan, Exelon
F. Polaski, Exelon
R. Skelskey, AmerGen
S. Hutchins, AmerGen
S. Rafferty-Czincila, Exelon
C. Myer, SNC
D. Spamer, Exelon
G. Krueger, Exelon
S. Getz, Exelon
D. Barnes, Exelon
M. Miller, Exelon
R. Barbieri, Exelon
J. Camire, Exelon
M. Pruskowski, Exelon
T. Rausch, Exelon
B. Meher, Exelon
M. Hessheimer, Sandia
K. Green, ISL
M. Fallin, Constellation Energy
N, Clunn, Asbury Park Press
J. Zielinski, Congressman Saxton Staff
P. Gunter, NIRS

The presentation slides, handouts used during the meeting, and a complete list of attendees are attached to the office copy of the meeting minutes. The presentations to the Subcommittee are summarized below.

Opening Remarks

Mr. Maynard, Chairman of the Plant License Renewal Subcommittee, convened the meeting and made a few introductory remarks. The purpose of this meeting is to review the LRA submitted by AmerGen for OCGS and the associated draft SER with open items prepared by the staff.

Staff Introduction

Ms. Lund, NRR, introduced several members of the staff including Mr. Ashley (License Renewal Program Manager), Mr. Modes (Inspection Team Leader), and Mr. Gillespie (Director for the Division of License Renewal). Ms. Lund stated that the LRA was submitted in July 2005 using the draft revision to the Generic Aging Lessons Learned (GALL) Report that was issued in January 2005. The LRA was reconciled with the final version of the GALL Report that was issued in September 2005. The staff has issued 108 Requests for Additional Information (RAIs), and the draft SER contains five open items.

Oyster Creek Generating Station License Renewal Application

Mr. Gallagher, AmerGen, introduced himself, Mr. Rausch (Site Vice President), Mr. Polaski (License Renewal Manager), Mr. Hufnagel, (Project Licensing Engineer), Mr. Quintenz (Site Lead License Renewal Engineer), and other members of AmerGen staff in attendance. The presentation by AmerGen described OCGS, its operating history, the license renewal project, and the commitment tracking process.

<u>Oyster Creek Generating Station Description</u>

Mr. Polaski, AmerGen, stated that OCGS is located in Ocean County New Jersey. The plant is a General Electric (GE) BWR-2 reactor in a Mark I containment. An interim spent fuel storage installation site is located at the plant. The ultimate heat sink is the Barnegat Bay. The overall core damage frequency for internal events is 1.1×10^{-5}. Commercial operation began in April 1969 under a provisional operating license. The full term operating license expires on April 9, 2009. The original and current licensed power level is 1930 MWt. There are no plans for a power uprate. OCGS has been operating under 24 month cycles since 1991.

<u>Operating History</u>

Mr. Gallagher, AmerGen, described the corrective actions implemented to address corrosion of the drywell liner. During refueling outages in the mid-1980s the sand bed drains were found to be clogged, and water was discovered in the sand bed regions. Leaks in the reactor cavity allowed water to flow through the gap between the drywell and the reactor building to the sand bed region. Approximately 1,000 ultrasonic (UT) thickness measurements were taken of the drywell to identify the thinnest locations in the sand bed region and upper elevations. Core samples were also taken to confirm the UT measurements and verify that the mechanism was general corrosion. A random UT inspection plan was implemented to verify the adequacy of measurement locations. The staff accepted this program in an SER dated November 1, 1995.

The corrective actions implemented in the early 1990s to address drywell corrosion included: (1) reanalyzing the containment peak pressure to establish additional shell thickness margin; (2) determining the acceptable shell thickness; (3) taking UT measurements to verify minimum thickness with margin; (4) reducing the source of water leakage; (5) removing sand from the sand bed region; (6) clearing the sand bed drains; and (7) coating the drywell shell in the sand bed region. These corrective actions were determined to be effective by the licensee because UT measurements taken in 1992 and 1994 confirmed that corrosion in the sand bed region had arrested. Since the UT measurements taken in 1996 contained some uncertainties, additional testing will be performed in 2006 to confirm that corrosion has stopped. Mr. Gallagher added that visual inspections of the coating were also performed in 1994.

The aging management program (AMP) for the drywell in the early 1990s consisted of taking UT measurements of the upper drywell and performing visual inspections of the coating in the sand bed region of the drywell shell every other refueling outage. This AMP has been enhanced to apply a strippable coating to the reactor cavity liner and monitor for water leakage. UT measurements will also be taken in the sand bed region of the drywell shell before entering the period of extended operation (PEO), then after four years, and then every ten years. The coating on the sand bed region of the drywell shell will be inspected before entering the PEO and then every ten years. The NRC will be notified within 48 hours of any deviations from expected inspection results.

Mr. Gallagher concluded that the corrective actions implemented to mitigate corrosion of the drywell shell have been effective and that an effective AMP has been developed to ensure continued safe operation of the plant.

The five open items listed in the SER are related to the drywell corrosion issue: (1) adequacy of the sample size for UT measurements at shell plate thickness transitions; (2) potential corrosion of the embedded shell; (3) impact of corrosion on the strength of the drywell shell related to buckling; (4) use of ASME Section III NE-3213.10 for analysis of localized thin shell areas; and (5) extent of follow-up exams of coated sand bed surfaces if leakage is detected.

Mr. Polaski briefly described other operating events associated with reactor vessel internals (core shroud, core spray spargers, top guide, and control rod drive stub tube), electrical cables, and underground piping.

<u>License Renewal Project</u>

AmerGen submitted the LRA on July 22, 2005 using the standard format in NEI 95-10 Revision 6. It was prepared using the January 2005 draft versions of NUREG-1800 and NUREG-1801. AmerGen prepared a reconciliation document comparing the LRA to the final versions of these documents. Of the 57 AMPs credited for license renewal, 50 are consistent with the GALL Report and seven are plant-specific. Thirty six are existing programs and 21 are new programs.

Mr. Polaski stated that in 1992 the two Forked River Combustion Turbines were credited as the alternate AC power supply for OCGS during a station blackout. Each combustion turbine is rated at 38 MWe. They are owned and operated by First Energy and are covered by the Maintenance Rule. Since the turbines have demonstrated high reliability, the initial AMP developed by AmerGen was based upon reliability monitoring. After discussions with the staff, the applicant elected to establish multiple GALL-based AMPs to manage aging of long-lived passive components in the turbines.

Commitment Management

The 65 commitments made by the applicant are listed in Appendix A of the SER. For each of these commitments a Passport commitment tracking number has been assigned and an associated action containing additional details has been issued. Mr. Quintenz, AmerGen, stated that 257 new and 111 enhanced implementation activities have been identified. 68% of these activities will be performed on-line. The remaining activities will be performed during the 2006 and 2008 refueling outages.

Mr. Gallagher summarized AmerGen's presentation by stating that the established AMPs will ensure safe operation and that license renewal commitments are on track for implementation prior to entering the PEO.

SER Overview

The presentation by Mr. Ashley, NRR, Mr. Modes, Region I, and Mr. Ashar, NRR, provided an overview of the staff's SER with open items. They described the staff's review of activities associated with scoping, screening, aging management, and time-limited aging analyses (TLAAs).

Mr. Ashley stated that the SER with open items was issued on August 18, 2006 with five open items and no confirmatory items. It contains three proposed license conditions. The staff has issued 108 RAIs and 366 audit questions. The one major component that had an expanded level of detail is the Forked River Combustion Turbine. The staff's audits and inspections were conducted between September 2005 and April 2006.

Scoping and Screening

The staff concluded that the applicant's scoping and screening results included all structures, systems, and components (SSCs) within the scope of license renewal and subject to an aging management review.

Onsite Inspection Results

Mr. Modes described the inspections performed by Region I in support of NRR's review of this LRA. A team of eight inspectors conducted a two-week inspection in accordance with Inspection Procedure 71002.

The scoping and screening portion of the inspection emphasized physical walkdowns of the plant and concentrated on non safety-related systems whose failure could impact safety-related systems. The inspection concluded that the methodology was adequate and consistently applied.

The aging management portion of the inspection reviewed 30 AMPs and two TLAA programs. Some minor inconsistencies were identified. The applicant either revised the LRA or entered the inconsistencies into the corrective action program. The inspection concluded that the existing AMPs were implemented as described in the LRA and that enhancements/exceptions to the AMPs were acceptable. The inspection also confirmed that commitments were captured in the Oyster Creek commitment tracking system.

Overall, the inspection results supported a conclusion that the proposed activities will reasonably manage the effects of aging on SSCs identified in the LRA. In addition the documentation supporting the LRA was in an auditable and retrievable form.

Mr. Modes concluded his presentation by describing the current performance of Oyster Creek. OCGS is in the regulatory response column of the NRC Action Matrix. There is an open cross-cutting issue in the area of human performance, and a white inspection finding in the area of emergency preparedness. All performance indicators for the plant are green.

Aging Management Programs and Aging Management Review

Mr. Ashley described the staff's evaluation of some of the AMPs and aging management reviews for OCGS.

The Protective Coating Monitoring and Maintenance Program is an existing program consistent with the GALL Report. This program is credited for aging management of the drywell and torus. 100% of the epoxy coating on the sand bed region of the drywell will be inspected before entering the PEO and every then every ten years. These inspections will be staggered such that at least three bays will be inspected every other refueling outage. Inspection of all torus bays will occur at a frequency of every other refueling outage. Should the current coating be replaced, the inspection frequency and scope will be reevaluated.

The Structures Monitoring Program is an existing program that is credited for aging management of station blackout systems, phase bus enclosure assemblies, and fire protection communication system structures. It includes elements of the Masonry Wall Program and Regulatory Guide 1.127 (Inspection of Water-Control Structures Associated with Nuclear Power Plants).

Mr. Ashley stated that the AMPs associated with the drywell shell are the (1) ASME Section XI, Subsection IWE Program; (2) Protective Coating and Monitoring and Maintenance Program; and (3) 10 CFR 50 Appendix J Program. For each refueling outage water leakage is monitored at refueling seals, drywell airgap drains, and sand pocket drains. UT measurements of the sand pocket region performed in 1992 and 1994 determined that corrosion of the drywell has been arrested. AmerGen has made 11 license renewal commitments related to the drywell.

Except for the fresh water pump-house, the below-grade environment of inaccessible concrete is non-aggressive based on the pH, chloride levels, and sulfate levels. Periodic tests of ground water will be performed as part of the Structures Monitoring Program.

Time Limited Aging Analyses (TLAAs)

Mr. Ashley described the staff's evaluation of TLAAs associated with neutron embrittlement, metal fatigue, and environmental qualification of electrical equipment.

Neutron embrittlement effects TLAAs associated with upper shelf energy (USE) and relief from inspections of reactor vessel circumferential welds. The percent drop in USE for the limiting plate and weld in OCGS meet the acceptance criteria established in BWRVIP-74. The reference temperature for the limiting plate and weld in OCGS also meet the acceptance criteria established in BWRVIP-05.

Metal fatigue is managed by the Fatigue Monitoring Program. The cumulative usage factor for components are projected to be less than 1.0 based on a 60-year life. The staff accepted the applicant's evaluations of these TLAAs.

The applicant's environmental qualification of electrical equipment program is consistent with the GALL Report. The staff found that this program is adequate to manage the effects of aging on the intended functions of electrical components.

The staff concluded that pending resolution of the five open items, the applicant has demonstrated that for the drywell corrosion TLAA, the effects of aging will be adequately managed for the PEO.

Mr. Ashley added that the list of TLAAs is adequate and that there are no plant-specific exemptions.

Confirmatory Analysis of Oyster Creek Drywell

Mr. Ashar, NRR, described structural analyses of the Oyster Creek drywell performed by Sandia National Laboratories under contract with the NRC. A 3-dimensional finite element model was created that included the entire drywell, the equipment hatch, and ten vent lines. The torus was not modeled. Gravity, seismic, and other loads taken from previous GE analyses were applied to the model. The thicknesses of the cylinder, upper sphere, and middle sphere were uniform values based upon extrapolated UT measurements. The lower sphere was divided into ten regions. The thickness of each of these regions was determined by averaging UT measurements.

Sandia's preliminary results indicate that all stresses meet ASME Code requirements. The three controlling load combinations are (1) refueling, (2) design basis accident with earthquake, and (3) post accident flooding with earthquake. For the refueling load combination ASME requires a safety factor of 2.0 for buckling. Sandia's preliminary analysis results show a safety factor of 3.85 for buckling assuming no degradation of the drywell. If drywell degradation is assumed, the buckling safety factor decreases to 2.15. For the design basis accident with earthquake load combination, buckling is not a concern. For the post accident flooding with earthquake load combination, the required ASME safety factor for buckling is 1.67. The analyses performed by Sandia show buckling safety factors 2.74 and 3.65 with and without degradation, respectively.

Oyster Creek Nuclear Generating Station License Extension: Drywell Shell Corrosion

The presentation by Mr. Gunter, Nuclear Information Resource Service, and Mr. Webster, Rutgers Environmental Law Clinic, focused primarily on the issue of drywell shell corrosion.

Mr. Webster described his concerns with whether the drywell shell currently meets the required safety margins and if so, whether any significant degradation in the future would be detected before these safety margins are violated. These concerns apply to both the embedded and sand bed regions of the drywell. Before 1992 there was no seal to prevent water from entering the gap between the drywell and concrete. Mr. Webster alleged that when the sand was removed in 1992, deep craters were found in the sand bed floor and water ponds were found.

Mr. Webster provided several reasons for possible corrosion of the drywell: (1) from 1960 to 1992 conditions in the embedded region were favorable for crevice corrosion; (2) verification that the elastomer seal has kept the embedded region dry has not been performed; (3) groundwater could be a source of moisture; (4) corrosion rates in the sand bed region do not bound corrosion rates in the embedded region; and (5) the removal of sand could have accelerated corrosion rates in the embedded region because of differential aeration. Mr. Webster added that monitoring for wet conditions and a comprehensive check of the current thickness of the embedded region is needed.

For the sand bed region, the three acceptance criteria for drywell thickness are a uniform wall thickness of 0.736 inches, no single point less than 0.49 inches, and one square foot per bay may be less than 0.736 inches but must be greater than 0.536 inches. Mr. Webster stated that he believed the problems with these criteria are that the sand bed thickness is not uniform, anti-symmetrical buckling is not considered, the assumption of a spherical shape is not justified, and the derivation of the small area criterion was not rigorous.

Mr. Webster stated that he believed the last UT measurements that were taken with procedures that are not in question occurred in 1992. UT results from 1994 were not validated and some of the 1996 results were anomalous in that a 50 mil increase in thickness was detected. Mr. Webster added that AmerGen's claim that the corrosion rate of the drywell in the sand bed region is zero was based in part on these UT results from 1994 and 1996.

Mr. Webster described a June 23, 2006 report in which Dr. Rudolf Hausler concluded that a more spatially comprehensive set of UT thickness measurements is needed to accurately represent the condition of the drywell.

Mr. Webster stated that he believed the current margins are not well known because water could have been accumulating over several years, visual monitoring of the epoxy coating is inadequate, the UT data is insufficient, and UT measurement uncertainty was not fully considered. Since the present situation is poorly defined, predictions about the future become highly uncertain.

Mr. Webster described several reasons why he believed the UT inspection program proposed by AmerGen is inadequate. The spatial scope is too small because AmerGen proposes to inspect the same locations measured in 1992, 1994, and 1996. The statistical techniques used in the data analyses are flawed in part because the assumption of linear corrosion rates is incorrect, the use of a 95% confidence interval is not justified, data filtering is inappropriate, and measurement uncertainties are not considered systematically. Visual examination of coating integrity is inadequate and should be augmented with other industry measurement standards. Water monitoring is inadequate because the applicant failed to monitor the sand bed drains for eight years and then dumped the collected water prior to testing. The inspection frequency can not be established until the safety margins and worse case corrosion rates are known. Mr. Webster suggested that another possible degradation mechanism for the drywell is chloride induced fatigue cracking.

Mr. Webster concluded by stating that he believed there is no current reasonable assurance of safety and that the proposed monitoring program for the drywell is inadequate.

Member Comments

General

Members Sieber and Bonaca asked about recent inspection findings at OCGS. The applicant stated that the white finding in the area of emergency preparedness was due to a failure to respond to plant conditions when an action level was reached. There were also other inspection findings dealing with procedural compliance that led to an open cross-cutting issue in the area of human performance.

Members Shack and Armijo asked about water chemistry. The applicant stated that hydrogen water chemistry was implemented in 1992 and is continually monitored to ensure that the potential is less than 230 millivolts. Noble metals were added in 2002.

Member Sieber and Consultant Barton asked several questions regarding the Forked River Combustion Turbines. Both turbines are committed to station blackout even though only one is needed. As part of a surveillance program, the turbines are tested at 10% load every outage. The applicant stated that the turbines are stable at this load. AmerGen meets regulatory with First Energy regarding scheduled maintenance of these turbines. The control room operators at OCGS are also notified of any unscheduled maintenance on these turbines.

Members Wallis and Abdel-Khalik asked about the 257 new and 111 enhanced implementation activities identified for license renewal. The applicant stated that these are activities that were generated as a result of creating the AMPs for license renewal. They include items like one-time and periodic inspections. Priorities were established based on the significance of the activity and the need to have the unit off-line in order to implement the activity.

Scoping and Screening

In response to a question by Member Sieber, the applicant stated that the steam dryers are in scope of license renewal.

Aging Management

Chairman Maynard and Consultant Barton asked about aging management of medium voltage underground cables. The applicant stated that four cables known to be susceptible to failure in wetted environments will be replaced with Okenite cables that are qualified for these conditions. The AMP for these cables is consistent with the GALL Report, but the testing methodology has not yet been approved by IEEE. The applicant added that some of the cables were rerouted through locations that are not susceptible to water intrusion.

Consultant Barton asked about aging management of underground piping. The applicant stated that the diesel fuel transfer line from the storage tank to the diesel generator leaked because the coating was damaged during installation. This line is currently being replaced.

AmerGen took exception to the use of the ASME Code Class I Small Bore Piping Program and has proposed a one-time inspection program instead. This program will examine one socket weld connected to the isolation condenser. Consultant Barton asked why the staff found this acceptable. The staff stated that the one socket weld is a worse-case location for small bore piping and will be destructively examined. Since small bore piping is not normally inspected, a sample size of one is acceptable. The staff added that this approach has been accepted at other facilities. Member Sieber noted that small bore piping, particularly socket welds, fail more frequently than large bore piping.

Drywell

Several Members asked about the foam material located in the gap between the drywell shell and concrete. After the drywell shell is erected, a foam known as Firebar D is applied to the exterior of the shell. Concrete is then poured around it. During the hydrostatic pressure test of the drywell, the foam compresses leaving a 1 inch air gap between the drywell and concrete. This air gap is needed to account for seismic movements of the containment shell. Water washing down this gap could pick up chlorides or other contaminants from Firebar D and carry it down to the sand bed region. The applicant stated that after the sand was removed, the water was tested and contained 45 parts per billion chloride and 17 parts per billion sulfate. Member Wallis noted that the presence of oxygen and water are needed for corrosion.

In response to a question from Member Bonaca the applicant stated that they are sure that the water found in the sand bed region originated from the refueling cavity. Cracks in the refueling cavity liner allowed water to accumulate in the sand bed region and corrode the drywell. Member Bonaca noted that these cracks violate a design requirement and defeat the purpose of inspecting the refueling seals and bellows. The applicant stated that one of the license renewal commitments is to apply a strippable coating to the reactor cavity liner before the reactor cavity is filled. However, there were two outages in the past in which the strippable coating was not applied because there were plans to decommission the plant. Member Bonaca added that unless the source of water is addressed there is no assurance of future performance of the liner. Member Shack asked if any leakage was observed with the strippable coating in place. The applicant responded that during the last two outages the strippable coating was used and no leakage was observed. The applicant added that even though there are no current licensing basis commitments to monitor for water leakage in this area, there is a license renewal commitment for daily leakage monitoring during outages.

In response to a question from Member Shack, the applicant stated that a cover plate is located at the top of the sand bed region.

Several Members asked about the clogged sand bed drains. The applicant stated that the drains themselves were filled with sand to prevent drainage of the sand and as a result of water intrusion, fines were mixed with the sand to plug the drains. All five drains were plugged.

There was an extensive discussion regarding the UT measurements of the drywell. The applicant stated that approximately 1,000 UT measurements were taken throughout the drywell to identify the thinnest locations. In the early 1980s UT measurements were taken from accessible portions of the inside of the drywell. These locations covered 360 degrees of the drywell. Based on these measurements, the drywell thickness varied from 0.85 inches to 1.1 inches. Some of these locations were in the sand bed region. Additional measurements were taken from two locations underneath the concrete from inside the drywell. The applicant concluded that the data from the accessible portions of the drywell were representative of the areas underneath the concrete.

After the sand was removed, a visual inspection of the exterior of the drywell showed general corrosion with some localized thinned areas. Additional UT measurements were taken from outside of the drywell. These results showed a thickness of 0.603 inches.

Several Members asked about the margins associated with the drywell thickness in the sand bed region. The applicant stated that an analysis was performed in which the thickness of the entire sand bed region was assumed to be uniformly reduced to 0.736 inches and the stresses were determined to be on the borderline of acceptability by the ASME Code. This analysis included the appropriate safety factors. Since the minimum measured value of the drywell in this area was 0.80 inches, there is a margin of 65 mils. Member Abdel-Khalik asked about the uncertainties associated with the measured values. The applicant stated that at the 95% confidence level the uncertainties are approximately plus or minus 10 mils. Chairman Maynard added that the ASME Code contains some conservatism as well. Member Sieber noted that if individual thickness measurements are taken, there is a chance that the thinnest location was not measured.

The applicant stated that two thickness criteria for the sand bed region of the drywell are a minimum average thickness of 0.736 inches and a minimum local thickness of 0.49 inches. The original thickness was 1.154 inches. The UT measurements from two of the bays indicate a minimum thickness of 0.80 inches. There are 19 monitoring locations on the interior of the drywell covering the 360 degrees of the sand bed region. For each of these locations, 49 UT measurements are taken and compared to these criteria.

Member Wallis asked about buckling of the drywell. The applicant stated that buckling is a concern when the refueling cavity is filled with water.

In response to a question by Member Wallis the applicant stated that the sand was permanently removed. The applicant added that an analysis was performed to demonstrate that even without the sand, the drywell stresses were acceptable.

Member Armijo asked about the access ports used to inspect the sand bed region. The applicant stated that ten ports are located in the concrete that allow a 360 degree visual inspection of the drywell. During operation the ports are filled with boron bags.

Members Wallis and Armijo asked about the condition of the exterior surface of the drywell in the sand bed region. The applicant stated that after the corrosion products were removed the surface was not completely smooth. There were pockmarks. The corrosion of that portion of the drywell near the base of the sand bed region where the shell becomes embedded in concrete was similar to other areas of the drywell.

Members Wallis and Abdel-Khalik asked about corrosion of the embedded portion of the drywell. The applicant stated that corrosion in this area of the drywell should be less than in the sand bed region because of the alkaline environment of the concrete. In addition the drywell skirt would prevent water from flowing to lower regions. Member Shack noted that this would be true if active corrosion was occurring in the sand bed region. The applicant stated that a silicon seal protects the drywell in the embedded region.

Consultant Barton and several Members asked about the epoxy coating applied to the exterior of the drywell shell in the sand bed region. The applicant stated that estimates of service life varied from 10 to 20 years but the vendor could not guarantee the coating. The current inspection program is to examine two of the ten bays every other refueling outage. In the future a minimum of three bays will be examined every other outage. Visual inspections are performed according to ASTM standards and certified inspectors look for blistering, flaking, or cracking of the coating. If blistering, flaking, or cracking of the coating is found, the scope of the inspection is increased.

The sand bed region of the drywell will be inspected by UT prior to entering the PEO, then after four years, and then every ten years. Several Members asked about the basis for these inspection intervals. Before entering the PEO, the five bays that have not been inspected will be examined. The applicant stated that the current projected corrosion rate would not result in the drywell falling below the minimum required thickness within four years. The basis of the ten year inspection interval is the ASME Section XI Inservice Inspection requirements. The applicant added that inspections of the three bays every other outage are staggered such that all of the bays will be examined every ten years.

Water from the sand bed drains are ultimately collected in five-gallon jugs in the torus room. Member Bonaca noted that during a walkdown of the torus room in March 2006 the water that had collected in these jugs was emptied before it was tested. The applicant stated that the commitment to analyze this water was never entered into the commitment tracking system. The applicant added that it does not believe this water was from an active leak. Member Wallis noted that water leakage may not collect in the drain lines because of evaporation. The staff stated that in addition to monitoring for leakage, UT measurements are taken of the drywell.

Chairman Maynard asked if the staff was confident that the actions identified by AmerGen will prevent and identify leakage. The staff stated that the five open items in the SER describe the information needed in order to make that determination.

Member Wallis asked why some bays experienced more corrosion than others. The staff responded that leaks in the refueling cavity liner were not uniform. The bays in which more water was collected corresponded to those bays with more corrosion.

Several Members asked about the basis for the drywell thicknesses used in Sandia's finite element model. The staff stated that individual UT measurements were averaged to get thicknesses for each of the bays. For two bays a localized thinned area directly below the vent lines was also modeled. Member Armijo noted that there were discrepancies between the thicknesses used by Sandia and those reported by the applicant. If results from the current inspection suggest that the thicknesses should be revised, the staff plans to revise the analyses. Member Shack asked if the model could be refined to include the localized corrosion points instead of just using average thicknesses. A Sandia representative stated that it would not be practical to build that level of detail into the model. Member Wallis suggested performing sensitivity studies that varied thicknesses and locations of locally thinned areas. Member Abdel-Khalik noted that the location of the locally-thinned areas in the Sandia model may not be the most limiting location. The staff responded that they were placed at this location because this is where the corrosion was observed.

Member Sieber asked about changes to the seismic loads on the drywell due to corrosion and removal of the sand. A representative from Sandia stated that the information available in the

FSAR was not complete enough to do as rigorous an analysis as that performed by GE. The objective of Sandia's work was to determine how the safety factors changed as a result of corrosion in the drywell. Since Sandia applied the same loads as GE, the results should be conservative. Member Sieber noted that the analysis results should indicate whether or not the drywell fails. The staff added that Sandia's analyses are only to confirm the applicant's calculation of margin in the drywell.

In response to a question from Member Shack, the staff stated that the loads used in Sandia's analyses were only design load combinations. They did not examine the loads needed to fail the drywell.

Several Members asked Mr. Webster about margins in the drywell. Mr. Webster stated that the margins analysis performed by Dr. Hausler was based on a statistical analysis of UT measurement data to calculate the amount of corrosion necessary to exceed the code's acceptance criteria.

In response to a question from Member Armijo, Mr. Webster stated that groundwater should be ruled out as a possible source of moisture.

In response to a question by Member Shack, Mr. Webster stated that the National Association of Corrosion Engineers International Standard Test Method TM 00384 is for inspection of internal tubular coatings.

Mr. Webster was unable to provide specific references regarding chloride induced stress corrosion cracking in carbon steels.

Mr. Webster stated that he was not aware of any specific techniques that could directly measure the thickness of the drywell embedded in concrete.

Staff Commitments

The staff will provide additional information to the ACRS regarding UT measurements of the drywell and how these measurements were used in Sandia's finite element analyses.

Subcommittee Discussion Follow-up Actions

Chairman Maynard stated that the ACRS has not reached a conclusion regarding this LRA.

The Subcommittee Chairman will summarize the discussions at the October 2006 ACRS meeting.

Background Materials Provided to the Subcommittee

1. Safety Evaluation Report with Open Items Related to the License Renewal of the Oyster Creek Generating Station, dated August 18, 2006

2. Oyster Creek Generating Station-Application for Renewed Operating License, dated July 22, 2005

3. Supplemental Information Related to the Aging Management Program for the Oyster Creek Drywell Shell, Associated with AmerGen's License Renewal Application, dated June 20, 2006

4. Audit and Review Report for Plant Aging Management Reviews and Programs – Oyster Creek Generating Station, dated August 18, 2006

5. Supplemental Response to NRC Request for Additional Information (RAI 2.5.1.19-1), dated September 28, 2005, Related to Oyster Creek Generating Station License Renewal Application, dated November 11, 2005

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NOTE:
Additional details of this meeting can be obtained from a transcript of this meeting available in the NRC Public Document Room, One White Flint North, 11555 Rockville Pike, Rockville, MD, (301) 415-7000, downloading on the Internet at http://www.nrc.gov/reading-rm/doc-collections/acrs/ can be purchased from Neal R. Gross and Co., 1323 Rhode Island Avenue, NW, Washington, D.C. 20005, (202) 234-4433 (voice), (202) 387-7330 (fax), nrgross@nealgross.com (e-mail).

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Mike Fenge

If you do not have access to ADAMS, or if there are problems in accessing the documents located in ADAMS, contact the NRC Public Document Room (PDR) Reference staff at 1–800–397–4209, 301–415–4737, or by e-mail to *pdr@nrc.gov*. These documents may also be viewed electronically on the public computers located at the NRC's PDR, O 1 F21, One White Flint North, 11555 Rockville Pike, Rockville, MD 20852. The PDR reproduction contractor will copy documents for a fee.

Dated at Region 1, 475 Allendale Road, King of Prussia this 12th day of September 2006.

For the Nuclear Regulatory Commission.

James P. Dwyer,

Chief, Commercial and R&D Branch, Division of Nuclear Materials Safety, Region I.

[FR Doc. 06–7898 Filed 9–20–06; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Advisory Committee on Reactor Safeguards; Meeting of the Subcommittee on Plant License Renewal; Notice of Meeting

The ACRS Subcommittee on Plant License Renewal will hold a meeting on October 3, 2006, Room T–2B3, 11545 Rockville Pike, Rockville, Maryland.

The entire meeting will be open to public attendance.

The agenda for the subject meeting shall be as follows:

Tuesday, October 3, 2006—1:30 p.m. until 5 p.m.

The purpose of this meeting is to discuss the License Renewal Application for Oyster Creek and the associated Safety Evaluation Report (SER) with Open Items prepared by the NRR staff. The Subcommittee will hear presentations by and hold discussions with representatives of the NRC staff, AmerGen Energy Company, and other interested persons regarding this matter. The Subcommittee will gather information, analyze relevant issues and facts, and formulate proposed positions and actions, as appropriate, for deliberation by the full Committee.

Members of the public desiring to provide oral statements and/or written comments should notify the Designated Federal Official, Mr. Cayetano Santos (telephone 301/415–7270) five days prior to the meeting, if possible, so that appropriate arrangements can be made. Electronic recordings will be permitted.

Further information regarding this meeting can be obtained by contacting the Designated Federal Official between 7:30 a.m. and 4:15 p.m. (ET). Persons planning to attend this meeting are urged to contact the above named individual at least two working days prior to the meeting to be advised of any potential changes to the agenda.

Dated: September 15, 2006.

David C. Fischer,

Acting Branch Chief, ACRS/ACNW.

[FR Doc. 06–7890 Filed 9–20–06; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Advisory Committee on Reactor Safeguards; Subcommittee Meeting on Planning and Procedures; Notice of Meeting

The ACRS Subcommittee on Planning and Procedures will hold a meeting on October 3, 2006, Room T–2B1, 11545 Rockville Pike, Rockville, Maryland.

The entire meeting will be open to public attendance, with the exception of a portion that may be closed pursuant to 5 U.S.C. 552b(c) (2) and (6) to discuss organizational and personnel matters that relate solely to the internal personnel rules and practices of the ACRS, and information the release of which would constitute a clearly unwarranted invasion of personal privacy.

The agenda for the subject meeting shall be as follows:

Tuesday, October 3, 2006, 10:30 a.m. until the conclusion of business.

The Subcommittee will discuss proposed ACRS activities and related matters. The Subcommittee will gather information, analyze relevant issues and facts, and formulate proposed positions and actions, as appropriate, for deliberation by the full Committee.

Members of the public desiring to provide oral statements and/or written comments should notify the Designated Federal Official, Mr. Sam Duraiswamy (telephone: 301–415–7364) between 7:30 a.m. and 4 p.m. (ET) five days prior to the meeting, if possible, so that appropriate arrangements can be made. Electronic recordings will be permitted only during those portions of the meeting that are open to the public.

Further information regarding this meeting can be obtained by contacting the Designated Federal Official between 7:30 a.m. and 4 p.m. (ET). Persons planning to attend this meeting are urged to contact the above named individual at least two working days prior to the meeting to be advised of any potential changes in the agenda.

Dated: September 14, 2006.

Michael R. Snodderly,

Branch Chief, ACRS/ACNW.

[FR Doc. 06–7889 Filed 9–20–06; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Advisory Committee on Reactor Safeguards; Meeting Notice

In accordance with the purposes of Sections 29 and 182b. of the Atomic Energy Act (42 U.S.C. 2039, 2232b), the Advisory Committee on Reactor Safeguards (ACRS) will hold a meeting on October 4–6, 2006, 11545 Rockville Pike, Rockville, Maryland. The date of this meeting was previously published in the **Federal Register** on Tuesday, November 22, 2005 (70 FR 70638).

Wednesday, October 4, 2006, Conference Room T–2B3, Two White Flint North, Rockville, Maryland

8:30 a.m.–8:35 a.m.: Opening Remarks by the ACRS Chairman (Open)—The ACRS Chairman will make opening remarks regarding the conduct of the meeting.

8:35 a.m.–9:30 a.m.: Draft Final Revision 3 to Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment" (Open)–The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding draft final revision 3 to Regulatory Guide 1.7, which provides guidance for implementing the risk-informed 10 CFR 50.44, "Combustible Gas Control for Nuclear Power Reactors."

9:30 a.m.–11:45 a.m.: Proposed Updates to Regulatory Guides and Standard Review Plan (SRP) Sections in Support of New Reactor Licensing (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding proposed updates to Regulatory Guides and SRP Sections that are being made in support of new reactor licensing, criteria used by the staff in selecting Regulatory Guides and SRP Sections applicable to future plant licensing, and staff's recommendations that the ACRS not review certain Regulatory Guides and SRP Sections along with the reasons therefor.

12:45 p.m.–2:15 p.m.: Master Integrated Plan for New Reactor Licensing Activities (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding the development of the Master

Advisory Committee on Reactor Safeguards
Plant License Renewal Subcommittee Meeting
Oyster Creek Generating Station
October 3, 2006
Rockville, MD

-PROPOSED SCHEDULE-

Cognizant Staff Engineer: Michael A. Junge mxj2@NRC.GOV (301) 415-6855

Topics	Presenters	Time
Opening Remarks	Otto Maynard, ACRS	1:30 pm - 1:35 pm
Staff Introduction	Louise Lund, NRR	1:35 pm - 1:40 pm
Oyster Creek License Renewal Application A. Application Background B. Description of Oyster Creek C. Operating History D. Scoping Discussion E. Application of GALL F. Commitment Process	Timothy Rausch, Michael Gallagher, Fred Polaski, and John Hufnagel, Amergen Energy Company	1:40 pm - 2:40 pm
Break		2:40 pm - 2:55 pm
SER Overview A. Scoping and Screening Results B. Onsite Inspection Results	Donnie Ashley, NRR Michael Modes, Region I	2:55 pm - 3:15 pm
Aging Management Program Review and Audits	Donnie Ashley, NRR	3:15 pm - 3:45 pm
Time-Limited Aging Analyses	Donnie Ashley, NRR	3:45 pm - 4:15 pm
Confirmatory Analysis of Drywell	Hans Ashar, NRR	4:15 pm - 4:45 pm
Public Comments	Paul Gunter, Nuclear Information Resource Service Richard Webster, Rutgers Environmental Law Clinic	4:45 pm - 5:15 pm
Subcommittee Discussion	Otto Maynard, ACRS	5:15 pm - 5:30 pm

NOTE:

- Presentation time should not exceed 50 percent of the total time allocated for a specific item. The remaining 50 percent of the time is reserved for discussion.
- 50 copies of the presentation materials to be provided.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

SUBCOMMITTEE MEETING ON PLANT LICENSE RENEWAL

October 3, 2006
Date

NRC STAFF SIGN IN FOR ACRS MEETING

PLEASE PRINT

	NAME	NRC ORGANIZATION
1	Donnie Ashley	NRR/DLR
2	Veronica Rodriguez	NRR/DLR
3	Lambros Lois	NRR/DSS/SBWB
4	Ganesh Cheruvenki	NRR/DCI/CVIB
5	Hans Ashar	NRR/DE
6	Doug Coe	OCM/PBL
7	David Shum	NRR/DSS/SBPB
8	Stephen Tingen	NRR/DE/EQV
9	Jake Zimmerman	NRR/DLR/RLRB
10	Jim Davis	NRR/DLR/RLRC
11	Dan Hoang	NRR/DLR/RLRC
12	Juan Ayala	NRR/DLR/RLRA
13	Devender Reddy	NRR/DLR/RLRB
14	John Fair	NRR/DE/EEMB
15	Rodrigo De La Garza	NRR/DLR
16	Scott Burnell	OPA
17	Louise Lund	NRR/DLR
18	John G. Lamb	OEDO
19	Roy Mathew	NRR
20	Syed Ali	RES

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

SUBCOMMITTEE MEETING ON PLANT LICENSE RENEWAL

October 3, 2006
Date

NRC STAFF SIGN IN FOR ACRS MEETING

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	NAME	NRC ORGANIZATION
1	Patricia Lougheed	RIII/DRS → Licence Renewal
2	Perry Buckberg	NRR/DLR/RLRA
3	DANIEL MERZKE	NRR/DLR/RLRA
4	LINH TRAN	NRR/DLR/RLRB
5	Jason A. Eagle	NRR/DSS/SBPB
6	NOEL DUDLEY	NRR/DLR/RLRA
7	Kaihwa HSU	NRR/DLR
8	MICHAEL MORGAN	NRR/DLP
9	Ken Chang	NRR/DLP
10	J. Raja	NRR/DE
11	Matthew A. Mitchell	NRR/DCI/CVIB
12	Tommy Le	NRR/DLR/RLRB
13	W. Bateman	NRR/DCI
14	S. Arora	NRR/DLR
15	C. Sydnor	NRR/DCI
16	H. Graves	RES/DFERR
17	Ching H NG	NRR/DE/
18	Robert Sun	NRR/DLR
19	Duc Nguyen	NRR
20	Amar Pal	NRR/DE

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

SUBCOMMITTEE MEETING ON PLANT LICENSE RENEWAL

October 3, 2006
Date

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	NAME	NRC ORGANIZATION
1	Rui Li	RES
2	James Canady, III	NRR/DLR
3	Michael C. Modes	REGION 1
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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

SUBCOMMITTEE MEETING ON PLANT LICENSE RENEWAL

October 3, 2006
Date

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	NAME	AFFILIATION
1	JOHN O'ROURKE	EXELON
2	MICHAEL GALLAGHER	EXELON
3	David Benson	The Press of Atlantic City
4	Alex Polonsky	Morgan Lewis
5	Thomas Quintenz	AmerGen
6	PAM COWAN	EXELON
7	JHANSI KANDASAMY	EXELON
8	FRED POLASKI	EXELON
9	JOHN HUFNAGEL	EXELON
10	Rick Skelskey	AmerGen
11	CHRIS WILSON	AMERGEN
12	Steven P. Hutchins	Amergen / Exelon
13	Kevin Muggleston	Exelon
14	Shannon Rafferty-Czincila	Exelon
15	TIMOTHY TRETTEL	AMERGEN
16	GREGORY HARTTRAFT	AMERGEN
17	CHALMER MYER	SNC
18	Debora Spamer	Exelon
19	DON Wartel	EXELON
20	GREG KRUEGER	EXELON

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

SUBCOMMITTEE MEETING ON PLANT LICENSE RENEWAL

October 3, 2006
Date

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	NAME	AFFILIATION
1	Louis J. Corsi	EXELON
2	Stuart C. Getz	EXELON
3	Thomas J. Mscisz	Exelon
4	Daniel T Barnes	Exelon
5	Ahmed M Ouaou	Exelon
6	Mark Miller	Exelon
7	Howie Ray	Exelon
8	Robert Barbieri	Exelon
9	Sylvain Schwartz	Exelon
10	JOHN A. CAMIRE	EXELON
11	Terence Schuster	Exelon
12	MARCIA PRUSKOWSKI	EXELON
13	Peter Tamburro	AmerGen
14	TIM RAUSCH	EXELON - AMERGEN
15	Rachelle Benson	AmerGen
16	Bill Maher	Exelon
17	Jason Petti	Sandia National Lab
18	Mike Hessheimer	Sandia National Labs
19	GLENN S. RITZ	FIRSTENERGY CORP
20	Kim Green	ISL, Inc.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

SUBCOMMITTEE MEETING ON PLANT LICENSE RENEWAL

October 3, 2006
Date

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1	Cliff Marks	ISL
2	MICHAEL FALLIN	CONSTELLATION ENERGY
3	Roger Rucker	First Energy
4	Nick Clunn	Asbury Park Press
5	Richard Veber	Rutgers
6	Joni Zielinski	Congressman Saxton
7	Jim LAIRD	Exelon
8	PAUL GUNTER	NIRS
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An Exelon Company

Oyster Creek License Renewal Presentation to ACRS Subcommittee

October 03, 2006

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An Exelon Company

AmerGen Representatives

- Michael Gallagher, Vice President, License Renewal Projects
- Timothy Rausch, Vice President, Oyster Creek
- Fred Polaski, License Renewal Manager
- Tom Quintenz, Site Lead License Renewal Engineer
- John Hufnagel, Licensing Lead

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Agenda

- Description of Oyster Creek
- Current Plant Status
- Operating History
- Drywell Corrosion
- NRC Open Items
- License Renewal Methodology & Results
- Commitment Management
- Status of Program Implementation
- Summary

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Description of Oyster Creek

- Located in Lacey Township, Ocean County, NJ
- Barnegat Bay is Ultimate Heat Sink
- GE BWR 2 with Mark I Containment
- Interim Spent Fuel Storage established onsite
- Overall CDF
  - Internal events: 1.1E-05/year
  - LERF: 5.8E-07/year

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Current Plant Status

- Operating in 20th cycle
- Transitioned to 24 month cycles in 1991
- Currently operating in end-of-cycle coast down
- Regulatory Oversight Program (ROP) status

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Operating History

- Full (Original) Design Power Level – 1930 MWth
- Commercial Operation
  - April 1969 Provisional Operating License (POL) issued
  - Aug 1969 Authorized to 1600 MWth
  - Dec 1970 Authorized to 1690 MWth
- Current Licensed Thermal Power 1930 MWth
  - Authorized in November 1971
  - No power uprates performed or planned
  - Design electrical rating 650 MWe
- Full Term Operating License
  - Issued July 1991
  - Expires April 09, 2009

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Drywell Corrosion

- Background
- Initial Corrective Actions
- Verification of effectiveness
- Initial Aging Management Program
- Enhanced Aging Management
- Conclusion

Background

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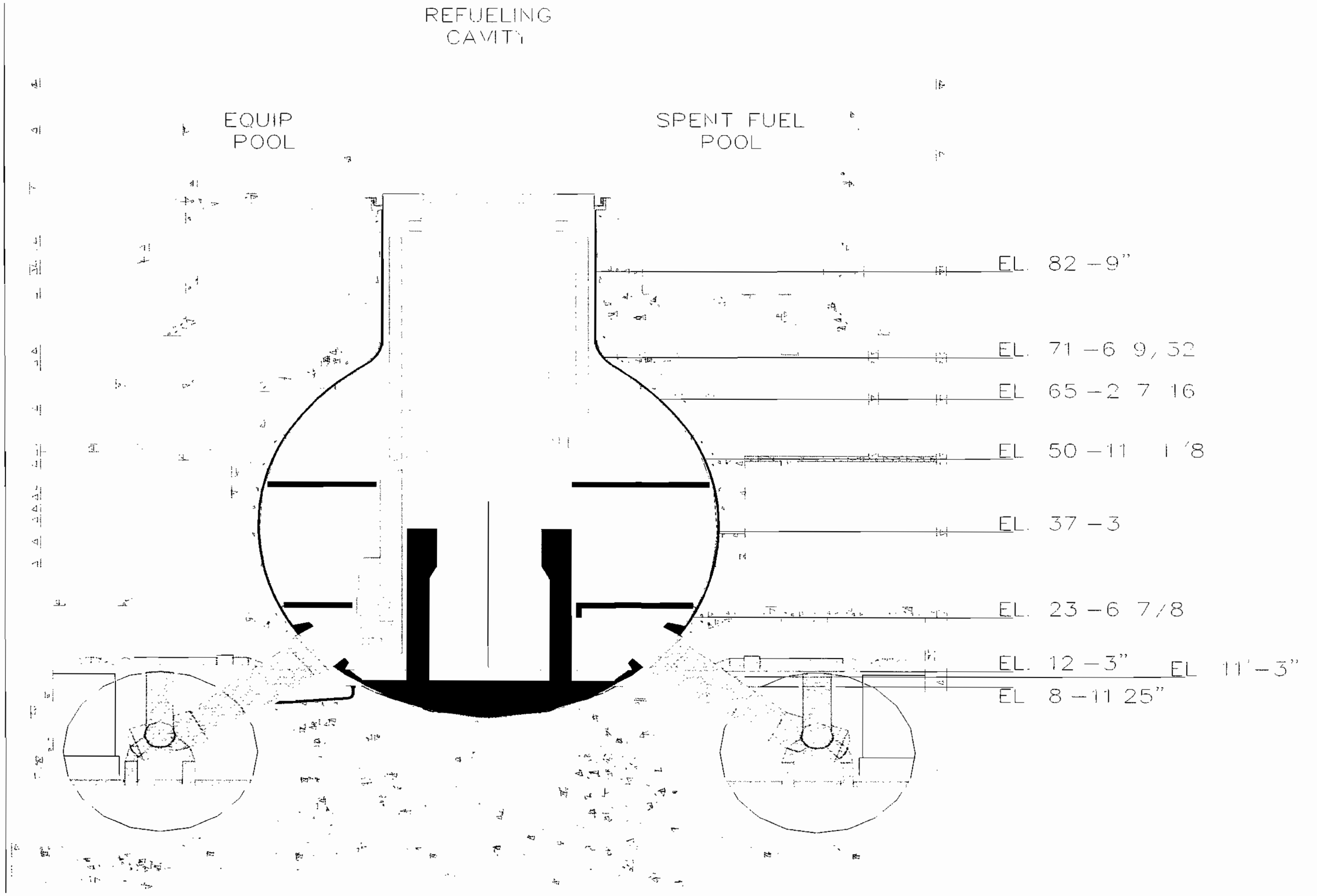

Sand Bed Area

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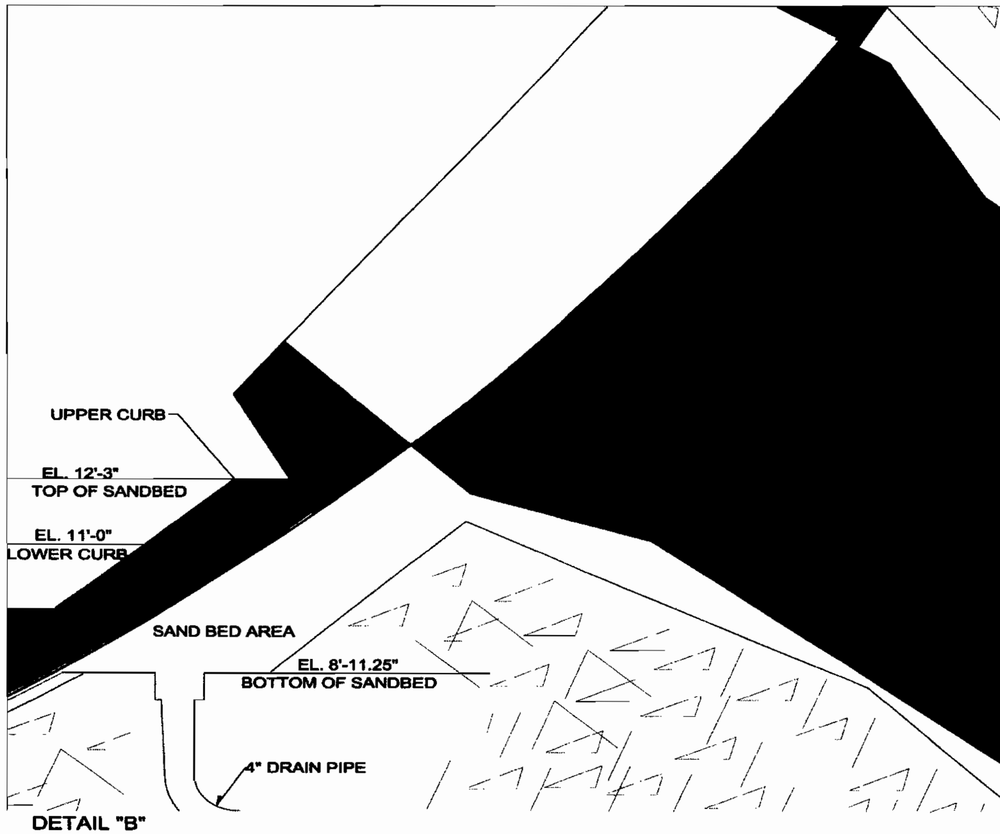

Torus Downcomer

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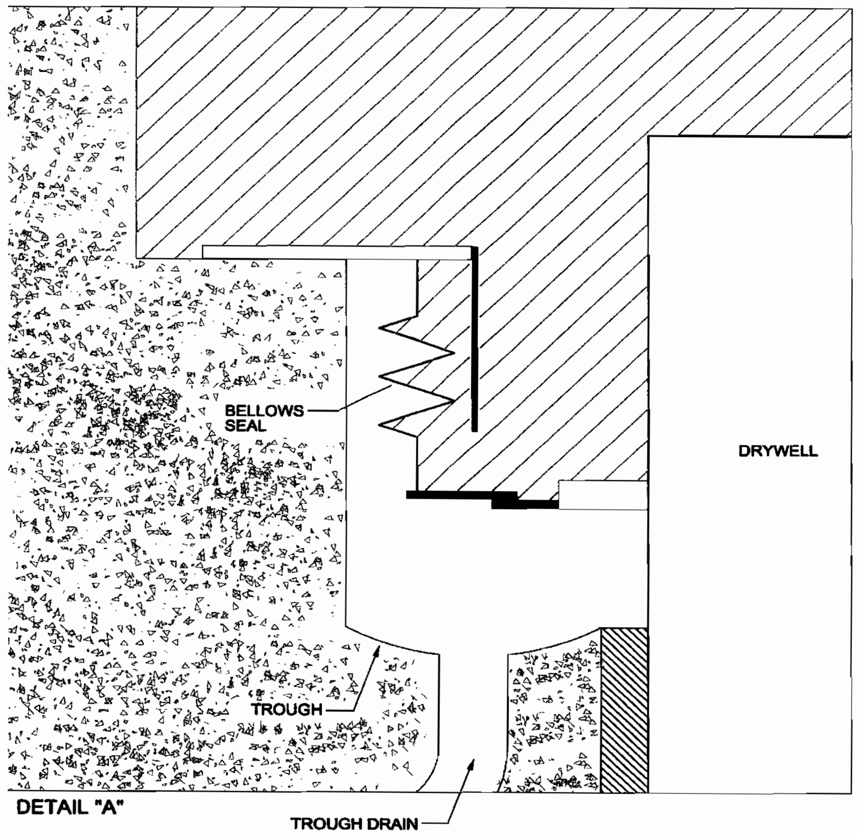
Reactor Cavity Seal Area



DETAIL "A"

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Liner Corrosion Identified

Mid 1980s

- Water leakage into the sand bed region was identified during refueling outages.
- The source was determined to be from the reactor cavity through the gap between the drywell and the reactor building, down to the sand bed region within the reactor building.
- The sand bed drains were clogged.

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Initial Corrosion Monitoring Activities

Post Mid 1980s

- Approximately 1000 UT Measurements taken to identify thinnest locations in sand bed region and upper elevations
- Core samples were taken to confirm UT measurements
  - Also confirmed that the mechanism is general corrosion
- Random UT Inspection Plan was implemented to verify adequacy of measurement locations
- Staff accepted the program in November 1, 1995 SER

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Corrective Actions Implemented

Early 1990s

- Containment Peak Pressure was reanalyzed to establish additional shell thickness margin
- The minimum acceptable shell thickness was determined
- UT measurements were taken to verify minimum thickness with margin
- Water leakage source was reduced
- The sand was removed from the sand bed region
- The sand bed drains were cleared
- The drywell shell in the sand bed region was coated

Corrective Actions Determined to be Effective
1994

- UT Measurements in 1992 and 1994 confirmed that the corrosion was arrested in the sand bed region
- 1996 UT measurements contain uncertainties – UT in 2006 will again confirm corrosion arrested
- Visual inspections of the coating were also performed

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Initial Aging Management Program

Established in Early 1990s

- Upper drywell UT measurements taken every other refueling outage
- Visual inspections of the sand bed region drywell shell coating performed every other refueling outage

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Aging Management Program

Enhanced in 2006

- Strippable coating for Reactor cavity
- Monitoring for water leakage
- Upper Drywell Shell UT measurements every other Refueling Outage
  - Leading corrosion indicator
- Sand bed region Drywell Shell UT measurements before PEO, then after 4 years and then every 10 years
  - NRC will be notified within 48 hours of any deviations outside expected results
- Sand bed region Drywell Shell coating visual inspections before PEO and 100% every 10 years

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Conclusion- Drywell Corrosion

- The corrective actions to mitigate drywell shell corrosion have been effective.
- The drywell shell corrosion was arrested in the sand bed region and continues to be very low in the upper drywell elevations.
- We have an effective aging management program to ensure continued safe operation.

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NRC SER Open Items

- Adequacy of sample size for UTs at drywell shell plate thickness transitions
- Potential corrosion of embedded shell
- Impact of corrosion on strength of drywell shell related to buckling analysis
- Use of ASME III Subsection NE-3213.10 for analysis of (localized) thin shell areas
- Extent of follow-up exams of coated sand bed surfaces if leakage is detected

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Operating History

- Reactor Vessel Internals
  - Core shroud
  - Core Spray spargers
  - Top Guide
  - Control Rod Drive Stub Tube
- Electrical cable
- Underground piping

License Renewal Methodology

- LRA submitted July 22, 2005
- NEI 95-10 Rev. 6 Standard Format
- Prepared using NUREG 1800 (SRP) and NUREG 1801 (GALL) January 2005 draft revisions
- AmerGen prepared a reconciliation document comparing the Oyster Creek LRA to NUREGs 1800 and 1801 Rev. 1.
- A third AMP/AMR audit week was added to the review

Aging Management Programs

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- 50 GALL programs
  - 18 existing
  - 14 existing requiring enhancements
  - 18 new (11 associated with Forked River Combustion Turbines and 1 with Meteorological Tower)
- 7 Plant specific programs
  - 2 existing
  - 2 existing requiring enhancements
  - 3 new (1 associated with Forked River Combustion Turbines)

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Forked River Combustion Turbines (FRCTs)

- The FRCTs are 2 peaking combustion turbines, 38 MWe each, installed in 1989
- Owned and operated by First Energy
- Credited as the Alternate AC power supply for SBO in 1992
- Covered by Maintenance Rule and Surveillance Testing Programs

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FRCTs

- Demonstrated high reliability (>99%) formed basis for initial aging management strategy
- LR application credited reliability monitoring as the aging management program
- After discussions with NRC, AmerGen elected to establish multiple GALL-based AMPs to manage aging of long-lived, passive components

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Commitment Management

- All 65 commitments are listed in Appendix A of the application.
- A Passport commitment tracking number has been issued for license renewal commitments
- An associated action containing the details was issued for each of the commitments
- Each implementing procedure is annotated to provide linkage to and preserve the details of the commitment
- Process controlled by the commitment management procedure

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Status of Program Implementation

- 257 new and 111 enhanced implementation activities identified
  - 13% in 2006 refueling outage scope
  - 19% in 2008 refueling outage scope
  - 68% to be performed on-line

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Summary

- Aging Management Programs are established to ensure safe operation for period of extended operation
- License renewal commitments are tracked and will be implemented as expected
- On track for completing activities prior to entering period of extended operation

Questions?

Advisory Committee on Reactor Safeguards (ACRS) License Renewal Subcommittee

Oyster Creek Generating Station

Safety Evaluation Report with Open Items

October 3, 2006

Donnie J. Ashley, Project Manager
Office of Nuclear Reactor Regulation

Introduction

- Overview
- Section 2: Scoping and Screening Review
- License Renewal Inspections
- Section 3: Aging Management Review Results
- Section 4: Time-Limited Aging Analyses (TLAAs)
- Confirmatory Analysis of Drywell

Overview - Status

- LRA submitted by letter, dated July 22, 2005
- LRA based on January 2005 GALL
  - Reconciliation document submitted
  - Reconciled to September 2005 GALL and SRP NUREG1800 and 1801
- SER issued August 18, 2006
- Five Open items and no Confirmatory Items
- 3 license conditions
- 108 RAIs issued, 366 audit questions
- One major component had expanded level of detail – Forked River Combustion Turbine (FRCT)

OI Cond

Overview – Audits and Inspections

- Scoping and Screening Methodology Audit
  - September 15 - 19, 2005
- AMP GALL Audit (started)
  - October 3, 2005
- AMP/AMR GALL Audit
  - January 23, February 13, and April 19, 2006.
- Regional Inspection
  - March 13 – 17 and March 27 - 31, 2006

Section 2: Scoping and Screening Review

Section 2.1 - Scoping and Screening Methodology

Section 2.2 – Plant-Level Scoping

Section 2.3 – Mechanical Systems sys

Section 2.4 – Containment, Structures, and Supports strs

Section 2.5 – Electrical Components and Commodity Groups

Section 2: Scoping and Screening Conclusion

- Scoping and screening results included all SSCs within the scope of license renewal and subject to AMR

License Renewal Inspections

Michael Modes
Region I

License Renewal Inspections

- Two–week onsite inspection during
  March 13 to March 17 and
  March 27 to March 31, 2006
- Scheduled to support NRR reviews
- Team of eight inspectors
- Inspection performed in accordance with NRC Inspection Procedure 71002

License Renewal Inspections

- Scoping and Screening
  - Concentrated on non-safety systems whose failure could impact safety systems
  - Emphasized physical walk downs of the plant
- Conclusion
  - Methodology was adequate and consistently applied

License Renewal Inspections

- Aging Management
  - 30 aging management programs plus 2 time-limited aging analyses
  - Focused on one system: Isolation Condenser
- Conclusions
  - Applicant implemented existing aging management programs as described in the application
  - Applicant provided acceptable enhancements and exceptions to the GALL report and captured them in the Oyster Creek commitment tracking system
  - In response to NRC identified inconsistencies, the Applicant revised the application or entered the inconsistencies into the corrective action program

Inspection Conclusion

- Overall, the inspection results support a conclusion that the proposed activities will reasonably manage the effects of aging in the systems, structures, and components identified in the application.

- The documentation supporting the application was in an auditable and retrievable form

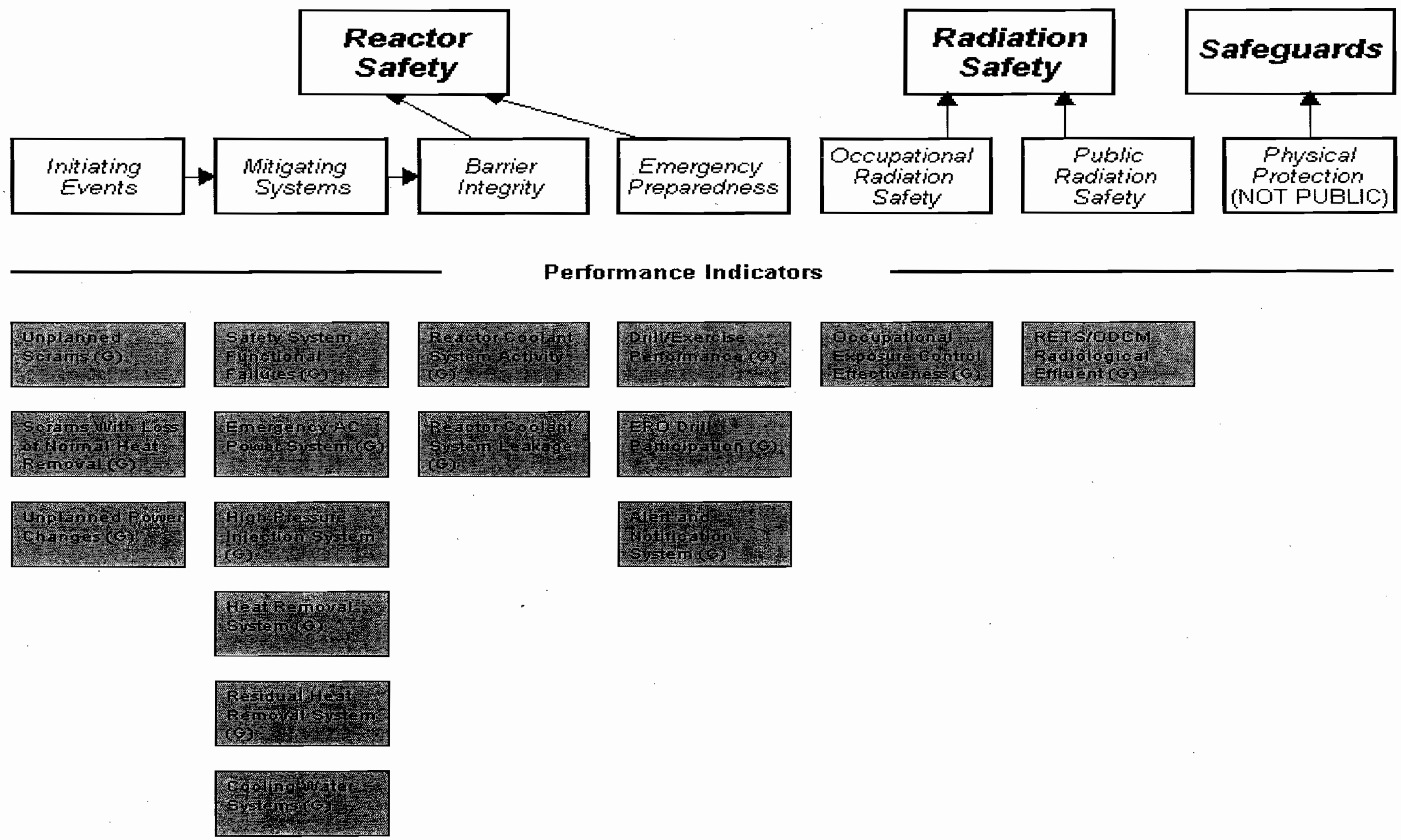
Current Performance

- Licensee is in the Regulatory Response Column (Column 2) of the NRC's Action Matrix

- The Licensee continues to follow the Revised Reactor Oversight Process

- One cross-cutting issue in the area of human performance (personnel).

Performance Indicators



Last Modified: August 2, 2006

Inspection Findings

Reactor Safety | **Radiation Safety** | **Safeguards**

Most Significant Inspection Findings

	Initiating Events	*Mitigating Systems*	*Barrier Integrity*	*Emergency Preparedness*	*Occupational Radiation Safety*	*Public Radiation Safety*	*Physical Protection (NOT PUBLIC)*
2Q/2006	G	G	G	No findings this quarter	No findings this quarter	G	
1Q/2006	G	G	No findings this quarter	No findings this quarter	G	G	
4Q/2005	No findings this quarter	G	No findings this quarter	No findings this quarter	No findings this quarter	No findings this quarter	
3Q/2005	G	No findings this quarter	G	W (1)	No findings this quarter	No findings this quarter	

Miscellaneous findings

Additional Inspection & Assessment Information

- **Assessment Reports/Inspection Plans:**
  - 2Q/2006
  - 1Q/2006
  - 4Q/2005
  - 3Q/2005
- **Cross Reference Of Assessment Reports**
- **List of Inspection Reports**
- **List of Assessment Letters/Inspection Plans**
- **Baseline Inspection Completion Information**

Last Modified: August 2, 2006

Section 3: Aging Management Programs (AMPs)

- 57 AMPs
  - 36 existing AMPs
  - 21 new AMPs (includes11 new AMPs for the FRCT)
- GALL Consistency
  - 12 Consistent with GALL Report amp
  - 38 Consistent with GALL exceptions/enhancements amp
  - 7 Plant Specific amp

Section 3 – Aging Management Example

- Protective Coating Monitoring and Maintenance Program
  - Existing plant program - Consistent with GALL AMP XI.S8, "Protective Coating Monitoring and Maintenance Program"
  - Credited for Drywell and Torus aging management
  - The inspection of 100% of the sandbed region epoxy coating before the PEO and every 10 years during the period of extended operation.
    - Inspections will be staggered such that at least three bays will be examined every other refueling outage.
  - The inspection of all 20 torus bays at a frequency of every other refueling outage for the current coating system. Should the current coating system be replaced, the inspection frequency and scope will be re-evaluated. Inspection scope will meet the requirements of ASME Section XI, Subsection IWE.

Section 3 – Aging Management Example

- Structures Monitoring Program
  - Existing program credited
  - 17 Commitments identified
  - The program includes elements of the Masonry Wall Program and the RG 1.127, Inspection of Water-Control Structures Associated With Nuclear Power Plants Program.
  - This program also includes structures for Station Blackout system, phase bus enclosure assemblies, and fire protection Communication System structures (Met-Tower)

Section 3 – Aging Management Example

- Periodic Monitoring of Combustion Turbine Power Plant - Station Blackout – FRCT
  - 10 Commitments to implement the Aging Management programs.

C

Section 3: Aging Management Review Overview

3.1	Reactor Vessel, Internals and Coolant System	6 systems
3.2	Engineered Safety Features	3 systems
3.3	Auxiliary Systems	41 systems
3.4	Steam and Power Conversion Systems	7 systems
3.5	Containments, Structures, and Component Supports	19 structures
3.6	Electrical and I&C Components	8 groups
3.7	Station Blackout System (Forked River Combustion Turbines), Radio Communications System, and Meteorological Tower (Met Tower was added to scope)	

Aging Management – Drywell Shell

- Three Aging Management Programs
  - ASME Section XI, Subsection IWE
  - Protective Coating Monitoring and Maintenance Program
  - 10 CFR 50 Appendix J Programs

- UT of sand-pocket region performed in 1992 and 1994 determined that corrosion rates had been arrested.

- Water leakage monitoring program (each refueling)
  - refueling seal
  - drywell air gap drains
  - sand pocket drains

- 11 Commitments for Drywell

Com OIs

Section 3 – Aging Management of In-Scope Inaccessible Concrete

	Acceptance Criteria	OCGS
pH	>5.5	5.6 – 6.4*
Chlorides	<500 ppm	3 - 138
Sulfates	<1500 ppm	7 - 73

- * Below-grade environment is non-aggressive except for fresh water pump-house
- Periodic testing of ground water will be performed for Structures Monitoring Program

Section 4 Time-Limited Aging Analyses (TLAA)

- 4.1 TLAA Process
- 4.2 Neutron Embrittlement of the RPV and Internals
- 4.3 Metal Fatigue
- 4.4 Environmental Qualification of Electrical Equipment
- 4.5 Loss of Prestress in Concrete Containment Tendons (N/A)
- 4.6 Fatigue Analysis of Primary Containment
- 4.7 Plant Specific TLAAs
  - 4.7.1 Crane Load Cycle Limit
  - 4.7.2 Drywell Corrosion
  - 4.7.3 Equipment Pool/Reactor Cavity Wall Rebar Corrosion
  - 4.7.4 Reactor Vessel Weld Flaw Evaluations
  - 4.7.5 CRD Stub Tube Flaw Analysis

Section 4.2 Neutron Embrittlement

Reactor Vessel Upper Shelf Energy (USE) – Analysis Summary

OCGS Reactor Vessel Material	Percent USE Reduction of OCGS Reactor Vessel Material	Percent USE Reduction Acceptance Criterion*	Evaluation Result
Limiting Plate 564-03D, E, F	29%	USE drop must be < 29.5%	Acceptable pursuant to 10 CFR 54.21(c)(1)(ii)
Limiting Weld 86054B & 1248	32%	USE drop must be < 39%	Acceptable pursuant to 10 CFR 54.21(c)(1)(ii)

*acceptance criteria established per BWRVIP-74.

Section 4.2.4 Reactor Vessel Circumferential Weld Examination Relief

RV Circumferential Weld Relief/ RV Axial Weld Probability of Failure Analyses

RV Material	TLAA Basis	Acceptance Criterion (°F)	OCGS Value (°F)
Limiting Circ. Weld	BWRVIP-05 Mean RT_{NDT} Value (°F)	<128.5	9.8
Limiting Axial Weld	BWRVIP-05 Mean RT_{NDT} Value (°F)	<114	50.3

- TLAAs for the Circ. Weld and Axial Weld Mean RT_{NDT} values were acceptable pursuant to 10 CFR 54.21(c)(1)(ii)

Section 4.3: Metal Fatigue

- Cumulative Usage Factor is projected to be less than ASME Code limit of 1.0 for components based on a 60-year life.
- Monitored by the Fatigue Monitoring Program
- Staff accepted the evaluations

Section 4.4 Environmental Qualification (EQ) of Electrical Equipment

- Applicant's EQ Program consistent with GALL AMP X.E1, "Environmental Qualification of Electrical Equipment"

- Staff concluded the EQ Program is adequate to manage the effects of aging on the intended function of electrical components

Section 4.6 Fatigue Analysis of Primary Containment

- Staff accepted the evaluations in accordance with 10 CFR 54.21(c)(1)(i)

Section 4.7.5 CRD Stub Tube Flaw Analysis

- Staff accepted the evaluations in accordance with 10 CFR 54.21(c)(1)(i)

TLAA Summary

- 10 CFR 54.3 - TLAA is list adequate
- 10 CFR 54.21(c)(2) – no plant-specific exemptions

Section 4.7.2 Drywell Corrosion

- On the basis of its review, the staff concludes that, pending resolution of <u>OIs</u> 4.7.2-1.1, 4.7.2-1.2, 4.7.2-1.3, 4.7.2-1.4, and 4.7.2-3, the applicant has demonstrated, that for the drywell corrosion TLAA, the effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

Confirmatory Analysis
Oyster Creek Drywell

Hans Ashar

NRR

Conclusions

- The staff has concluded that pending resolution of the open items, there is reasonable assurance that the activities authorized by the renewed license will continue to be conducted in accordance with the CLB, and that any changes made to the OCGS CLB in order to comply with 10 CFR 54.29(a) are in accord with the Act and the Commission's regulations.

OFFICE OF NUCLEAR REACTOR REGULATION

UNITED STATES NUCLEAR REGULATORY COMMISSION

5 open items:

- **<u>OI 4.7.2-1.1:</u>** Drywell Corrosion Sampling in the transition area. Question on the appropriate number of locations on the drywell for periodic ultrasonic testing
- **<u>OI 4.7.2-3:</u>** Questions about the implementation of the Protective Coating Monitoring and Maintenance Program. The extent of inspections of epoxy-coated drywell surfaces

- **<u>OI 4.7.2-1.2:</u>** Drywell Corrosion Inaccessible areas embedded concrete. the possibility of corrosion of drywell liner plates embedded in concrete between the containment floor and foundation
- **<u>OI 4.7.2-1.3:</u>** Buckling Analysis. the appropriateness of certain technical assumptions in AmerGen's analysis of the potential for "buckling," of the drywell shell
- **<u>OI 4.7.2-1.4:</u>** Drywell Shell Thickness and the Minimum Available Thickness Margin. The use of an ASME Code provision to simulate the behavior in thinned areas

Go to 3.5 Go to 4.7.2

BACK

License Conditions:

- The first license condition requires the applicant to include the UFSAR supplement required by 10 CFR 54.21(d) in the next UFSAR update, as required by 10 CFR 50.71(e), following the issuance of the renewed license.
- The second license condition requires future activities identified in the UFSAR supplement to be completed prior to the period of extended operation.
- The third license condition requires all surveillance capsules placed in storage to be maintained for future insertion. Any changes to storage requirements must be approved by the staff as required by 10 CFR Part 50, Appendix H.

BACK

FRCT New AMPs and Commitments

Bolting Integrity	B.1.12A
Closed Cycle Cooling Water System	B.1.14A
Above Ground Steel Tanks	B.1.21A
Fuel Oil Chemistry	B.1.22A
One Time Inspection	B.1.24A
Selective Leaching of Materials	B.1.25A
Buried Piping Inspection	B.1.26A
Periodic Monitoring of FRCT – Electrical	B.1.37
Inspection of Piping and Ducts	B.1.38A
Lubricating Oil Analysis Program	B.1.39A
Periodic Inspection	B.2.5A

BACK

TLAA Criteria

- *Time-limited aging analyses*, for the purposes of this part, are those licensee calculations and analyses that:

(1) Involve systems, structures, and components within the scope of license renewal;

(2) Consider the effects of aging;

(3) Involve time-limited assumptions defined by the current operating term, for example, 40 years;

(4) Were determined to be relevant by the licensee in making a safety determination;

(5) Involve conclusions or provide the basis for conclusions related to the capability of the system, structure, and component to perform its intended functions, as delineated in § 54.4(b); and

(6) Are contained or incorporated by reference in the CLB.

BACK

Scoping and Screening Systems

- **Reactor Vessel, Internals, and Reactor Coolant System (8)**
  - Control Rods
  - Fuel Assemblies
  - Isolation Condenser System
  - Nuclear Boiler Instrumentation
  - Reactor Head Cooling System
  - Reactor Internals
  - Reactor Pressure Vessel
  - Reactor Recirculation System

- **Engineered Safety Features Systems(4)**
  - Automatic Depressurization System
  - Containment Spray System
  - Core Spray System
  - Standby Gas Treatment System (SGTS)

- **Auxiliary Systems (41)**
  - "C" Battery Room Heating & Ventilation
  - 4160V Switchgear Room Ventilation
  - 480V Switchgear Room Ventilation
  - Battery and MG Set Room Ventilation
  - Chlorination System
  - Circulating Water System
  - Containment Inerting System
  - Containment Vacuum Breakers
  - Control Rod Drive System
  - Control Room HVAC
  - Cranes and Hoists
  - Drywell Floor and Equipment Drains
  - Emergency Diesel Generator & Aux Sys
  - Emergency Service Water System
  - Fire Protection System
  - Fuel Storage and Handling Equipment
  - Hardened Vent System
  - Heating & Process Steam System
  - Hydrogen & Oxygen Monitoring System
  - Instrument (Control) Air System
  - Main Fuel Oil Storage & Transfer sys
  - Misc.Floor and Equipment Drain System
  - Nitrogen Supply System
  - Noble Metals Monitoring System
  - Post-Accident Sampling System
  - Process Sampling System
  - Radiation Monitoring System
  - Radwaste Area Heat&Vent System
  - Reactor Building CCWater System
  - Reactor Building Floor and Equipment Drains
  - Reactor Building Ventilation System
  - Reactor Water Cleanup System
  - Roof Drains and Overboard Discharge
  - Sanitary Waste System
  - Service Water System
  - Shutdown Cooling System
  - Spent Fuel Pool Cooling System
  - Standby Liquid Control System (Liquid Poison System)
  - Traveling In-Core Probe System
  - Turbine Building CCW System
  - Water Treatment & Distribution System

- **Steam and Power Conversion Systems (7)**
  - Condensate System
  - Condensate Transfer System
  - Feedwater System
  - Main Condenser
  - Main Generator and Auxiliary System
  - Main Steam System
  - Main Turbine and Auxiliary System

Structures in Scope

- **Structures**
  - Primary Containment
  - Condensate Transfer Building
  - Emerg Diesel Generator Building
  - Fire Pumphouses
  - Intake Structure and Canal (Ultimate Heat Sink)
  - New Radwaste Building
  - Turbine Building
  - Reactor Building
  - Dilution Structure
  - Exhaust Tunnel
  - Heating Boiler House
  - Miscellaneous Yard Structures
  - Office Building
  - Ventilation Stack
  - Chlorination Facility
  - Fire Pond Dam
  - Oyster Creek Substation

- **Component Supports Commodity Group**
  - In its responses dated October 12, November 11, and December 9, 2005, and May 18 and June 7, 2006, the applicant stated that it had determined that the repeater located at the Meteorological Tower (Met Tower) is credited for communication capabilities for some 10 CFR Part 50, Appendix R, scenarios. Therefore, the repeater and associated support equipment, including the backup gas (propane) engine generator located at the Met Tower, are now within the scope of license renewal and subject to an AMR.

- **Electrical Systems and Electrical Commodity Group**
  - Credit for STATION BLACKOUT EQUIPMENT
  - In LRA Table 2.5.1.19, the ACC combustion turbines are identified as one combustion turbine power plant unit within the scope of license renewal and subject to an AMR. As described in SER Section 2.5.5.2, in its response to RAI 2.5.1.19-1, the applicant stated that it had revised the combustion turbine power plant unit scoping and screening methodology. Mechanical, electrical, and structural component types were itemized in detail consistent with scoping and screening methodology for other the other license renewal systems and structures.

Consistent with GALL 12 out of 57

- Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS) (B.1.10) – New
- Flow-Accelerated Corrosion (B.1.11)
- Compressed Air Monitoring (B.1.17)
- One-Time Inspection (B.1.24) - New
- Selective Leaching of Materials (B.1.25) – New
- 10 CFR Part 50, Appendix J (B.1.29)
- Masonry Wall Program (B.1.30)
- Protective Coating Monitoring and Maintenance Program (B.1.33)
- Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements (B.1.34) – New
- Inaccessible Medium-Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements (B.1.36) - New
- Environmental Qualification (EQ) Program (B.3.2)
- Electrical Cable Connections - Metallic Parts - Not Subject to 10 CFR 50.49 Environmental Qualification Requirements (B.1.40) - New

Consistent With Exceptions/Enhancements

- ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B.1.1)
- Water Chemistry (B.1.2)
- Reactor Head Closure Studs (B.1.3)
- BWR Vessel ID Attachment Welds (B.1.4)
- BWR Feedwater Nozzle (B.1.5)
- BWR Control Rod Drive Return Line Nozzle (B.1.6)
- BWR Stress Corrosion Cracking (B.1.7)
- BWR Penetrations (B.1.8)
- BWR Vessel Internals (B.1.9)
- Bolting Integrity (B.1.12)
- Open-Cycle Cooling Water System (B.1.13)
- Closed-cycle Cooling Water System (B.1.14)
- Boraflex Rack Management Program (B.1.15)
- Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems (B.1.16)
- BWR Reactor Water Cleanup System (B.1.18)
- Fire Protection (B.1.19)
- Fire Water System (B.1.20)
- Above Ground Outdoor Tanks (B1.21)
- Fuel Oil Chemistry (B1.22)
- Reactor Vessel Surveillance (B.1.23)
- One Time Inspection (B1.24)
- Selective Leaching of Materials(B1.25)
- Buried Piping Inspection (B.1.26)
- Buried Piping and Tank Inspection (MetTower B1.26B)
- ASME Section XI, Subsection IWE (B.1.27)
- ASME Section XI, Subsection IWF (B.1.28)
- Structures Monitoring Program (B.1.31)
- Inspection of Water-Control Structures (B.1.32)
- Electrical Cables and Connections Not Subject to E.Q. Used in Instrument Circuits (B.1.35)
- Metal Fatigue of Reactor Coolant Pressure Boundary (B.3.1)

- Bolting Integrity - FRCT (B.1.12A) - New
- Aboveground Steel Tanks - FRCT (B.1.21A) - New
- Closed-cycle Cooling Water System - FRCT (B.1.14A) - New
- Fuel Oil Chemistry - FRCT (B.1.22A) - New
- One-Time Inspection - FRCT (B.1.24A) - New
- Buried Piping Inspection - FRCT (B.1.26A) - New
- Selective Leaching of Materials - FRCT (B.1.25A) – New
- Periodic Monitoring of Combustion Turbine –FRCT(B.1.37A)
- Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components - FRCT (B.1.38A) - New
- Lubricating Oil Analysis Program - FRCT (B.1.39) - New
- Buried Piping and Tank Inspection-Met Tower Repeater Engine Fuel Supply (B.1.26B) - New

Plant-Specific (7 out of 57)

- Periodic Testing of Containment Spray Nozzles (B.2.1)
- Lubricating Oil Monitoring Activities (B.2.2)
- Generator Stator Water Chemistry Activities (B.2.3)
- Periodic Inspection of Ventilation Systems (B.2.4)
- Periodic Inspection Program (B.2.5) – New
- Wooden Utility Pole Program (B.2.6) - New
- Periodic Inspection Program - FRCT (B.2.7) - New

Drywell Commitments

- Ultrasonic Testing (UT) thickness measurements of the drywell shell in the sand bed region will be performed once prior to the PEO another four years later and then a frequency of every 10 years
- Consistent with current practice, a strippable coating will be applied to the reactor cavity liner to prevent water intrusion into the gap between the drywell shield wall and the drywell shell during periods when the reactor cavity is flooded.
- The reactor cavity seal leakage trough drains and the drywell sand bed region drains will be monitored for leakage.
  - The sand bed region drains will be monitored daily during refueling outages.
- Prior to the period of extended operation, AmerGen will perform additional visual inspections of the epoxy coating that was applied to the exterior surface of the Drywell shell in the sand bed region, such that the coated surfaces in all 10 Drywell bays will have been inspected at least once.

Drywell Commitments

- Prior to the period of extended operation, AmerGen will perform additional visual inspections of the epoxy coating that was applied to the exterior surface of the Drywell shell in the sand bed region, such that the coated surfaces in all 10 Drywell bays will have been inspected at least once.
- A visual examination of the drywell shell in the drywell floor inspection access trenches will be performed to assure that the drywell shell remains intact.
- Conduct UT thickness measurements in the upper regions of the drywell shell every other refueling outage at the same locations as are currently measured.
- During the next UT inspections to be performed on the drywell sand bed region, an attempt will be made to locate and evaluate some of the locally thinned areas identified in the 1992 inspection from the exterior of the drywell.

Drywell Commitments

- Conduct UT thickness measurements on the 0.770 inch thick plate at the junction between the 0.770 inch thick and 1.154 inch thick plates, in the lower portion of the spherical region of the drywell shell.

- Conduct UT thickness measurements in the drywell shell “knuckle” area, on the 0.640 inch thick plate above the weld to the 2.625 inch thick plate.

- When the sand bed region drywell shell coating inspection is performed, the seal at the junction between the sand bed region concrete and the embedded drywell shell will be inspected per the Protective Coatings Program.

- The reactor cavity concrete trough drain will be verified to be clear from blockage once per refueling cycle

BACK

Sandia Analysis Modeling Assumptions

Model Geometry

- 360 model of drywell and vent lines
- Torus not modeled
- Equipment Hatch and 10 vent lines are modeled

mod1

General loads

- Model includes gravity and dead loads
- Other dead loads were taken from previous GE analysis
- Seismic loads included with static coefficients from FSAR

Controlling Load Cases

- Refueling
- Design Basis Accident with Earthquake
- Post Accident Flooding with Earthquake

Modeling Corrosion

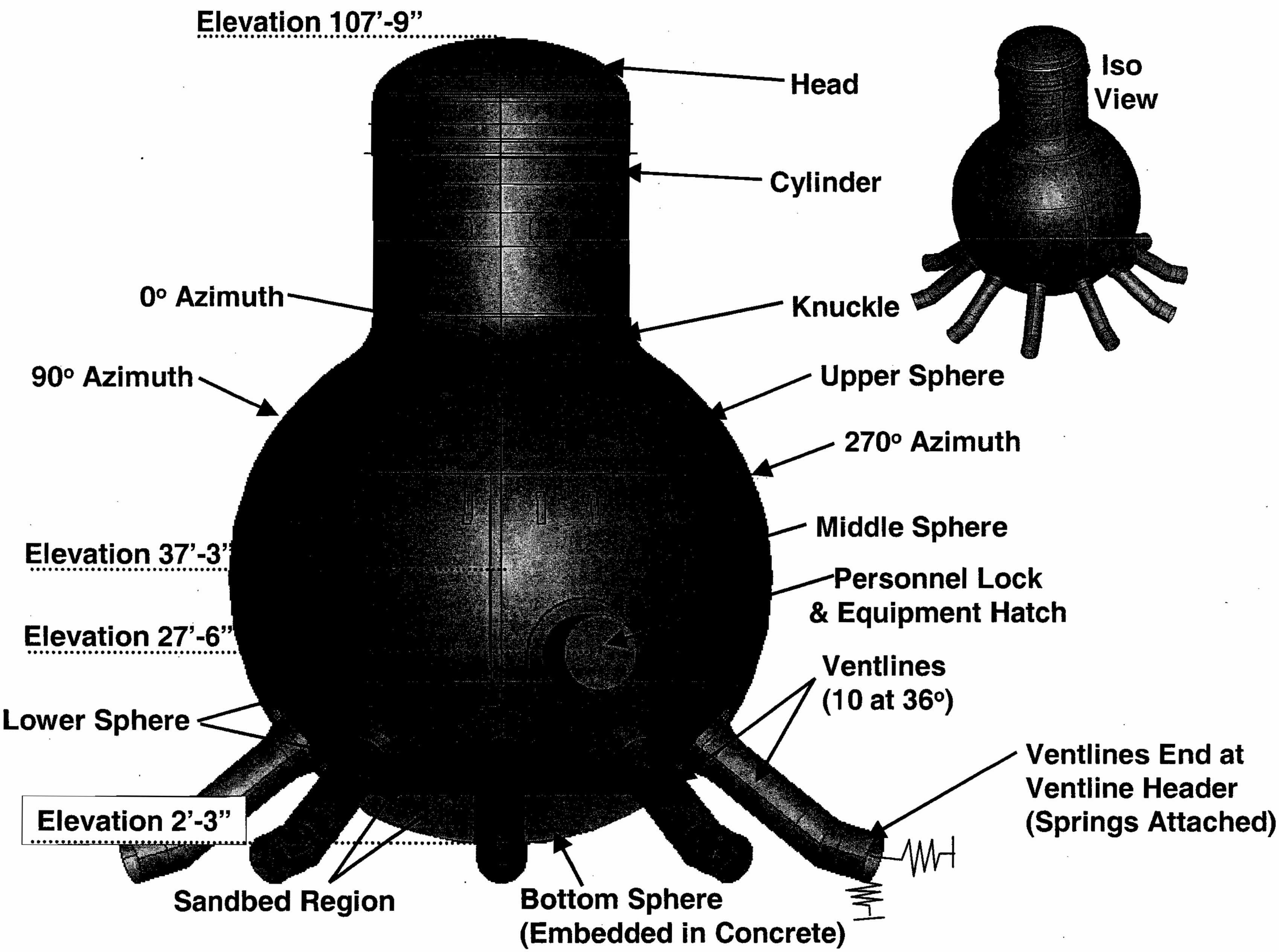
mod2

- Cylinder, upper sphere, and middle sphere assigned uniform thicknesses
- Thickness is based on extrapolated UT measurements
- Lower sphere assumed 10 regions
- Each region assigned thickness based on average of UT data points

Sandia Analysis Preliminary Results

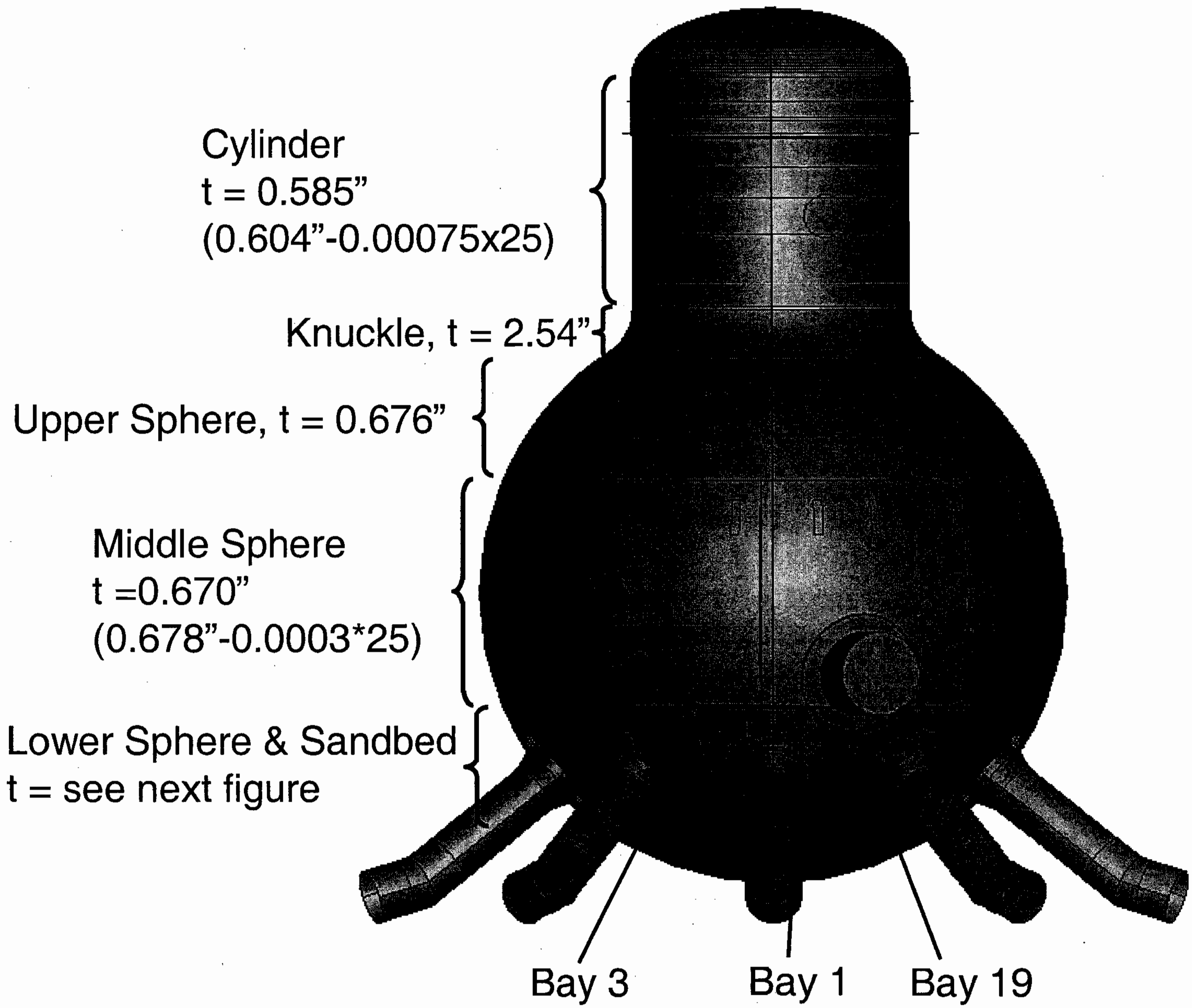
- **Refueling Load Combination (dead + live + seismic+ refueling loads)**
  - All stresses within ASME Service Level B requirements
  - Sandbed buckling (factor of safety of 2.00 required in ASME – 284)
    - With no degradation SF=3.85
    - With degradation SF=2.15
- **Accident Load (dead + internal pressure + thermal + seismic loads)**
  - All stresses within ASME Service Level C requirements
  - Buckling is not controlling
- **Post-Accident Load Case**
  - All stresses within ASME Service Level D requirements
  - Sandbed buckling (factor of safety of 1.67 required in ASME – 284)
    - With no degradation SF=3.65
    - With degradation SF=2.74

Oyster Creek Drywell Model



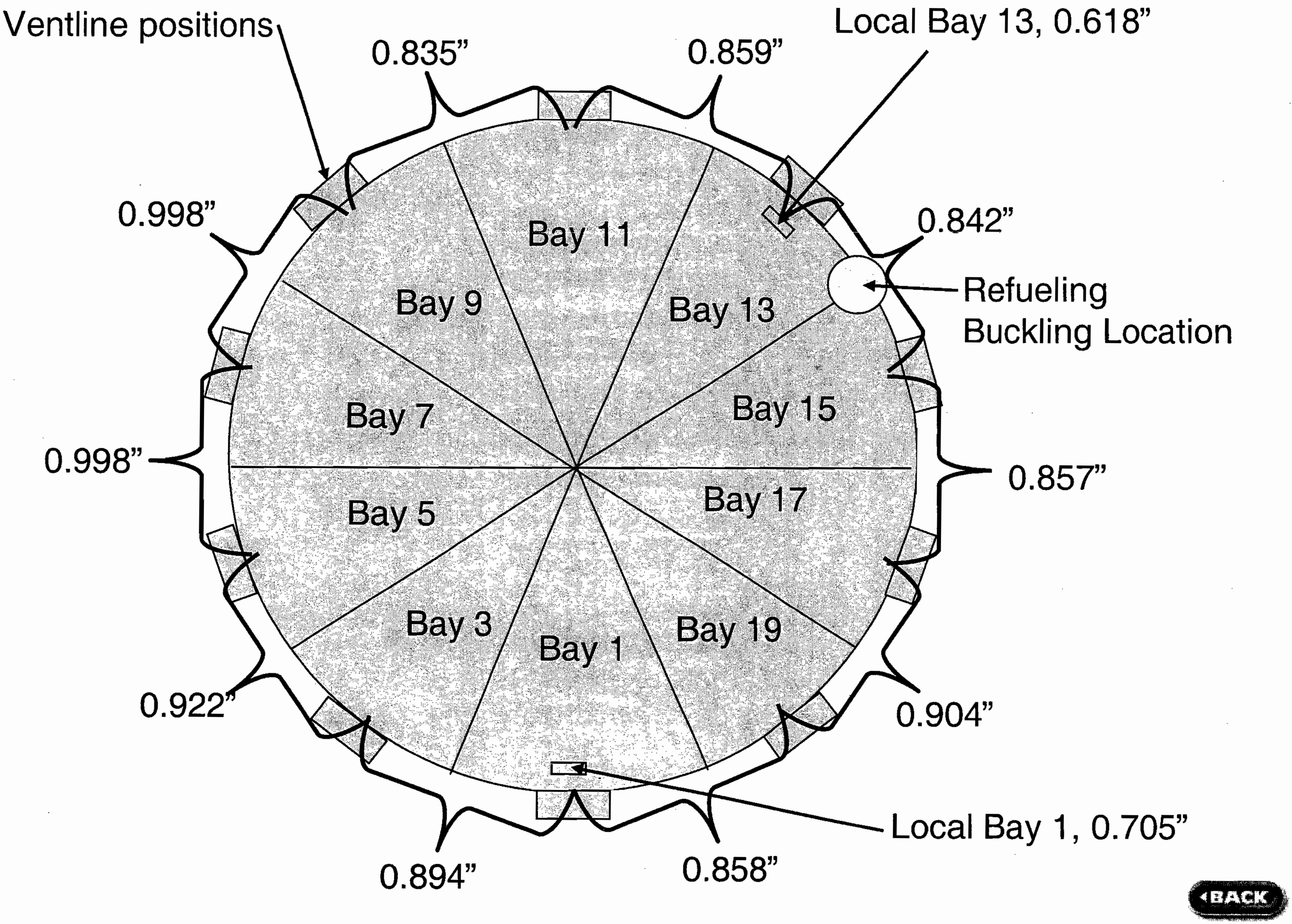
BACK

Oyster Creek Drywell Model – Assumed Thickness



BACK

Estimated Thicknesses in the Lower Sphere



From: Bill Hering <bill.hering@smelectric.com>
To: "'cxs3@nrc.gov'" <cxs3@nrc.gov>
Date: 10/03/2006 11:09:36 AM
Subject: OYSTER CREEK

Gentlemen, Thank's for the opportunity to comment on the Oyster Creek license application regarding the meeting in Rockville this afternoon.

My history with Oystercreek goes back many years, and briefly, my background is 40 years active in the IBEW construction trades - Current today LU 164 Jersey City - and I am a professional Occupational Safety & Health Trainer with the US Department of Labor both Mine Safety and Health Administration and OSHA - Past President of the American Society of Safety Engineers New Jersey Chapter 1999 / 2000, and I was the Project Manager 15 years ago when we built the Nuclear Simulator at Oyster Creek, and the Safety Manager overseeing the Homeland Security upgrade with the electrical contractor, two years ago, at the Oyster Creek facility.

My Comments are first 100% in favor of License Renewal, based on every single thing I had the opportunity to see and be part of with this facility. Yes, it's the oldest facility, but gentlemen, the safety record based on industry standards is impeccable. For that matter, the Nuclear Industry has had a fabulous record in the entire country, thanks to the oversight by the NRC and committee's as your's.

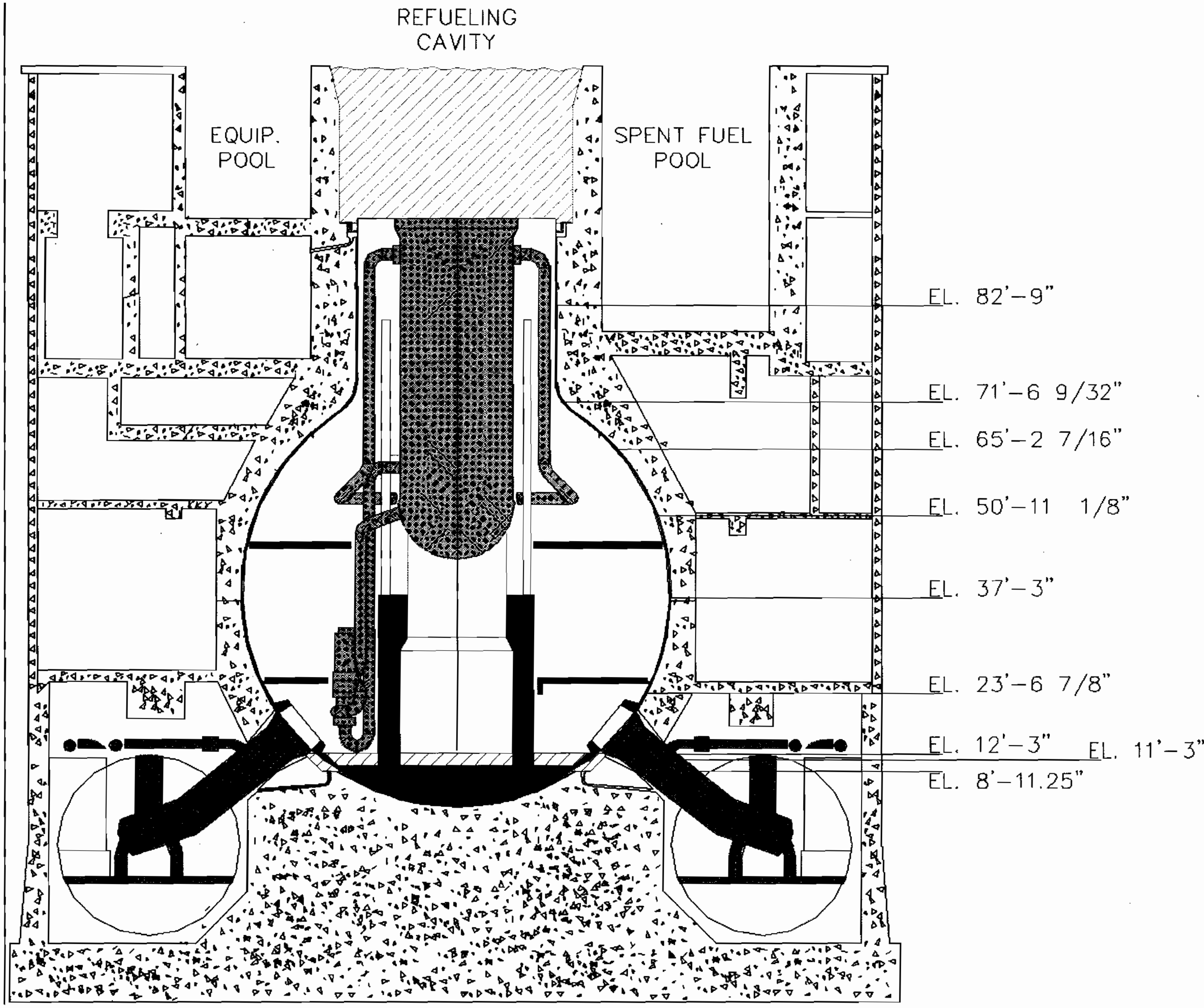
If the various components of this plant are meeting these industry standards and with new technology constantly at our doorstep to enhance these safety benchmarks, we need to have a common sense approach to these re-newals for this industry. Various opponents to this application and other plants operations seem to have a mission which is either to far right or to far left... I'll leave that to your judgment...in short this plant is a safe reliable source of 650 megawatts of clean electrical energy that is in dire demand in our state.

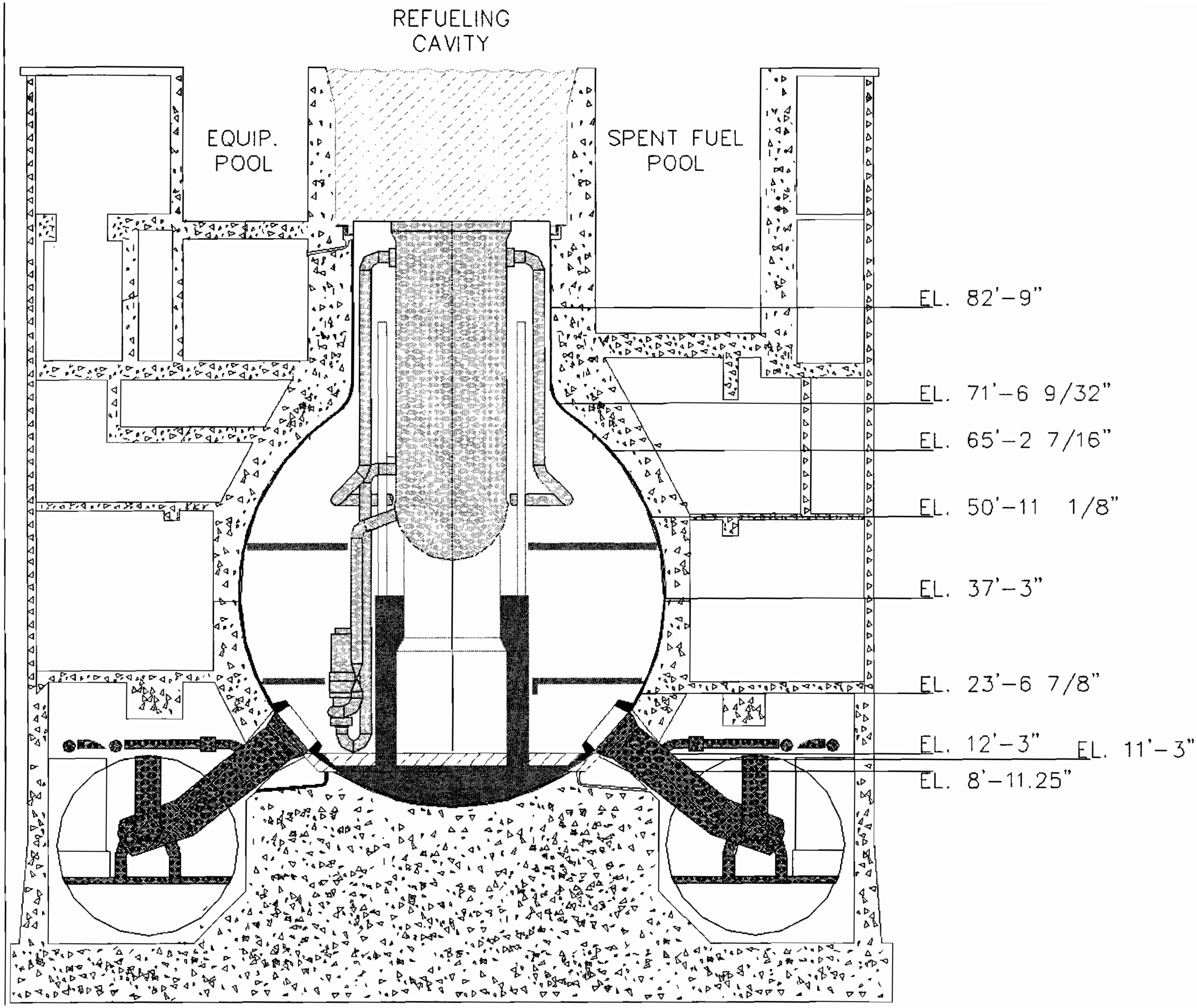
The license process by statue is 20 years - HOWEVER - I believe that perhaps in 5 years or so, a new type reactor will be finalized and hopefully replace older reactors on all Nuclear sites in the future in our great nation.

Another very interesting point is that recent polls concerning the license renewal of Oyster Creek have shown independently over 80 % of those from 5 miles, 10 miles and the State as a whole have registered IN FAVOR of the plants license renewal.

As far as homeland security, this plant has the latest in technology as I witnessed it being built and certified.
The spent fuel is on site either way, so again - common sense - let's keep the plant running as the spent fuel will be there anyway for some time until all the bug's are cleared with YUCCA Mountain and security will need to remain in tact.

I think I've said enough. Gentlemen, PLEASE TAKE ALL THESE POINTS INTO CONSIDERATION AND UNDERSTAND THAT THE NEGATIVE COMMENTS SEEM TO ONLY REPRESENT A SMALL MINORITY OF OUR POPULATION, WE NEED NOT ONLY RELICENSE EXISTING, BUT BUILD MORE NUCLEAR Facilities ASAP AS THE NEED FOR CLEAN


REFUELING
CAVITY
EQUIP.
POOL
SPENT FUEL
POOL
EL. 82'-9"
EL. 71'-6 9/32"
EL. 65'-2 7/16"
EL. 50'-11 1/8"
EL. 37'-3"
EL. 23'-6 7/8"
EL. 12'-3"
EL. 11'-3"
EL. 8'-11.25"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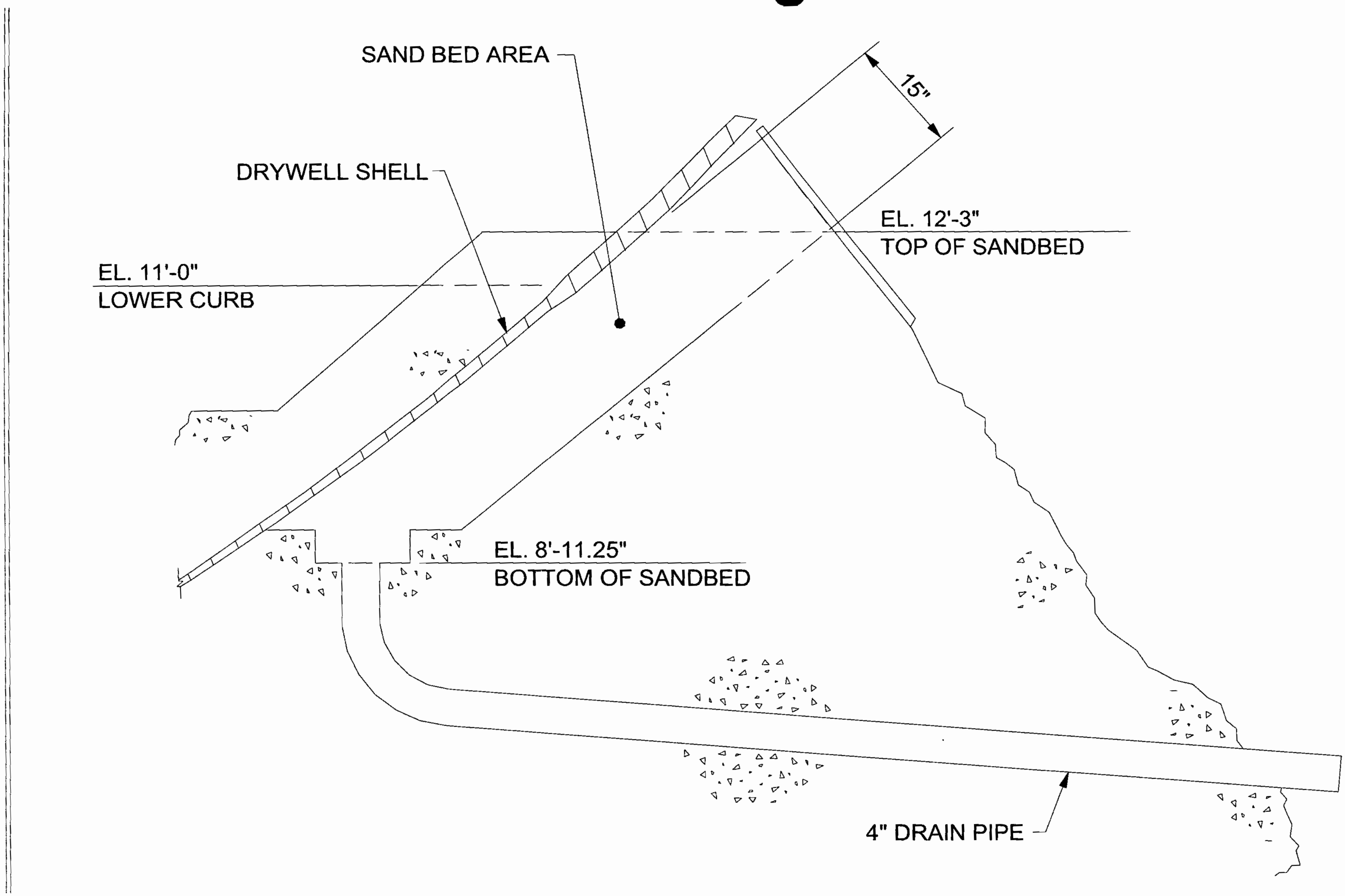
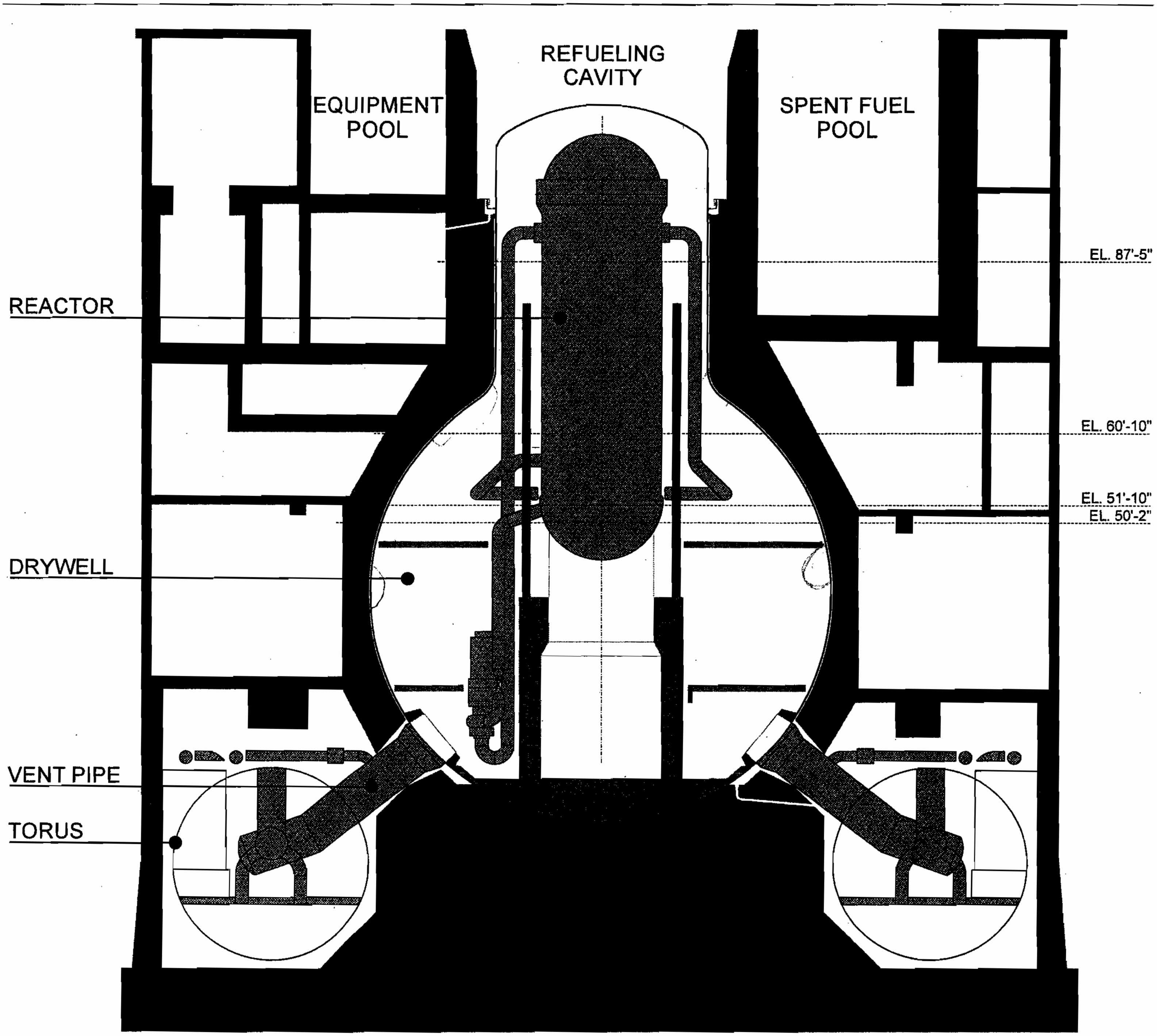
EL. 65'-2 7/16"
EL. 50'-11 1/8"
EL. 37'-3"
EL. 23'-6 7/8"
EL. 12'-3"
EL. 11'-3"
EL. 8'-11.25"

Backup Slides

AmerGen
An Exelon Company

Sand Bed Region





\sand bed\sheet2.dgn Feb. 16, 2006 13:19:48

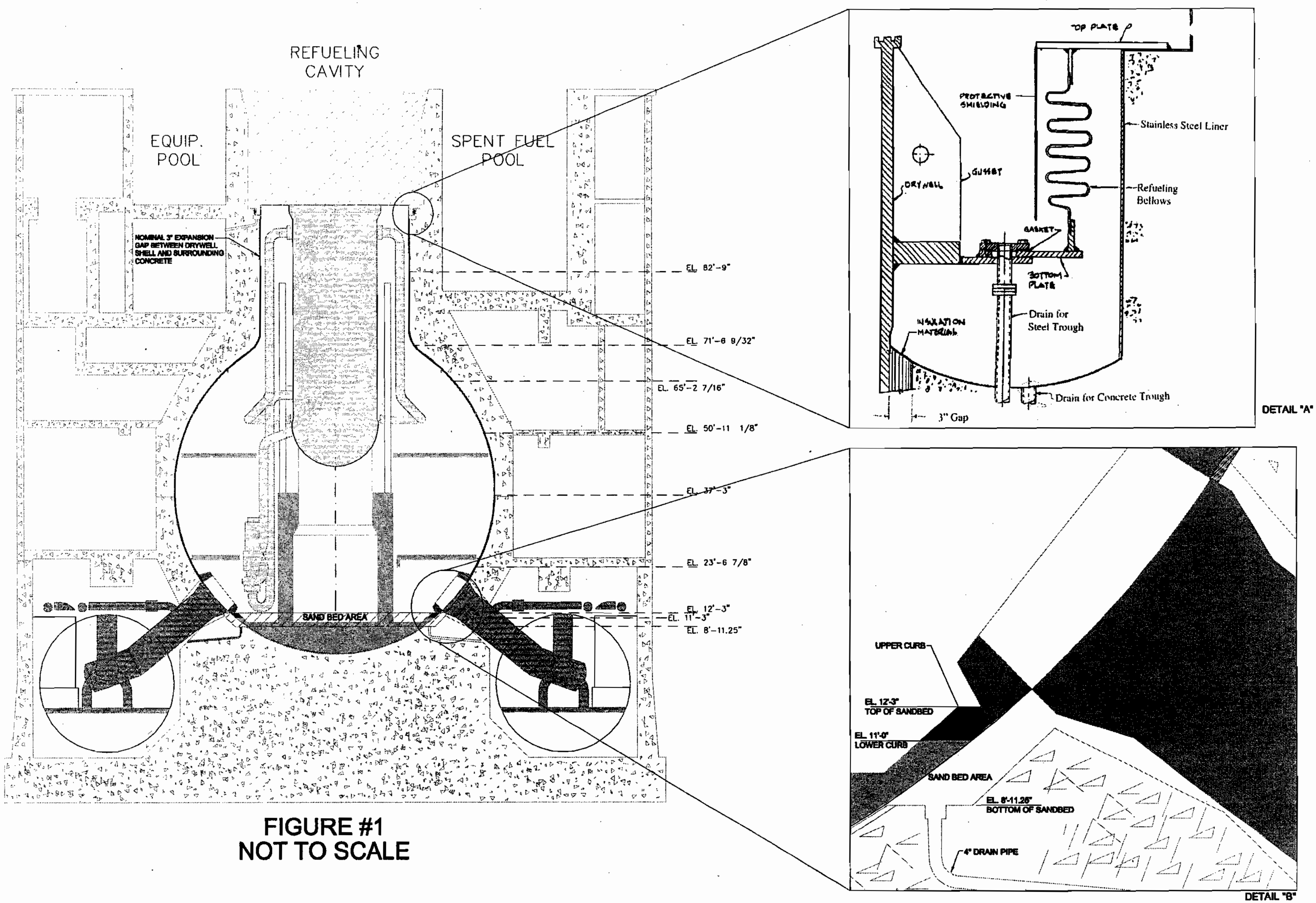
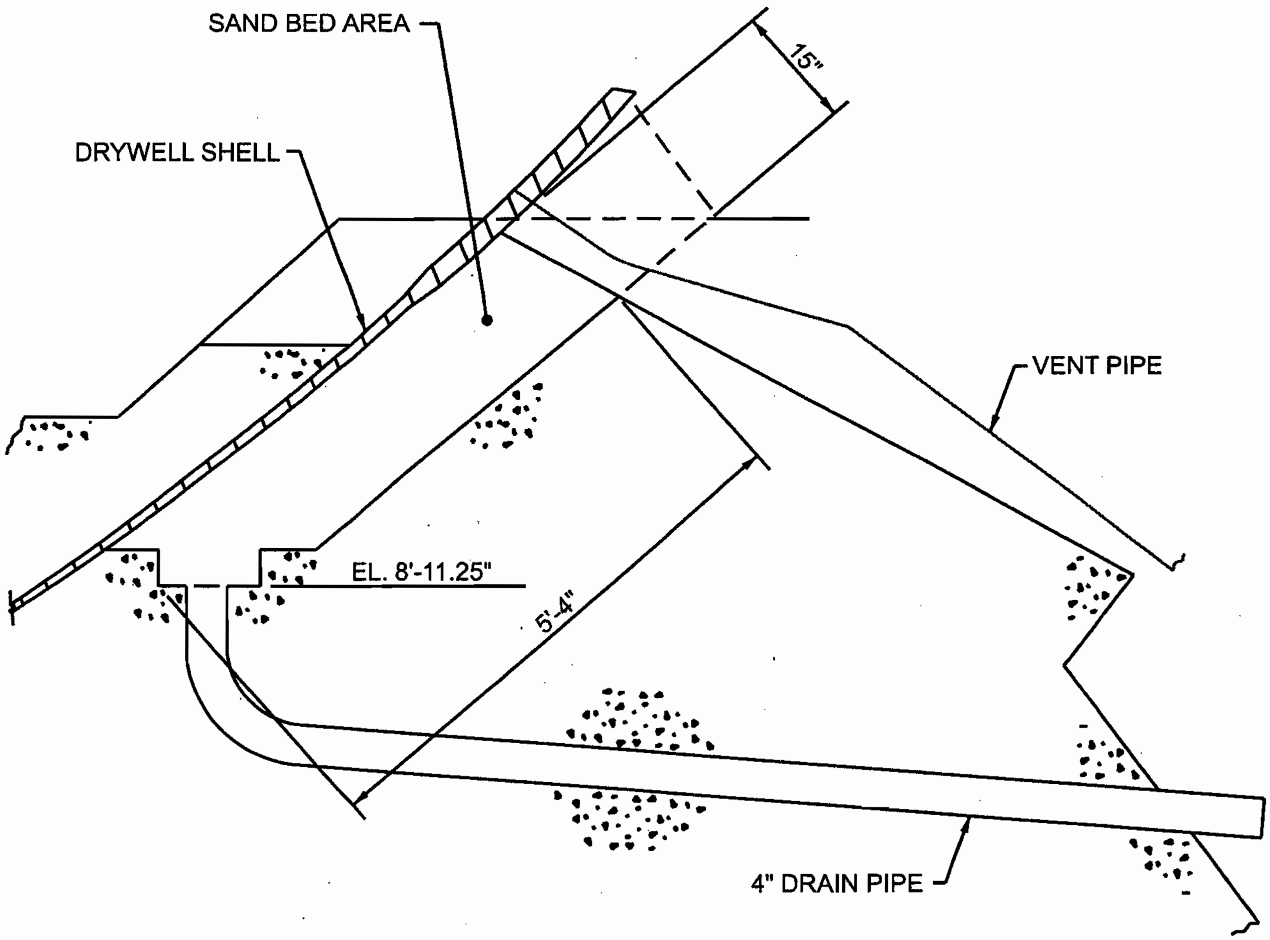
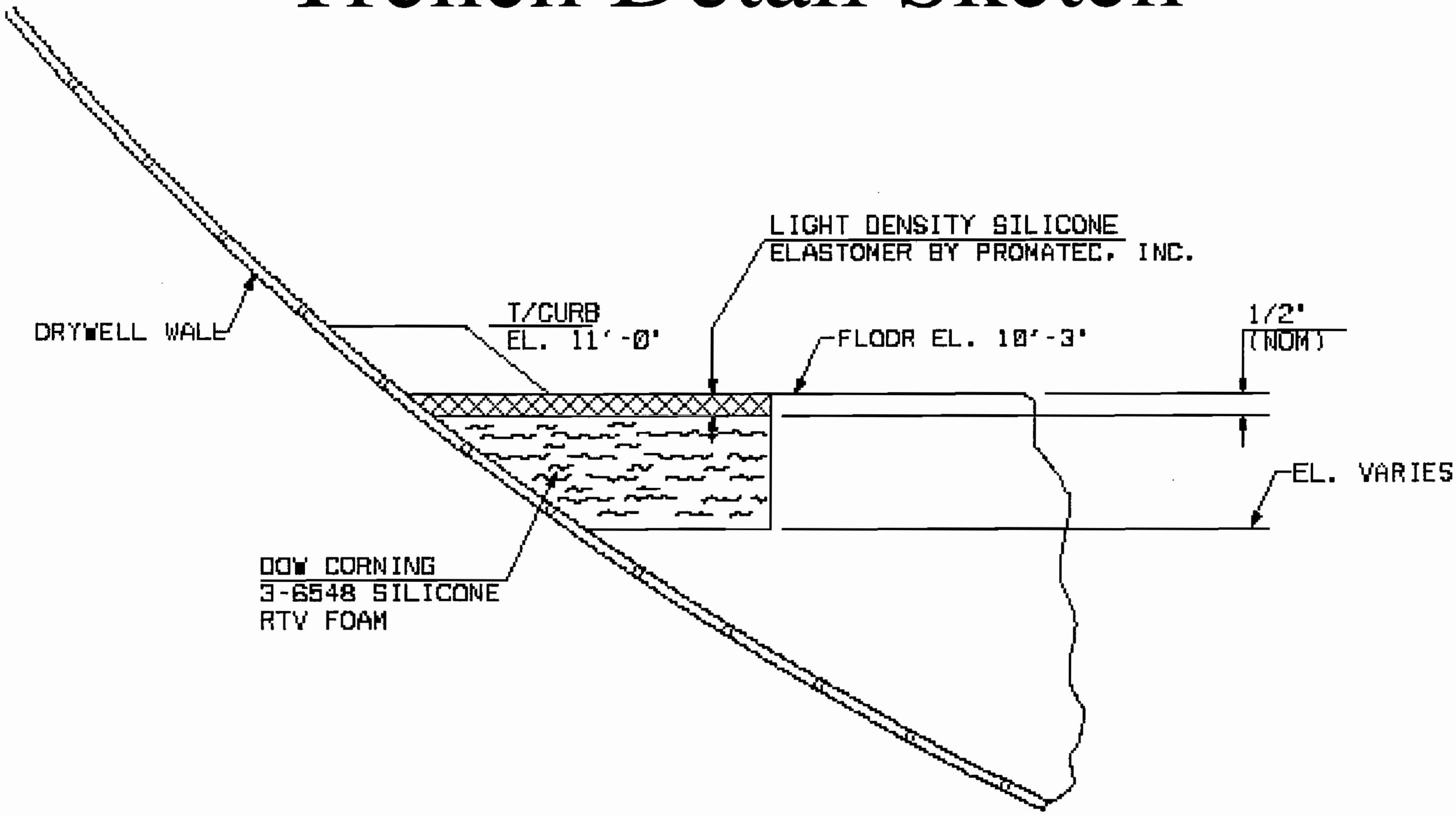


FIGURE #1
NOT TO SCALE



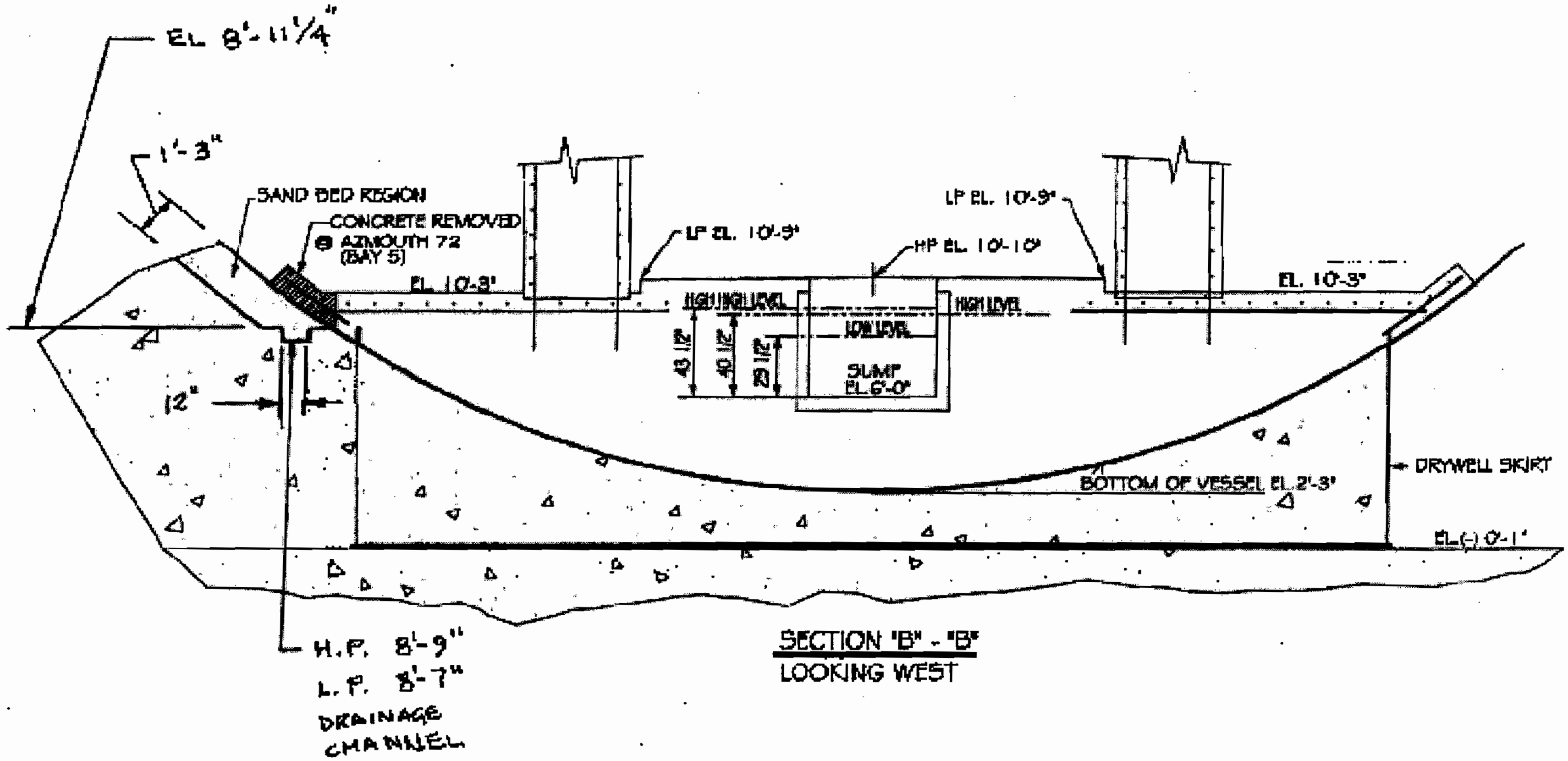
SECTIONAL VIEW OF SAND BED AREA
AT VENT PIPE
(TYPICAL TEN PLACES)

Trench Detail Sketch

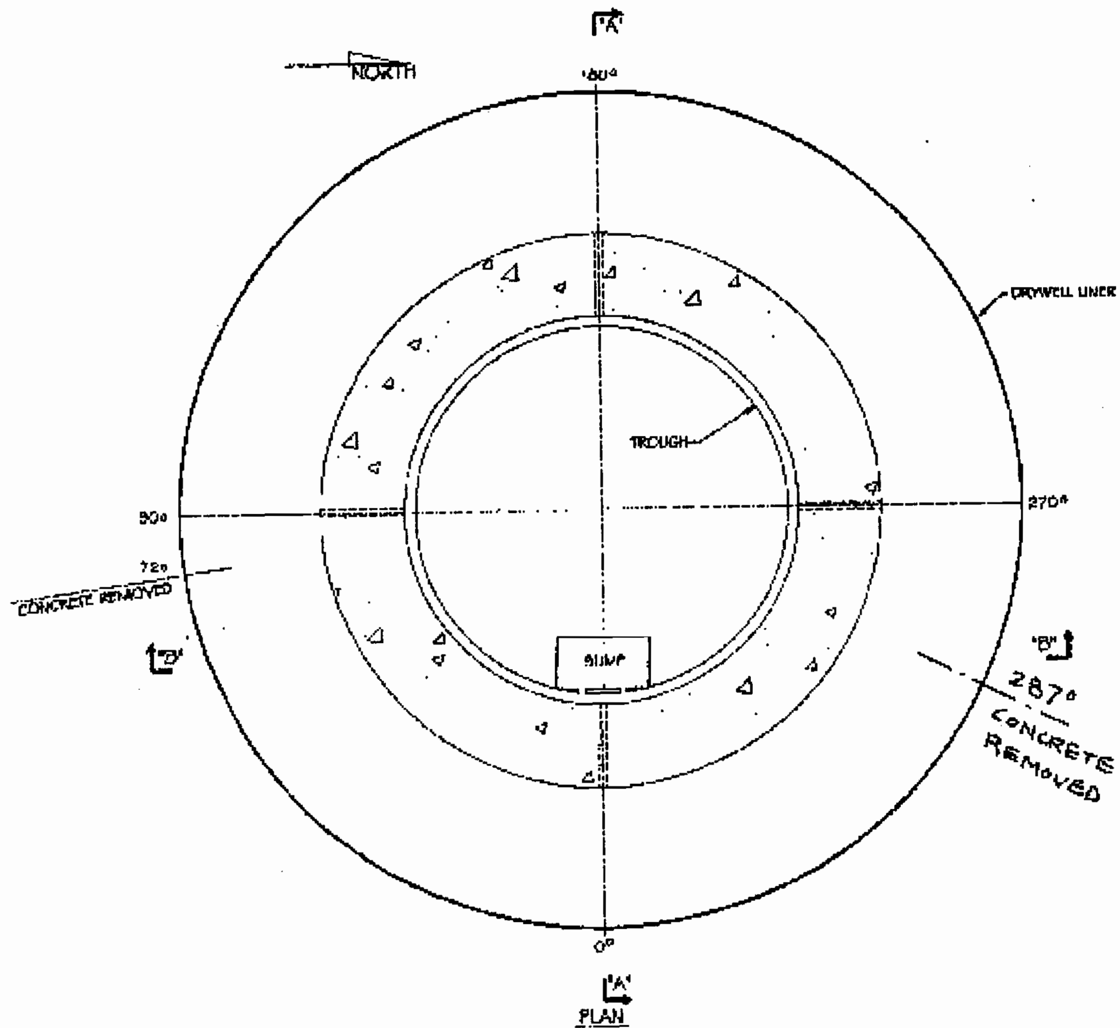


CROSS-SECTION AREA
OF TRENCHES IN BAYS 5 & 17

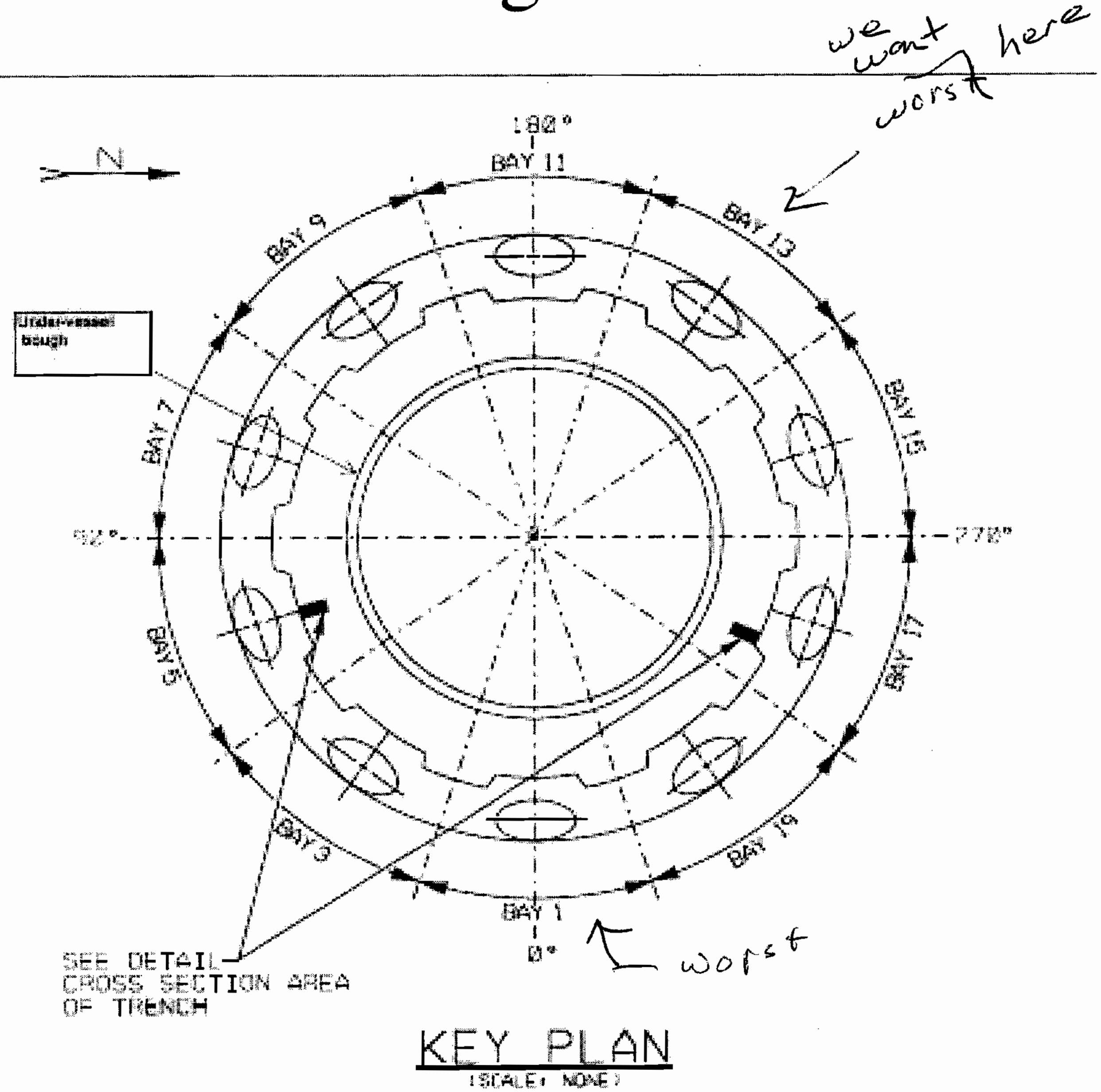
Sketch Showing Lower Drywell - Sandbed, Trench and Sump



Top View – Drywell Floor Sketch



Plan View of Trough and Trenches



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

August 23, 2006

MEMORANDUM TO: ACRS Plant License Renewal Subcommittee Members

FROM: Michael A. Junge, Senior Staff Engineer
Technical Support Branch, ACRS

SUBJECT: REVIEW MATERIALS FOR THE MEETING OF THE LICENSE RENEWAL SUBCOMMITTEE ON OCTOBER 3, 2006 RELATED TO THE INTERIM REVIEW OF THE LICENSE RENEWAL OF THE OYSTER CREEK GENERATING STATION

The purpose of this memorandum is to forward background materials related to the License Renewal Subcommittee Meeting on October 3, 2006 with staff of the Office of Nuclear Reactor Regulation and AmerGen Power Company representatives to discuss the interim review of the license renewal of Oyster Creek Generating Station..

To prepare for the meeting, the following documents are attached:

1) Draft Proposed Agenda
2) Status Report
3) Safety Evaluation Report with Open Items Related to the License Renewal of the Oyster Creek Generating Station, dated August 18, 2006. This is included on a CD.
4) Oyster Creek Generating Station- Application for Renewed Operating Licenses, dated July 22, 2005 This is included on a CD.
5) Supplemental Information Related to the Aging Management Program for the Oyster Creek Drywell Shell, Associated with AmerGen's License Renewal Application, dated June 20, 2006.
6) Audit and Review Report for Plant Aging Management Reviews and Programs- Oyster Creek Generating Station dated August 18, 2006.
7) Supplemental Response to NRC Request for Additional Information (RAI 2.5.1.19-1), dated September 28, 2005, Related to Oyster Creek Generating Station License Renewal Application, dated November 11, 2005.

For additional information, please contact me at (301) 415-6855 or MXJ2@NRC.GOV.

Attachments: As stated

cc: w/o Attachments: J. Larkins M. Snodderly S. Duraiswamy

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

Advisory Committee on Reactor Safeguards
Plant License Renewal Subcommittee Meeting
Oyster Creek Generating Station
October 3, 2006
Rockville, MD

-PROPOSED SCHEDULE-

Cognizant Staff Engineer: Michael A. Junge mxj2@NRC.GOV (301) 415-6855

Topics	Presenters	Time
Opening Remarks	O. Maynard, ACRS	1:30 pm - 1:35 pm
Staff Introduction	Louise Lund, NRR	1:35 pm - 1:40 pm
Oyster Creek License Renewal Application A. Application Background B. Description of Oyster Creek C. Operating History D. Scoping Discussion E. Application of GALL F. Commitment Process	Timothy Rausch, Michael Gallagher, Fred Polaski, and John Hufnagel, Amergen Energy Company	1:40 pm - 2:40 pm
Break		2:40 pm - 2:55 pm
SER Overview A. Scoping and Screening Results B. Onsite Inspection Results	Donnie Ashley, NRR Michael Modes, Region I	2:55 pm - 3:15 pm
Aging Management Program Review and Audits	Donnie Ashley, NRR	3:15 pm - 4:00 pm
Time-Limited Aging Analyses	Donnie Ashley, NRR	4:00 pm - 4:30 pm
Public Comment	Dennis Zannoni, State of NJ	4:30 pm - 5:00 pm
Subcommittee Discussion	O. Maynard, ACRS	5:00 pm - 5:30 pm

NOTE:

- Presentation time should not exceed 50 percent of the total time allocated for a specific item. The remaining 50 percent of the time is reserved for discussion.

- 50 copies of the presentation materials to be provided.

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

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS SUBCOMMITTEE ON PLANT LICENSE RENEWAL OYSTER CREEK GENERATING STATION OCTOBER 3, 2006 ROCKVILLE, MARYLAND

- STATUS REPORT -

PURPOSE

The purpose of this meeting is to review the License Renewal Application (LRA) for Oyster Creek Generating Station (OCGS), and the associated Draft Safety Evaluation Report (SER) with Open Items dated August 18, 2006. The Subcommittee will hear presentations by and hold discussions with representatives of the staff and AmerGen Energy Company.

BACKGROUND

The Oyster Creek Generating Station (OCGS) is a single unit facility. It is located in Lacey Township, Ocean County, New Jersey, approximately two miles south of the community of Forked River, about two miles inland from the shore of Barnegat Bay and seven miles west-north-west of Barnegat Light. The site, about 800 acres, is approximately nine miles south of Toms River, New Jersey, about fifty miles east of Philadelphia, Pennsylvania, and sixty miles south of Newark, New Jersey. The reactor is a single cycle, forced circulation boiling water reactor (BWR-2) with a Mark 1 type Containment. The reactor produces steam for direct use in the steam turbine. The primary containment is of the Mark 1 design that consists of a drywell, a suppression chamber in the shape of a torus and a connecting vent system between the drywell and the suppression chamber.

Initial criticality was achieved on May 3, 1969 and Oyster Creek Generating Station was placed in commercial operation on December 23, 1969 under a Provisional Operating License. On July 2, 1991, the NRC issued a Full Term Operating License (Facility Operating License No. DPR-16) which superseded the Provisional Operating License in its entirety. On August 8, 2000, Oyster Creek Generating Station was acquired by and the license transferred to AmerGen. The License permits steady-state reactor core power levels not in excess of 1930 megawatts (thermal) and is in effect until midnight on April 9, 2009.

DISCUSSION

By letter dated July 22, 2005(ADAMS Accession No. ML052080048), AmerGen submitted the License Renewal Application (LRA) for OCGS in accordance with Title 10, Part 54, of the *Code of Federal Regulations* (10 CFR Part 54).

AmerGen is requesting renewal of the operating licenses for OCGS, (Facility Operating License DPR-16) for a period of 20 years beyond the current expiration date of April 9, 2009. The staff of the U.S. Nuclear Regulatory Commission (NRC or the staff) reviewed the license renewal application (LRA) for Oyster Creek Generating Station in accordance with the NRC regulations and NUREG-1800, Revision 1, “Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants,” dated September 2005. Title 10, Section 54.29, of the *Code of Federal Regulations* (10 CFR 54.29) provides the standards for issuance of a renewed license.

The licensee stated that it had not identified any Technical Specification (TS) changes necessary to support issuance of the renewed operating license.

The staff used the following Interim Staff Guidance (ISG) in the Oyster Creek LRA review: Station Blackout (SBO) Scoping, Concrete Aging Management Program (AMP), Fire Protection (FP) System Piping, and Identification & Treatment of Electrical Fuse Holders.

The Draft SER with Open Items presents the status of the staff's review of information submitted through July 10, 2006. It contains five open items, no confirmatory items, 3 proposed license conditions, and 65 commitments.

<u>OPEN ITEMS</u>

1. In RAI 4.7.2-1 dated March 10, 2006, the staff requested that the applicant provide the following information: For the drywell corrosion during the late 1980s and the new corrosion found during the subsequent inspections, provide the process used to establish confidence that the sampling done to identify the areas of corrosion has been adequate.

2. In RAI 4.7.2-1 dated March 10, 2006, the staff requested that the applicant provide the following information: For the drywell corrosion during the late 1980s and the new corrosion found during the subsequent inspections, provide the process used to establish confidence that the sampling done to identify the areas of corrosion has been adequate.

3. In RAI 4.7.2-1 dated March 10, 2006, the staff requested that the applicant provide the following information: A summary of the factors considered in establishing the minimum required drywell thickness.

4. In RAI 4.7.2-1 dated March 10, 2006, the staff requested that the applicant provide the following information: A summary of the factors considered in establishing the minimum required drywell thickness.

5. In RAI 4.7.2-3 dated March 10, 2006, the staff noted that leakage from the refueling seal has been identified as one of the reasons for accumulation of water and contamination of the sand-pocket area. The refueling water passes through the gap between the shield concrete and the drywell shell in the long length of inaccessible areas. As there is a potential for corrosion, ASME Code Subsection IWE would require augmented inspection of this area. The staff requested that the applicant provide a summary of inspections (visual and NDE) and mitigating actions to prevent water leaks from the refueling seal components.

PROPOSED LICENSE CONDITIONS

1. The first license condition requires the applicant to include the UFSAR supplement required by 10 CFR 54.21(d) in the next UFSAR update, as required by 10 CFR 50.71(e), following the issuance of the renewed license.

2. The second license condition requires future activities identified in the UFSAR supplement to be completed prior to the period of extended operation.

3. The third license condition requires all surveillance capsules placed in storage to be maintained for future insertion. Any changes to storage requirements must be approved by the staff as required by 10 CFR Part 50, Appendix H.

COMMITMENTS

Commitments made by the licensee are listed in detail in Appendix A to the SER. The licensee made 65 commitments related to the AMPs to manage aging effects of structures and components prior to the periods of extended operation. The following are a summary:

1. ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD. Existing program is credited. For the isolation condensers this program also includes enhancement activities identified in NUREG-1801, "Generic Aging Lessons Learned (GALL) Report," lines IV.C1-5 and IV.C1-6. These enhancement activities consist of:
   (1) Temperature and radioactivity monitoring of the shell-side (cooling) water, which will be implemented prior to the period of extended operation.
   (2) Eddy current testing of the tubes, with inspection (VT or UT) of the tubesheet and channel head, which will be performed during the first ten years of the extended period of operation.
2. Water Chemistry existing program is credited.
3. Reactor Head Closure Studs existing program is credited.
4. BWR Vessel ID Attachment Welds existing program is credited.
5. BWR Feedwater Nozzle. The existing program is credited. The Oyster Creek Feedwater Nozzle Program will be enhanced
6. BWR Control Rod Drive Return Line Nozzle Existing program is credited.
7. BWR Stress Corrosion Cracking existing program is credited. The program will be enhanced.
8. BWR Penetrations existing program is credited.
9. BWR Vessel Internals Existing program is credited. The program will be enhanced.
10. Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS).
11. Flow-Accelerated Corrosion iexisting program is credited.

12. Bolting Integrity existing program is credited. Program site implementing documents will be enhanced.
13. Open-Cycle Cooling Water System Existing program is credited. The program will be enhanced.
14. Closed-Cycle Cooling Water System Existing program is credited.
15. Boraflex Monitoring Existing program is credited.
16. Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems existing program is credited. The scope of the program will be increased and enhanced.
17. Compressed Air Monitoring existing program is credited.
18. BWR Reactor Water Cleanup System Existing program is credited. Based on Generic Letter 89-10 containment isolation valve upgrades/enhancements, an effective Hydrogen Water Chemistry program, and the complete lack of cracking found during any of the RWCU piping weld inspections performed under Generic Letter 88-01, all inspection requirements for the portion of the RWCU System outboard of the second containment isolation valves have been eliminated.
19. Fire Protection existing program is credited. The program will be enhanced.
20. Fire Water System existing program is credited. The program will be enhanced.
21. Aboveground Outdoor Tanks is a new program..
22. Fuel Oil Chemistry will be enhanced.
23. Reactor Vessel Surveillance will be enhanced.
24. One-Time Inspection is a new program.
25. Selective Leaching of Materials is a new program.
26. Buried Piping Inspection existing program is credited. The program will be enhanced.
27. ASME Section XI, Subsection IWE existing program is credited. The program will be enhanced.
28. ASME Section XI, Subsection IWF existing program is credited. The scope of the program will be enhanced.
29. 10 CFR Part 50, Appendix J existing program is credited.
30. Masonry Wall Program existing program is credited.
31. Structures Monitoring Program existing program is credited.
32. RG 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants existing program is credited.
33. Protective Coating Monitoring and Maintenance Program existing program is credited.
34. Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements is a new program.
35. Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits existing program is credited. The program will be enhanced.
36. Inaccessible Medium Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements is a new program.
37. Periodic Testing of Containment Spray Nozzles existing program is credited.
38. Lubricating Oil Monitoring Activities existing plant specific program is credited.
39. Generator Stator Water Chemistry Activities existing program is credited.

40. Periodic Inspection of Ventilation Systems existing plant specific program is credited.
41. Periodic Inspection Program is a new program.
42. Wooden Utility Pole Program is a new program.
43. Periodic Monitoring of Combustion Turbine Power Plant - Electrical A new plant specific program is credited.
44. Metal Fatigue of Reactor Coolant Pressure Boundary existing program is credited.
45. Environmental Qualification (EQ) Program existing program is credited.
46. New P-T curves Revised pressure-temperature (P-T) limits for a 60-year licensed operating life have been prepared and will be submitted to the NRC for approval.
47. Circumferential Weld Exam Relief Apply for extension Reactor Vessel Circumferential Weld Examination Relief for 60-year operation.
48. Axial weld Exam Relief Apply for extension Reactor Vessel Axial Weld Examination Relief for 60-year operation.
49. Measure Drywell wall thickness Drywell wall thickness will be monitored to ensure minimum wall thickness is maintained. The ASME Section XI, Subsection IWE Program, will manage the aging effects.
50. Fluence Methodology The NRC has issued a SER for RAMA approving RAMA for reactor vessel fluence calculations. Oyster Creek will comply with the applicable requirements of the SER.
51. Bolting Integrity - FRCT. The Bolting Integrity - FRCT Program is a new program.
52. Closed-Cycle Cooling Water System - FRCT. The Closed-Cycle Cooling Water System – FRCT Program is a new program.
53. Aboveground Steel Tanks - FRCT. The Above ground Steel Tanks - FRCT Program is a new program.
54. Fuel Oil Chemistry – FRCT. The Fuel Oil Chemistry - FRCT Program is a new program.
55. One-Time Inspection - FRCT. The One-Time Inspection – FRCT program will provide measures to verify that an aging management program is not needed, confirms the effectiveness of existing activities, or determines that degradation is occurring which will require evaluation and corrective action. The program will be implemented prior to the period of extended operation.
56. Selective Leaching of Materials -FRCT. The Selective Leaching of Materials - FRCT Program is a new program.
57. Buried Piping Inspection – FRCT. The Buried Piping Inspection - FRCT Program is a new program.
58. Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components-FRCT. The Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components - FRCT Program is a new program.
59. Lubricating Oil Analysis Program – FRCT. The Lubricating Oil Analysis Program – FRCT is a new program.
60. Periodic Inspection Program - FRCT. The Periodic Inspection Program - FRCT is a new program.
61. Buried Piping and Tank Inspection – Met Tower Repeater Engine Fuel Supply. The Buried Piping and Tank Inspection – Met Tower Repeater Engine Fuel Supply Program is a new program.

62. AmerGen will commit to perform monitoring of any leakage from the spent fuel pool liner via the pool leak chase piping.
63. AmerGen will replace the previously un-replaced, buried safety-related ESW piping prior to the period of extended operation.
64. Electrical Cable Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements. The Electrical Cable Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program is a new program .
65. Corrective Action, Confirmation and Administrative Controls for Forked River Combustion Turbine activities. Prior to the period of extended operation, AmerGen will ensure that procedures are established to implement the program elements of Corrective Action, Confirmation, and Administrative Controls, as described in Sections A.0.5 and B.0.3 of Enclosure 1 of AmerGen letter 2130-06-20334, for the Forked River Combustion Turbine aging management activities.

SCOPING & SCREENING AND AUDIT OF AMPs & AMRs

The staff performed a scoping and screening methodology inspection, AMP inspection, and an audit of the AMPs and aging management reviews (AMRs).

The staff's scoping and screening methodology inspection has been completed, with an exit meeting scheduled September 13, 2006. The report will be issued shortly after the exit meeting.
The audit of the AMPs and AMRs is documented in a report by Brookhaven National Laboratory dated May 9, 2006. The audit examined 29 AMPs and the associated AMRs in the LRA. The project team reviewed 28 AMPs and associated AMRs that the licensee claimed were consistent with the GALL Report. The project team also reviewed one plant-specific AMP. The audit verified that the AMPs were consistent with GALL. The audit also concluded that the AMRs were consistent with the GALL Report.

TLAAs

Based on OCGS's current licensing basis, UFSAR, and design-basis documents, the following categories of Time Limited Aging Analyses (TLAAs) were considered:

- neutron embrittlement of reactor vessel and internals
- metal fatigue of the reactor vessel, internals, and reactor coolant pressure boundary (RCPB) piping and components
- environmental qualification (EQ) of electrical equipment
- loss of prestress in concrete containment tendon
- fatigue analysis of primary containment, attached piping, and components
- reactor building crane, turbine building crane, heater bay crane load cycles
- drywell corrosion
- equipment pool and reactor cavity walls rebar corrosion
- reactor vessel weld flaw evaluations
- control rod drive (CRD) stub tube flaw analysis

On the basis of its review, the staff concludes, subject to the resolution OIs 4.7.2-1.1, 4.7.2-1.2, 4.7.2-1.3, 4.7.2-1.4, and 4.7.2-3, that the applicant has provided an adequate list of TLAAs, as defined in 10 CFR 54.3. Further, the staff concludes that the applicant has demonstrated that (1) the TLAAs will remain valid for the period of extended operation, as required by 10 CFR 54.21(c)(1)(I), (2) the TLAAs have been projected to the end of the period of extended operation, as required by 10 CFR 54.21(c)(1)(ii), or (3) that the aging effects will be adequately managed for the period of extended operation, as required by 10 CFR 54.21(c)(1)(iii). The staff also reviewed the UFSAR supplement for the TLAAs and found that the supplement contains descriptions of the TLAAs sufficient to satisfy the requirements of 10 CFR 54.21(d). In addition, consistent with 10 CFR 54.21©)(2), the staff concludes that no plant-specific, TLAA-based exemptions are in effect.

EXPECTED SUBCOMMITTEE ACTION

The Subcommittee Chairman will provide a report to the Full Committee during the February 2006 ACRS meeting.

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

September 22, 2006

MEMORANDUM TO: ACRS Plant License Renewal Subcommittee Members

FROM: Michael A. Junge, Senior Staff Engineer
Technical Support Branch, ACRS

SUBJECT: REVIEW MATERIALS FOR THE MEETING OF THE LICENSE RENEWAL SUBCOMMITTEE ON OCTOBER 3, 2006 RELATED TO THE INTERIM REVIEW OF THE LICENSE RENEWAL OF THE OYSTER CREEK GENERATING STATION

The purpose of this memorandum is to forward background materials related to the License Renewal Subcommittee Meeting on October 3, 2006 with staff of the Office of Nuclear Reactor Regulation and AmerGen Power Company representatives to discuss the interim review of the license renewal of Oyster Creek Generating Station..

To prepare for the meeting, the following documents are attached:

1) Oyster Creek Generating Station - NRC License Renewal Inspection Report 05000219/2006007, dated September 21, 2006

Documents previously sent:

1) Draft Proposed Agenda
2) Status Report
3) Safety Evaluation Report with Open Items Related to the License Renewal of the Oyster Creek Generating Station, dated August 18, 2006. This is included on a CD.
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For additional information, please contact me at (301) 415-6855 or MXJ2@NRC.GOV.

Attachments: As stated

cc: w/o Attachments: J. Larkins M. Snodderly S. Duraiswamy

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

Advisory Committee on Reactor Safeguards
Plant License Renewal Subcommittee Meeting
Oyster Creek Generating Station

October 3, 2006
Rockville, MD

-PROPOSED SCHEDULE-

Cognizant Staff Engineer: Michael A. Junge mxj2@NRC.GOV (301) 415-6855

Topics	Presenters	Time
Opening Remarks	O. Maynard, ACRS	1:30 pm - 1:35 pm
Staff Introduction	Louise Lund, NRR	1:35 pm - 1:40 pm
Oyster Creek License Renewal Application A. Application Background B. Description of Oyster Creek C. Operating History D. Scoping Discussion E. Application of GALL F. Commitment Process	Timothy Rausch, Michael Gallagher, Fred Polaski, and John Hufnagel, Amergen Energy Company	1:40 pm - 2:40 pm
Break		2:40 pm - 2:55 pm
SER Overview A. Scoping and Screening Results B. Onsite Inspection Results	Donnie Ashley, NRR Michael Modes, Region I	2:55 pm - 3:15 pm
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Public Comment	Dennis Zannoni, State of NJ	4:30 pm - 5:00 pm
Public Comment	Paul Gunter, NIRS Richard Webster, Rutger Environmental Law Clinic	5:00 pm-5:30 pm
Subcommittee Discussion	O. Maynard, ACRS	5:30 pm - 6:00 pm

NOTE:

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- 50 copies of the presentation materials to be provided.

September 21, 2006

Mr. Christopher M. Crane
President and CEO
AmerGen Energy Company, LLC
200 Exelon Way, KSA 3-E
Kennett Square, PA 19348

SUBJECT: OYSTER CREEK GENERATING STATION - NRC LICENSE RENEWAL INSPECTION REPORT 05000219/2006007

Dear Mr. Crane:

On March 31, 2006, the NRC completed the onsite portion of the inspection of your application for license renewal of your Oyster Creek Generating Station. The inspection continued in our Region I office until early September 2006. The enclosed report documents the results of the inspection, which were discussed on September 13, 2006, with members of your staff in an exit meeting open for public observation at the Lacey Township Town Hall.

The purpose of this inspection was to examine the plant activities and documents that supported the application for a renewed license of Oyster Creek Generating Station. The inspection reviewed the screening and scoping of non-safety related systems, structures, and components, as required in 10 CFR 54.4(a)(2), and determined whether the proposed aging management programs are capable of reasonably managing the effects of aging. These NRC inspection activities constitute one of several inputs into the NRC review process for license renewal applications.

The inspection team concluded screening and scoping of non-safety related systems, structures, and components, was implemented as required in 10 CFR 54.4(a)(2), and the aging management portion of the license renewal activities were conducted as described in the License Renewal Application. The inspection results support a conclusion that the proposed activities will reasonably manage the effects of aging in the systems, structures, and components identified in your application. The inspection concluded the documentation supporting the application was in an auditable and retrievable form.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Website at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

Donald E. Jackson, Acting Chief
Engineering Branch 1
Division of Reactor Safety

Docket No. 50-219
License No. DPR-16

Enclosure: Inspection Report 05000219/2006007

cc w/encl:
Chief Operating Officer, AmerGen
Site Vice President, Oyster Creek Nuclear Generating Station, AmerGen
Plant Manager, Oyster Creek Generating Station, AmerGen
Regulatory Assurance Manager, Oyster Creek, AmerGen
Senior Vice President - Nuclear Services, AmerGen
Vice President - Mid-Atlantic Operations, AmerGen
Vice President - Operations Support, AmerGen
Vice President - Licensing and Regulatory Affairs, AmerGen
Director Licensing, AmerGen
Manager Licensing - Oyster Creek, AmerGen
Vice President, General Counsel and Secretary, AmerGen
T. O'Neill, Associate General Counsel, Exelon Generation Company
J. Fewell, Assistant General Counsel, Exelon Nuclear
Correspondence Control Desk, AmerGen
J. Matthews, Esquire, Morgan, Lewis & Bockius LLP
Mayor of Lacey Township
K. Tosch, Chief, Bureau of Nuclear Engineering, NJ Dept of Environmental Protection
R. Shadis, New England Coalition Staff
N. Cohen, Coordinator - Unplug Salem Campaign
W. Costanzo, Technical Advisor - Jersey Shore Nuclear Watch
E. Gbur, Chairwoman - Jersey Shore Nuclear Watch
E. Zobian, Coordinator - Jersey Shore Anti Nuclear Alliance
P. Baldauf, Assistant Director, Radiation Protection and Release Prevention, State of New Jersey
R. Webster, Rutgers Environmental Law Clinic

Distribution w/encl: **(VIA E-MAIL)**
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U. S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No:	50-219
License No:	DPR-16
Report No:	05000219/20006007
Licensee:	AmerGen Energy Company, LLC
Facility:	Oyster Creek Generating Station
Location:	Forked River, New Jersey
Dates:	March 13 - 17, 2006 and March 27 - 31, 2006
Inspectors:	M. Modes, Team Leader, Division of Reactor Safety (DRS) P. Kaufman, Sr. Reactor Inspector, DRS G. Meyer, Sr. Reactor Inspector, DRS S. Chaudhary, Health Physicist, Division of Nuclear Materials Safety (DNMS) T. O'Hara, Reactor Inspector, DRS J. Lilliendahl, Reactor Inspector, DRS D. Johnson, Reactor Inspector, DRS D. Werkheiser, Resident Inspector, Division of Reactor Projects (DRP)
Approved By:	Donald E. Jackson, Acting Chief Engineering Branch 1 Division of Reactor Safety

Enclosure

SUMMARY OF FINDINGS

IR 05000219/2006007; 03/13/2006 - 03/17/2006, 03/27/2006 - 03/31/2006, Oyster Creek Generating Station; Inspection of the Scoping of Non-Safety Systems and the Proposed Aging Management Procedures for the Oyster Creek Generating Station Application for Renewed License.

This inspection of license renewal activities was performed by eight regional office engineering inspectors. The inspection was conducted in accordance with NRC Manual Chapter 2516 and NRC Inspection Procedure 71002. This inspection did not identify any "findings" as defined in NRC Manual Chapter 0612. The inspection team concluded screening and scoping of non-safety related systems, structures, and components, were implemented as required in 10 CFR 54.4(a)(2), and the aging management portions of the license renewal activities were conducted as described in the License Renewal Application. The inspection results support a conclusion that the proposed activities will reasonably manage the effects of aging in the systems, structures, and components identified in your application. The inspection concluded the documentation supporting the application was in an auditable and retrievable form.

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Report Details

4. OTHER ACTIVITIES (OA)

4OA2 Other - License Renewal

a. Inspection Scope

This inspection was conducted by NRC Region I and headquarters based inspectors in order to evaluate the thoroughness and accuracy of the screening and scoping of non-safety related systems, structures, and components, as required in 10 CFR 54.4(a)(2) and to evaluate whether aging management programs will be capable of managing the identified aging effect in a appropriate manner.

The inspection team selected a number of systems for review, using the NRC accepted guidance, in order to determine if the methodology applied by the applicant appropriately captured the non-safety systems affecting the safety functions of a system, component, or structure within the scope of license renewal.

The inspection team selected a sample of aging management programs to verify the adequacy of the applicant's documentation and implementation activities. The selected aging management programs were reviewed to determine whether the proposed aging management implementing process would adequately manage the effects of aging on the system.

The inspectors reviewed supporting documentation and interviewed applicant personnel to confirm the accuracy of the license renewal application conclusions. For a sample of plant systems and structures, inspectors performed visual examinations of accessible portions of the systems to observe aging effects.

a.1. Scoping of Non Safety-Related Systems, Structures, and Components

To assess the thoroughness and accuracy of the methods used to bring systems, structures, and components within scope of the application and to screen non-safety related systems, structures, and components, as required in 10 CFR 54.4(a)(2), the inspectors reviewed the applicant's program guidance procedures and summaries of results for Oyster Creek. The inspectors determined the applicant's procedures to be consistent with the NRC accepted guidance in Sections 3, 4, and 5 of Appendix F to NEI 95-10, Revision 5 (3: non-safety related systems, structures, and components within scope of the current licensing basis, 4: non-safety related systems, structures, and components directly connected to safety-related systems, structures, and components, and 5: non-safety related systems, structures, and components not directly connected to safety-related systems, structures, and components). Also, the inspectors determined that the applicant appropriately utilized the guidance in their process for determining which systems were within scope.

The applicant based the scoping and screening results on a technical review and walkdown of all applicable plant areas by qualified plant personnel. The inspectors reviewed the set of license renewal drawings, which were color-coded based on the results. The inspectors interviewed personnel and independently inspected numerous areas within the plant to confirm that appropriate systems, structures, and components had been included within the license renewal scope, that systems, structures, and components excluded from the license renewal scope had an acceptable basis, and that the boundary for determining scope within the systems, including anchors, was appropriate. For systems, structures, and components selected from the results, the inspectors confirmed that the in-plant configuration was accurate and acceptably categorized, and for systems, structures, and components selected within the plant, the inspectors confirmed that the categorization result in program documents was appropriate. The in-plant areas and systems reviewed included the following:

- Reactor Building;
- Turbine Building;
- Intake Structure;
- Ventilation Stack;
- Diesel Generator Building;
- Diesel Fuel Oil Building;
- Fire Protection System;
- Isolation Condenser System;
- Hardened Vent System;
- Nitrogen Supply System;
- Instrument Air System; and
- Service Water System.

The inspectors determined the personnel involved in the process were knowledgeable and appropriately trained, and that the applicant had implemented an acceptable method of scoping and screening of non-safety related systems, structures, and components.

a.2. <u>Programs</u>

<u>One-Time Inspection Program</u>

The One-Time Inspection Program is a new aging management program intended to verify the effectiveness of other aging management programs, including Water Chemistry, Closed Cycle Cooling Water Systems, and Fuel Oil Chemistry Programs, by reviewing various aging effects for impact. Where corrosion resistant materials and/or non-corrosive environments exist, the One-Time Inspection Program is intended to verify that an aging management program is not needed during extended operations by confirming that aging effects are not occurring or are occurring in a manner that does not affect the safety function of systems, structures, and components within the scope of the application. Non-destructive evaluation will be performed by qualified personnel using procedures and processes consistent with the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME) and 10 CFR 50, Appendix B. The

One-Time Inspection Program will be implemented prior to the period of extended operation.

The inspectors reviewed the program description, implementation plan, and inspection sample basis, and discussed the planned activities with the responsible staff.

For the One-Time Inspection Program, the inspectors concluded the applicant performed adequate evaluations and reviews of industry experience and plant history to determine an acceptable approach to identifying, assessing and managing any aging effects detected. The applicant developed adequate guidance for implementation of the One-Time Inspection Program.

Bolting Integrity

The Bolting Integrity Program is an existing program credited with managing the loss of material, cracking, and loss of prestress aging effects in safety-related bolting at Oyster Creek. The aging effects are managed by visual inspection for leakage during system pressure tests, normal plant operation, and periodic system maintenance, and repaired in accordance with maintenance procedures and the ASME Code.

The inspectors reviewed the program basis document, implementing procedures, documented reviews, and a bolting-related apparent cause evaluation, and interviewed the responsible plant personnel regarding these documents. In addition, the inspectors walked down portions of the Standby Liquid Control, Isolation Condenser, Control Rod Drive, and Reactor Building Closed Cooling Water Systems to confirm that the program had maintained acceptable bolting conditions.

For the Bolting Integrity Program, the inspectors concluded that the applicant had performed adequate evaluations as well as industry experience and historical reviews to determine the aging effects are managed by the Bolting Integrity Program. The applicant provided adequate guidance to ensure the aging effects are appropriately managed.

Buried Piping Inspection

The Buried Piping Inspection Program is an existing program credited with managing the loss of material aging effects on the external surfaces of piping in a soil environment, including the service water, emergency service water, and condensate transfer systems. The aging effects are managed by preventive measures, i.e., coatings, wrapping, and condition monitoring measures, including visual inspections and periodic system pressure testing. As described in Appendix B, Part 1.26 of the application, the applicant plans to enhance the program by augmenting the visual inspections prior to extended operations and performing periodic visual inspections, and to include additional piping, such as the fire protection system.

The inspectors reviewed the program basis document, system drawings, implementing procedures, and documented reviews, and interviewed the responsible plant personnel regarding these documents. Also, the inspectors walked down the service water and emergency service water systems in the vicinity of buried piping.

For the Buried Piping Inspection Program, the inspectors concluded that the applicant had performed adequate evaluations as well as industry experience and historical reviews to determine the aging effects managed by the Buried Piping Inspection Program. The applicant provided adequate guidance to ensure the aging effects are appropriately managed.

<u>Flow-Accelerated Corrosion Program</u>

The Flow-Accelerated Corrosion Program is an existing program credited with managing the corrosion aging effects in all carbon steel piping and components containing high-energy fluids at Oyster Creek Generating Station. The aging effects are managed by using ultrasonic and radiographic testing to detect wall thinning and by predicting wear rates to support the proactive replacement of system piping. In addition, the program provides for the performance of follow-up inspections to confirm predictions and to determine the need for repairs or replacements as necessary.

The inspectors reviewed the piping ultrasonic testing wall thickness results from previous inspections and reviewed the CHECWORKS® computer analysis of the future wall thickness forecasts. The inspector also reviewed recent changes to the CHECWORKS® model to ensure previously identified deficiencies have been corrected. The inspectors noted that recent replacements, initiated as a result of this program, were implemented preventively due to identified flow-accelerated corrosion. The replacement piping material was more resistant to corrosion than the original piping material.

For the Flow-Accelerated Corrosion Program, the inspectors concluded the applicant conducted adequate evaluations as well as industry experience and historical reviews and, as a consequence, the effects of aging will be reasonably managed by the proposed program.

<u>Water Chemistry Program</u>

The Water Chemistry Program is an existing program credited with managing the effects of aging on piping, piping components, piping elements, and systems, such as the condensate and feedwater, and condensate storage tank in Oyster Creek Generating Station. The aging effects are managed by monitoring and control of reactor water chemistry to minimize contaminant concentration and mitigate loss of material due to general, crevice and pitting corrosion and cracking caused by stress corrosion cracking.

Water chemistry control is administered in accordance with the Boiling Water Reactor Vessel and Internals Project guideline BWRVIP-29 and Electric Power Research Institute guideline EPRI TR-103515. The inspectors reviewed the chemistry procedures and sampling results to confirm that the guidance contained in BWRVIP-29 and EPRI TR-103515 was being implemented.

For the Water Chemistry Program, the inspectors concluded the applicant had conducted adequate evaluations as well as industry experience and historical reviews to determine aging effects managed by an aging management program. The applicant provided adequate guidance to ensure aging effects are appropriately managed.

Closed-Cycle Cooling Water Systems Program

The Closed-Cycle Cooling Water Systems Program is an existing program credited with managing loss of material, cracking, and buildup-of-deposit aging effects in components exposed to closed-cycle cooling water environments at the Oyster Creek Generating Station. Systems within the scope of the closed-cycle cooling water program include the turbine building closed cooling, reactor building closed cooling, and emergency diesel generator closed cooling water systems. The aging effects are managed by monitoring and control of cooling water chemistry, performing surveillance tests, and through periodic inspection of system components in a manner consistent with EPRI TR-107396 guidelines.

The inspectors observed cleaning of the turbine closed-cooling water heat exchanger and performed a walkdown of the system with plant personnel. In addition, the inspectors reviewed closed-cycle cooling water chemistry procedures and reviewed past chemistry sample results to confirm that the requirements of EPRI TR-107396 are being met.

For the Closed-Cycle Cooling Water Systems Program, the inspectors concluded the applicant had conducted adequate evaluations as well as industry experience and historical reviews to determine aging effects managed by an aging management program. The applicant provided adequate guidance to ensure aging effects are appropriately managed.

10 CFR Part 50, Appendix J Program

The 10 CFR Part 50, Appendix J Program is an existing program credited with managing the aging degradation of pressure retaining boundaries of piping and components of the various systems penetrating the containment at the Oyster Creek Generating Station. In addition, the program also detects age related degradation in material properties of gaskets, o-rings, and packing materials for the primary containment pressure boundary access points. The aging effects are managed by performing containment leak rate tests to assure that leakage through primary containment and systems and components penetrating primary containment does not exceed allowable leakage limits specified in the Technical Specifications.

The inspectors reviewed Oyster Creek's procedures for leak rate testing. In addition, the inspectors reviewed corrective actions for components that did not meet leak rate test acceptance criteria. The inspectors noted that corrective actions taken to repair these components were acceptable.

For the 10 CFR Part 50, Appendix J Program, the inspectors concluded the applicant had conducted adequate evaluations as well as industry experience and historical reviews to determine aging effects managed by an aging management program. The applicant provided adequate guidance to ensure aging effects are appropriately managed.

Fuel Oil Chemistry Program

The Fuel Oil Chemistry Program is an existing program that will be modified for the purpose of managing the affects of pitting and corrosion in the diesel fuel oil tank at the Oyster Creek Generating Station. The aging effects are managed by the addition of biocides and corrosion inhibitors to minimize biological activity and mitigate corrosion, periodic cleaning, and applying coating to the internal surfaces of the tank.

The inspectors reviewed the schedule for implementation of the enhancements. The inspectors reviewed recent sample results and tank thickness measurements to verify that results were within the acceptable range.

For the Fuel Oil Chemistry Program, the inspectors concluded the applicant had conducted adequate evaluations as well as industry experience and historical reviews to determine aging effects managed by an aging management program. The applicant provided adequate guidance to ensure aging effects are appropriately managed.

Boiling Water Reactor Feedwater Nozzle Program

The Boiling Water Reactor Feedwater Nozzle Program is an existing program that will be modified to implement the recommendations of the Boiling Water Owners Group Licensing Topical Report: General Electric NE-523-A71-0594. These enhancements will be implemented prior to entering the period of extended operation per Oyster Creek Assignment Report AR# 00330592, A.1.05 Commitment (BWR Feedwater Nozzle). The program is credited with managing the aging effects of cracking in the feedwater nozzles. The program is administered by the station in-service inspection plan ER-OC-330-1001, "ISI Program Plan Fourth Ten-Year Inspection Interval," and implemented by station procedure ER-AA-330-002, "In-service Inspection of Section XI Welds and Components". The station in-service inspection program incorporates the requirements of the ASME Code. The aging effects are managed by periodic ultrasonic testing inspections of critical regions of the feedwater nozzles. The ultrasonic test inspections are performed at intervals not exceeding ten years and was embraced in an NRC safety evaluation.

Inspections performed in 1977 identified cracks in the Oyster Creek nozzles. These cracks were repaired. The inspectors reviewed plant modification #166-76-4, "Feedwater Nozzle Cladding Removal and Sparger Replacement" and reviewed selected ultrasonic testing examination reports of the feedwater nozzles. To minimize thermal cycling and fatigue induced cracking, the thermal sleeves were modified to a piston design. Subsequent inspections found no indications in the feedwater nozzles. The inspectors reviewed Focused Area Self-Assessment Report Oyster Creek Inservice Inspection Program, completed in June 2004. The inspectors determined the feedwater nozzle program at Oyster Creek effectively monitored the feedwater nozzles for cracking.

For the Boiling Water Reactor Feedwater Nozzle Program, the inspectors concluded the applicant had conducted adequate evaluations as well as industry experience and historical reviews to determine aging effects managed by an aging management program. The applicant provided adequate guidance to ensure aging effects are appropriately managed.

Boiling Water Reactor Stress Corrosion Cracking Program

The Boiling Water Reactor Stress Corrosion Cracking Program is an existing aging management program credited with managing crack initiation and growth due to intergranular stress corrosion cracking in stainless steel and nickel alloy reactor coolant pressure boundary piping, welds, components and piping four inches and larger nominal pipe size exposed to reactor coolant above 200°F. The aging effects are managed by preventive measures which include monitoring and controlling water impurities by improved water chemistry control activities and by providing replacement stainless steel components in a solution annealed condition with a maximum carbon content of 0.035% wt. and a minimum ferrite level of 7.5%. Inspection and flaw evaluations are conducted in accordance with Oyster Creek in-service inspection program plan ER-OC-330-1001, "ISI Program Plan Fourth Ten-Year Inspection Interval" and Oyster Creek's augmented inspection program for IGSCC ER-OC-330-1002, "IGSCC Inspection Plan Fourth Ten-Year Inspection Interval," which incorporates the technical basis and guidance described in NUREG-0313, NRC Generic Letter 88-01, and staff-reviewed Boiling Water Reactor Vessel Internal Inspection Program BWRVIP-75.

The inspectors noted that where pre-emptive piping replacement was accomplished the replacement piping material used was more resistant to intergranular stress corrosion cracking than the original piping material. The applicant replaced the following system piping material with intergranular stress corrosion cracking resistant material:

1) all isolation condenser large bore piping outside the drywell from the drywell penetrations to the isolation condensers during refueling outage 1R13 in 1991;

2) all piping within the four isolation condenser drywell penetrations and the two reactor water cleanup system drywell penetrations which contained welds that were not inspectible;

3) the head cooling spray nozzle assembly, the 4 inch tee and flange of the reactor vent line.

To further mitigate the initiation and propagation of intergranular stress corrosion cracking the applicant implemented hydrogen water chemistry during cycle 12 in 1990 and noble metals chemical additions during 1R19 refueling outage in 2002. Additionally, all accessible welds susceptible to intergranular stress corrosion cracking in reactor coolant boundary piping systems inside the drywell (except the reactor water cleanup system) were stress improved.

The Boiling Water Reactor stress corrosion cracking aging management program uses ultrasonic testing to detect intergranular stress corrosion cracking flaws in the reactor coolant boundary piping prior to loss of intended functions of the components. Of the 380 welds included in the scope of Generic Letter 88-01, Oyster Creek identified, during the period the program was implemented, there were 11 welds with indications of intergranular stress corrosion cracking. Nine welds have been repaired with full structural overlays (four in the core spray system, four in the reactor recirculation system, and one in the shutdown cooling system). Two reactor recirculation system welds, which were both stress improved before initial inspections had indications of intergranular stress corrosion cracking, remained in service without repair. Both of these welds in the reactor recirculation system have been re-examined in 2002 and 2004 using the Improved Performance Demonstration Initiative ultrasonic test examination technique and the welds did not exhibit any indication of intergranular stress corrosion cracking. No new indications of intergranular stress corrosion cracking have been detected by inspections during the past six refueling outages. As a result of the implemented preventive measures to mitigate intergranular stress corrosion cracking Oyster Creek has no indications of intergranular stress corrosion cracking at this time. Therefore the inspectors determined that the Boiling Water Reactor Stress Corrosion Program at Oyster Creek has been effective in monitoring and mitigating intergranular stress corrosion cracking in the reactor coolant boundary piping systems.

For the Boiling Water Reactor Stress Corrosion Cracking Program, the inspectors concluded the applicant had conducted adequate evaluations as well as industry experience and historical reviews to determine aging effects managed by an aging management program. The applicant provided adequate guidance to ensure aging effects are appropriately managed.

Periodic Inspection Program

The Periodic Inspection Program is a new program under development at Oyster Creek that consists of periodic inspections of selected systems in the scope of license renewal that require periodic monitoring of aging effects, and are not covered by other existing periodic monitoring programs to verify the integrity of the systems and confirm the absence of identified aging effects. The Periodic Inspection Program manages the aging effect of change in material properties, loss of material and reduction of heat transfer for systems, components, and environments. The aging effects are managed by periodic condition monitoring examinations performed at susceptible locations in the systems, intended to assure that existing environmental conditions are not causing material degradation that could result in a loss of system intended functions. The initial periodic inspections of this new aging management program will be implemented near the end of the current operating term but prior to the period of extended operation. Subsequent periodic inspections will be performed on a frequency not to exceed once every ten years.

The Periodic Inspection Program provides inspection criteria, requires evaluation of the inspection results, and provides recommendations for additional inspections, as necessary. Inspections will be performed in accordance with station procedures that are based on applicable codes and standards. Inspection methods may include visual examinations VT-1 or VT-3 of disassembled components or volumetric non-destructive examination techniques. Some of the implementing procedures for the Periodic Inspection Program were reviewed by the inspectors, including existing nondestructive examination procedures ER-OC-330-1001, ISI Program Plan "Fourth Ten-Year Inspection Interval," ER-AA-35-014, "VT-1 Visual Examinations," ER-AA-335-016, "VT-3 Visual Examination of Component Supports and Attachments," and ER-AA-335-032, "Ultrasonic Through Wall Sizing in Pipe Welds". A periodic inspection table, which was in draft at the time of this inspection, is a listing of selected systems and components to be periodically inspected to verify the integrity of the system and confirm the absence of identified aging effects was also reviewed. Based on review of the implementing documents and procedures, the inspectors determined that the Periodic Inspection Program, when implemented at Oyster Creek, will provide assurance that systems and components are routinely inspected for age related degradation of change in material properties, loss of material and reduction of heat transfer for systems, components, and environments, and will adequately manage the identified aging effects.

For the Periodic Inspection Program, the inspectors concluded the applicant had conducted adequate evaluations as well as industry experience and historical reviews to determine aging effects managed by an aging management program. The applicant provided adequate guidance to ensure aging effects are appropriately managed.

Wooden Utility Pole Program

The Wooden Utility Pole Program is a new program credited with managing the aging effects of loss of material and change in material properties in all wooden utility poles which support an intended function for the offsite power systems at the Oyster Creek Generating Station. The aging effects are managed by inspection of wooden poles every ten years by a qualified inspector.

The team reviewed program bases documents and industry guidance. The inspectors also conducted interviews and performed walkdowns with plant personnel. During the walkdown, one pole (JC 514A L) was noted to be degraded. The applicant was able to show that the condition had been previously analyzed and that plans are in place to adequately reinforce the pole.

For the Wooden Utility Pole Program, the inspectors concluded the applicant had conducted adequate evaluations as well as industry experience and historical reviews to determine aging effects managed by the Wooden Utility Pole Program. The applicant provided adequate draft guidance to ensure aging effects will be appropriately managed.

Periodic Testing of Containment Spray Nozzles

The Containment Spray Nozzle Program is an existing program credited with demonstrating that the drywell and torus spray nozzles are not blocked by debris or corrosion products. Carbon steel piping upstream of the drywell and torus spray nozzles is subject to possible general corrosion that could result in plugging nozzles with rust. Periodic air tests verify that the drywell and torus spray nozzles are free from plugging and are therefore available to provide the steam quenching functions of the nozzles.

The team conducted interviews and reviewed program bases documents and previous test results. The team noted that the existing program has been effective at identifying and correcting degraded conditions.

For the Periodic Testing of Containment Spray Nozzles Program, the inspectors concluded the applicant had conducted adequate evaluations as well as industry experience and historical reviews to determine aging effects managed by the Containment Spray Nozzle Program. The applicant provided adequate guidance to ensure aging effects are appropriately managed.

Inaccessible Medium-Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements

The Inaccessible Medium-Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program is a new program developed for the purpose of aging management credited with managing the moisture related aging effects in medium-voltage cable systems at the Oyster Creek Generating Station. The aging effects are managed by cable testing and periodic inspection of manholes.

The team reviewed program bases documents and industry guidance. The inspectors also conducted interviews and performed walkdowns with plant personnel. The manhole inspection frequency was initially established at the NUREG-1801, Revision 1,"Generic Aging Lessons Learned (GALL) Report," recommended two-year frequency based on Oyster Creek's operating experience that does not indicate a trend or recurrence of cable submergence in manholes. However, NRC inspectors identified, approximately 2 inches of water in the manhole selected by the NRC team for inspection. Consequently, the applicant entered this issue into the corrective action system (CA #IR 469998, #IR 471363) and documented the need to re-evaluate the adequacy of the manhole inspection frequency.

Due to several medium-voltage cable failures in Oyster Creek's operating experience, a medium-voltage cable testing program is currently in place. Because of the limited success of the previous DC step voltage testing method, Oyster Creek has begun implementing a new method of cable testing provided by DTE Energy for most of the medium-voltage cables. NUREG 1801 (XI.E3) specifies the test method should be state-of-the-art at the time the test is performed. Although the new DTE Energy testing method is not yet recognized as an industry standard, it is a form of partial discharge testing (partial discharge testing is one of the recognized standards specifically listed in the NUREG-1801), and the applicant expects formal acceptance of the new testing method as an industry standard prior to extended operation.

The applicant has agreed to maintain the current testing frequency limit of six years in LRCR 289 for the first six years, after which the frequency may be re-evaluated and extended up to ten years. This change will provide sufficient time for successful operating experience prior to expanding to the NUREG 1801 recommended ten year frequency.

For the Inaccessible Medium-Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program, the inspectors concluded the applicant had conducted adequate evaluations as well as industry experience and historical reviews to determine aging effects managed by the Inaccessible Medium-Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program. The applicant provided adequate guidance to ensure aging effects are appropriately managed.

<u>Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements</u>

The Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program is a new program credited with managing the heat, radiation, and moisture aging effects in non-environmentally qualified electrical cables and connections in Oyster Creek Generating Station. Connections include splices, terminal blocks, connectors, and fuse blocks. The aging effects are managed by periodic inspections.

The team reviewed program bases documents, a draft implementing procedure and industry guidance. The inspectors also conducted interviews and performed walkdowns with plant personnel. Inspections will be done of all accessible cables and connections in adverse localized environments. This aging management program focuses on a representative sample of accessible cables and connections with sampling structured to include key areas of concern. Plant locations containing cables within scope that do not include adverse general or localized conditions may be excluded from inspections based on engineering evaluations.

Because there were several examples of polyvinyl chloride cable insulation bleeding in Oyster Creek's operating experience, the applicant agreed to specifically include polyvinyl chloride cable insulation bleeding as an aging effect to be addressed in this program. Although polyvinyl chloride cable insulation bleeding has not led to any equipment degradation at Oyster Creek, there have been instances cited in NRC's Information Notices 91-20 and 94-78 where polyvinyl chloride insulation bleeding under unfavorable configurations caused hardened plasticizer to degrade equipment. As a consequence of the NRC's review, the applicant entered this issue into their corrective action system (AR 00472707) in order to evaluate the current extent-of-condition of polyvinyl chloride cable insulation bleeding and determine if their original screening of this aging affect should be revised.

For the Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program, the inspectors concluded the applicant had conducted adequate evaluations as well as industry experience and historical reviews to determine aging effects managed by the Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program. The applicant provided adequate draft guidance to ensure aging effects are appropriately managed.

Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrument Circuits

The Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrument Circuits Program is an existing program modified for the purpose of aging management that manages aging of the cables of the Intermediate Range Monitoring, Local Power Range Monitoring/Average Power Range Monitoring, Reactor Building High Radiation Monitoring, and Air Ejector Off-Gas Radiation Monitoring systems that are sensitive instrumentation circuits with low-level signals and are located in areas where the cables and connections could be exposed to adverse localized environments caused by heat, radiation, or moisture. The aging effects are managed by calibration, current/voltage, and time domain reflectometry testing. The current program will be enhanced to include a review of the calibration and cable testing results for cable aging degradation.

The team reviewed program bases documents, draft implementing procedure and industry guidance. The inspectors also conducted interviews and performed walkdowns with plant personnel.

For the Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrument Circuits Program, the inspectors concluded the applicant had conducted adequate evaluations as well as industry experience and historical reviews to determine aging effects managed by the Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrument Circuits Program. The applicant provided adequate draft guidance to ensure aging effects are appropriately managed.

Fire Protection Program

The Fire Protection Program is an existing program modified for the purpose of aging management credited with managing the fire barrier function aging effects in fire protection systems and a diesel-driven fire pump inspection program. The aging effects are managed by periodic inspection of fire barrier penetration seals, fire barrier walls, ceilings, floors, and all fire rated doors. The program is credited with managing loss of material aging effects in fuel oil lines of the diesel driven fire pump through periodic testing of the pump. This aging management program will also manage the aging effects of in-scope carbon dioxide and halon suppression systems, once enhancements are made to periodically inspect these systems.

The inspectors reviewed the Fire Protection Program as well as supporting documents to verify the effectiveness of the Fire Protection Program. The inspectors also conducted interviews and performed walkdowns of various fire protection systems with plant personnel to observe the effectiveness of the existing Fire Protection Program. Enhancements to the existing program include guidance to identify fire barrier degradation, surface integrity and clearance on fire doors inspected every two years, fire pump diesel fuel supply system external surface corrosion examinations, and external corrosion and damage inspections for halon and low-pressure carbon dioxide fire suppression systems. The inspectors noted an acceptable exception in the application of the NUREG-1801 guidance for 6-month periodicity on visual inspection and functional testing of halon and carbon dioxide fire suppressions. Oyster Creek Generating Station performs in-depth operational tests and inspections on an 18-month periodicity. The applicant does perform a weekly tank/charge check and a monthly valve position alignment check and will include visual inspections of external surfaces as an enhancement prior to the period of extended operation.

For the Fire Protection Program, the inspectors concluded the applicant had conducted adequate evaluations as well as industry experience and historical reviews to determine aging effects managed by the fire protection program. The applicant has provided adequate guidance to ensure aging effects are appropriately managed.

Fire Water System Program

The Fire Water System Program is an existing program modified for the purpose of aging management credited with managing the loss of material, microbiological influenced corrosion, and biofouling aging effects in fire water systems at Oyster Creek Generating Station. The aging effects are managed by periodic maintenance, testing, and inspection of system piping and components in accordance with codes and standards. The inspectors reviewed program bases documents, completed testing and maintenance procedures, corrective action reports, design documents, and industry guidance. The inspectors also conducted interviews and performed walkdowns of the fire water system with plant personnel. The fire water system is maintained in a pressurized state which provides the applicant with constant system integrity status. The piping internals are routinely inspected at various locations throughout the system for loss of material and biofouling. The following enhancements have been noted:

Sprinkler head inspections in accordance with NFPA 25 “Standard for the Inspection, Testing, and Maintenance of Water-Based Fire Protection Systems” (1998 Edition).

Samples will be submitted to a testing laboratory prior to being in service 50 years.

Inclusion of inspection of selected portions of the fire protection system piping located aboveground and exposed to water, by non-intrusive volumetric examinations.

Performance of water sampling for the presence of microbiological corrosion every 5 years.

Inclusion of visual inspection of the water storage tank heater pressure boundary components during the periodic tank internal inspection.

For the Fire Water System Program, the inspectors concluded the applicant had conducted adequate evaluations, as well as industry experience and historical reviews to determine aging effects managed by the fire water system program. The applicant provided an acceptable plan to implement adequate guidance and terms to ensure aging effects are appropriately managed.

Periodic Inspection of Ventilation Systems Program

The Periodic Inspection of Ventilation Systems Program is an existing program at Oyster Creek Generating Station modified for the purpose of aging management. The program is credited with managing loss of material, changes in material properties, and degradation of heat transfer in ventilation systems in the scope of license renewal (flexible connections, fan and filter housings, and access door seals). Instrument piping and valves, restricting orifices and flow elements, thermowells, and Standby Gas Treatment System ducting exposed to soil or sand will be added to the scope as enhancements to the program. The aging effects are managed by periodic inspections that will be condition monitoring examinations performed at susceptible locations in the systems, intended to assure that existing environmental conditions are not causing material degradation that could result in a loss of system intended functions.

The inspectors reviewed program bases documents, completed testing and maintenance procedures, corrective action reports, design documents, and industry guidance. The inspectors also conducted interviews and performed walkdowns of accessible portions of Standby Gas Treatment and Reactor Building Ventilation Systems with plant personnel.

Complete visual inspections and performance tests of all ventilation systems in scope are performed during system preventive maintenance activities on a frequency not to exceed five years. This includes system leakage and filter efficiency tests for Standby Gas Treatment, Reactor Building and Control Room Ventilation systems. An additional noted enhancement includes adding specific guidance to inspect for loss of material and material property changes.

For the Periodic Inspection of Ventilation Systems Program, the inspectors concluded the applicant had conducted adequate evaluations as well as industry experience and historical reviews to determine aging effects managed by an aging management program. The applicant provided adequate guidance to ensure aging effects are appropriately managed.

Periodic Inspection - Forked River Combustion Turbine

The Periodic Inspection - Forked River Combustion Turbine program is a new program credited with addressing the two Forked River Combustion Turbine power plant components in the scope of license renewal that require periodic monitoring of aging effects, and are not covered by other aging management programs. In the applicant's response to the NRC's requests for additional information, RAI 2.5.1.19-1, dated October 12, 2005 and November 11, 2005, the applicant expanded the single aging management for the Forked River Combustion Turbine to twelve aging management programs. This periodic inspection program is one of twelve programs that monitor the aging effects of the Forked River Combustion Turbine.

The Periodic Inspection - Forked River Combustion Turbine aging management program manages the aging effect of change in material properties, loss of material and reduction of heat transfer for systems, components, and environments. The aging effects are managed by periodic inspections that will be condition monitoring examinations performed at susceptible locations in the systems, intended to assure that existing environmental conditions are not causing material degradation that could result in a loss of system intended functions. These inspections will be performed on a periodicity not to exceed once every 10 years and will coincide with major combustion turbine maintenance inspections.

The two Forked River Combustion Turbines are owned, operated, and maintained by FirstEnergy, under contract to supply station blackout services to Oyster Creek Generating Station. The inspectors reviewed program bases documents, maintenance rule performance data, walkdown reports, and action logs. Applicable portions of the Interconnect and Station Blackout Agreements were reviewed. The inspectors also conducted interviews and performed walkdowns with Oyster Creek Generating Station and FirstEnergy personnel of the Forked River Combustion Turbine facility and portions of its switchyard. The inspectors observed maintenance activities conducted by General Electric for FirstEnergy on Forked River Combustion Turbine #2 during a minor outage.

Though the Forked River Combustion Turbines are operated and maintained by FirstEnergy, Oyster Creek Generating Station assigns a system engineer to monitor their performance via monthly data sets and logs received from the onsite FirstEnergy engineers. Significant events and maintenance on the Forked River Combustion Turbine are logged and evaluated by the Oyster Creek system engineer for further action.

At the time of this inspection, the implementing procedures for this program were not developed. Hence, the aging management program elements have not been negotiated with FirstEnergy to be added into the Station Blackout Agreement. The Office of Nuclear Reactor Regulation accepted AmerGen's response to 2.5.1.15-1 and 2.5.1.19-1 and requests for additional information. Based on discussions with applicant personnel and reviews of supporting documents, the inspectors concluded that the applicant has plans to develop adequate guidance and terms for implementation of the Periodic Inspection - Forked River Combustion Turbine Program. AmerGen will negotiate those terms into the station blackout agreement.

For the Periodic Inspection - Forked River Combustion Turbine Program, the inspectors concluded the applicant had conducted adequate evaluations as well as industry experience and historical reviews to determine aging effects managed by an aging management program. The applicant provided an acceptable plan to implement adequate guidance and terms to ensure aging effects are appropriately managed.

ASME, Section XI, Subsection IWE Program

The ASME, Section XI, Subsection IWE Program is an existing program modified for the purpose of aging management credited with managing the aging effects in drywell containment systems in Oyster Creek Generating Station. ASME Section XI, Subsection IWE provides for inspection of primary containment components and the containment vacuum breakers system piping and components. It covers steel containment shells and their integral attachments; containment hatches and airlocks, seals and gaskets, containment vacuum breakers system piping and components, and pressure retaining bolting. The aging effects are managed by periodic visual inspections, and periodic ultrasonic testing wall thickness measurements. Additionally, the applicant will conduct monitoring of leakage from the drywell sand bed region drains going forward, as an additional method to detect conditions favorable for corrosion to occur. Only the visual and ultrasonic examinations are given credit for managing the affects of aging.

The inspectors reviewed all of the licensee's ultrasonic thickness testing inspection results for the condition of the drywell from 1983 through 2002, evaluations and calculations of corrosion rates and projections of wall thickness for several locations on the drywell. Also, the inspectors reviewed video records of the sand bed region condition and the removal of the sand from the sand bed region. The inspectors reviewed the structural analysis performed to confirm the structural integrity of the drywell after the amount of corrosion had been determined. The inspectors reviewed the most recently completed visual inspection results of the drywell sand bed exterior coating and the UT measurements from higher elevations of the drywell.

For the ASME, Section XI, Subsection IWE Program, the inspectors concluded the applicant had conducted adequate evaluations as well as industry experience and historical reviews to determine aging effects managed by the ASME, Section XI, Subsection IWE Program. The applicant provided adequate guidance to ensure aging effects are appropriately managed, pending resolution of Safety Evaluation Report Open Items OI 4.7.2-1.1 through OI 4.7.1-1.4, and OI 4.7.2-3.

Protective Coating Monitoring and Maintenance Program

The Protective Coating Monitoring and Maintenance Program is an existing program credited with managing the aging effects on the internal and external surfaces of the torus and the condition of the drywell in the sand bed region in systems in Oyster Creek Generating Station. The aging effects are managed by visual inspections of the protective coatings on each component, and examination, evaluation and repair of all coating defects observed.

The inspectors reviewed the past inspection results in each area to understand what conditions are being documented, the method of evaluation of recorded indications, the repair methods used to fix any damaged or degraded coating. The inspectors also looked at the licensee's cause determination for the underlying corrosion phenomena and actions being taken to monitor the condition.

The team concluded that as long as the coating integrity was maintained by this program, the presence of water, as indicated by collection from the former sandbed area drains, would not affect the rate of corrosion of the drywell at the former sandbed area.

For the Protective Coating Monitoring and Maintenance Program, the inspectors concluded the applicant had conducted adequate evaluations as well as industry experience and historical reviews to determine aging effects managed by the Protective Coating Monitoring and Maintenance Program. The applicant provided adequate guidance to ensure aging effects are appropriately managed.

Above-Ground Outdoor Tank Monitoring Program

The Above-Ground Outdoor Tank Monitoring Program is a new program credited with managing the aging effects on above ground steel tanks in systems at the Oyster Creek Generating Station. The aging effects will be managed by periodic visual inspections, some nondestructive evaluation inspections based upon maintenance history and industry experience.

The inspectors reviewed the Oyster Creek Generation Station template used to guide and control this inspection effort, conducted field walkdowns of four of the tanks covered by the program and reviewed the industry operating experience which the licensee has used to prepare this inspection program.

For the Above-Ground Outdoor Tank Monitoring Program, the inspectors concluded the applicant had conducted adequate evaluations as well as industry experience and historical reviews to determine aging effects managed by the Above Ground Outdoor Tank Monitoring Program. The applicant provided adequate guidance to ensure aging effects are appropriately managed.

ASME Section XI, Subsection IWF

The ASME Section XI, Subsection IWF Program is an existing program credited with managing the aging effects in the ASME Section XI, Subsection IWF. Subsection IWF provides for periodic visual examination of ASME Section XI Class 1, 2, 3 and MC components and piping support members for loss of mechanical function and loss of material. Bolting is also included with these components, inspecting for loss of material and for loss of preload by inspecting for missing, detached, or loosened bolts.

The aging effects are managed by periodic visual examinations for corrosion and loss of material in structural members, loss of preload in bolting; missing, detached, or loosened members or bolts; and any degradation of protective coatings. The program has been enhanced by including additional MC components in the approved ASME Section XI, Inservice Inspection program.

The inspectors reviewed the program description, program basis documents, the currently approved ASME Section XI, Subsection IWF program, and the results of previous inspections and examinations. The documents reviewed and discussions with cognizant individuals indicated the operating experience of the In-service Inspection program at Oyster Creek, which includes ASME Section XI, Subsection IWF aging management activities, has not shown any adverse trend. Periodic self-assessments of the program have been performed to identify the areas that need improvement to maintain the quality and integrity of the program. The proposed aging management program based on the ASME Section XI, Subsection IWF, is generally consistent with the elements of XI.S3 of NUREG-1801 with some exceptions; e.g., NUREG1801 specifies ASME Section XI, 2001edition, including the 2002 and 2003 addenda, whereas, the station program is based on ASME Section XI, 1995 edition with 1996 addenda, an acceptable alternate edition of the code. The enhancements include additional MC supports and inspection of underwater supports.

For the ASME Section XI, Subsection IWF Program, the inspectors concluded the applicant had conducted adequate evaluations as well as industry experience and historical reviews to determine aging effects managed by an aging management program. The applicant provided adequate guidance to ensure aging effects are appropriately managed.

Masonry Wall Program

The Masonry Wall Program is credited with managing the aging effects in masonry walls at the Oyster Creek Generating Station as part of the Structural Monitoring Program. The aging effects are managed by a program of inspection of masonry walls for cracking on a frequency of four years to assure that the established evaluation basis for each masonry wall remains valid during the period of extended operation.

The inspectors reviewed the program description, program basis documents, the currently approved station procedures, the results of prior inspections, discussions with cognizant personnel, and a walkthrough visual examination of accessible masonry walls to assess the effectiveness of the current program. The scope of the program includes all masonry walls that perform intended functions in accordance with 10 CFR 54.4, and were covered by I. E. Bulletin 80-11.

The inspections are implemented though station procedures. Maintenance history has revealed minor degradation of masonry block walls; but none that could impact their intended function. In response to I.E. Bulletin 80-11, "Masonry Wall Design," and Information Notice 87-67, "Lessons Learned from Regional Inspections of Licensee Actions in Response to I.E. Bulletin 80-11," various actions have been taken. Actions have included program enhancements, follow-up inspections to substantiate masonry wall analyses and classifications, and the development of procedures for tracking and recording changes to the walls. These actions have addressed all concerns raised by I.E. Bulletin 80-11 and Information Notice 87-67, namely unanalyzed conditions, improper assumptions, improper classification, and lack of procedural controls. A review of operating experience indicates that the program is effective for managing aging effects of masonry walls.

For the Masonry Wall Program, the inspectors concluded the applicant had conducted adequate evaluations as well as industry experience and historical reviews to determine aging effects managed by an aging program. The applicant provided adequate guidance to ensure aging effects are appropriately managed.

Structures Monitoring Program

The Structures Monitoring Program is an existing program that has been modified, and will be further modified, for the purpose of aging management of structures and structural components, including structural bolting within the scope of license renewal at Oyster Creek Station. The program was developed based on Regulatory Guide 1.160, Revision 2, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," and NUMARC 93-01 Revision 2, "Industry Guidelines for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," to satisfy the requirement of 10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants".

The scope of the program also includes condition monitoring of masonry walls and water-control structures as described in the Masonry Wall Program and in the RG 1.127, and Inspection of Water-Control Structures Associated With Nuclear Power Plants aging management program. The enhanced program includes structures that are not monitored under the current term but require monitoring during the period of extended operation. Aging effects are managed by periodic visual inspections by qualified personnel to monitor structures and components for applicable aging effects. Specifically, concrete structures are inspected for loss of material, cracking, and a change in material properties. Steel components are inspected for loss of material due to corrosion. Masonry walls are inspected for cracking, and elastomers will be monitored for a change in material properties. Earthen structures associated with water-control structures and the Fire Pond Dam will be inspected for loss of material and loss of form. Component supports will be inspected for loss of material, reduction or loss of isolation function, and reduction in anchor capacity due to local concrete degradation.

Exposed surfaces of bolting are monitored for loss of material, due to corrosion, loose nuts, missing bolts, or other indications of loss of preload. The scope of the program will be enhanced to include structures that are not monitored under the current term but require monitoring during the period of extended operation.

The inspectors reviewed the program description, program basis documents, the currently approved station procedures, the results of prior inspections, discussions with cognizant personnel, and a walkthrough visual examination of accessible structural items, including reinforced concrete and structural steel members, components and systems to assess the effectiveness of the current program. The scope of the program also includes all masonry walls that perform intended functions in accordance with 10 CFR 54.4, and were covered by I. E. Bulletin 80-11. The inspections included a review of station procedures, maintenance history, inspection findings and followup of inspection findings, and current inspection schedules. Inspection frequency is every four years; except for submerged portions of water-control structures, which will be inspected when the structures are dewatered, or on a frequency not to exceed ten years. The program contains provisions for more frequent inspections to ensure that observed conditions that have the potential for impacting an intended function are evaluated or corrected in accordance with the corrective action process. The Structures Monitoring Program is consistent with the ten elements of aging management program XI.S6, "Structures Monitoring Program," specified in NUREG-1801.

For the Structures Monitoring Program, the inspectors concluded the applicant had conducted adequate evaluations as well as industry experience and historical reviews to determine aging effects managed by an aging management program. The applicant provided adequate guidance to ensure aging effects are appropriately managed.

Inspection of Water Control Structures

The Inspection of Water Control Structures Program is an existing program modified for the purpose of the aging management program credited with managing the aging effects in Water Control Structure systems at the Oyster Creek Generating Station. The aging effects are managed by periodic inspections of the water control structures for structural and hydraulic degradation, and potential loss of function of intended service. The Water Control Structure Program is a subpart of the main Structures Monitoring Program. It is based on the guidance provided in RG 1.127 and ACI 349.3R and will provide for periodic inspection of the Intake Structure and Canal, the Fire Pond Dam, and the Dilution structure. The program will be used to manage loss of material, cracking, and change in material properties for concrete components, loss of material and change in material properties for wooden components, and loss of material, and loss of form of the dam, and the canal slopes. Inspection frequency is every four years; except for submerged portions of the structures, which will be inspected when the structures are de-watered, or on a frequency not to exceed ten years. The program will be enhanced to ensure that water-control structures aging effects are adequately managed during the period of extended operation.

The inspectors reviewed the program description, program basis documents, the currently approved station procedures, the results of prior inspections, discussions with cognizant personnel, and a walkthrough visual examination of accessible water control structures, including components and systems to assess the effectiveness of the current program. As the Water Control Structures Monitoring Program is a subpart of the larger Structures Monitoring Program, this review was performed in conjunction with the comprehensive review of the main Structures Monitoring Program. Inspection of Water-Control Structures Associated with Nuclear Power Plants program is consistent with the ten elements of aging management program.

For the Inspection of Water Control Structures Program, the inspectors concluded the applicant had conducted adequate evaluations as well as industry experience and historical reviews to determine aging effects managed by an aging management program. The applicant provided adequate guidance to ensure aging effects are appropriately managed.

Metal Fatigue of Reactor Coolant Pressure Boundary

The Metal Fatigue of Reactor Coolant Pressure Boundary Aging Management Program monitors select components in the reactor coolant pressure boundary by tracking and evaluating contributing plant events. The Metal Fatigue of Reactor Coolant Pressure Boundary program monitors operating transients and, by way of a computer program, calculates up-to-date fatigue usage factors.

The design basis metal fatigue analyses for the reactor coolant pressure boundary are considered time limited aging analysis for the purposes of license renewal. The Metal Fatigue of Reactor Coolant Pressure Boundary Program provides an analytical basis for confirming that the number of cycles, established by the analysis of record, will not be exceeded before the end of the period of extended operation. In order to determine cumulative usage factors more accurately, the program will implement FatiguePro® fatigue monitoring software. FatiguePro® calculates cumulative fatigue using both cycle-based and stress-based monitoring. This provides an analytical basis for confirming that the number of cycles established by the analysis of record will not be exceeded before the end of the period of extended operation.

For the Metal Fatigue of Reactor Coolant Pressure Boundary Aging Management Program, the inspectors reviewed the program including the basis calculations, ongoing monitoring, corrective actions, limiting components, and current cumulative usage factors for the limiting components. The applicant provided adequate guidance to ensure aging effects are appropriately managed.

Isolation Condenser System Review

The Oyster Creek license renewal application listed a number of plant systems within the scope of license renewal. From this list the inspectors selected the isolation condenser system for a focused review to determine whether the applicant's aging management programs were adequate to effectively manage aging effects related to this component. The following aging management programs are credited for managing aging effects of the isolation condenser system: ASME Section XI In-service Inspection, Subsection IWB, IWC, and IWD; Bolting Integrity; BWR Stress Corrosion Cracking; One Time Inspection; Structures Monitoring Program; and, Water Chemistry. The inspectors focused on the loss of material aging effect to determine how it would be managed by the identified programs applied specifically to the Isolation Condenser System.

Although the Oyster Creek 10 CFR 50, Appendix K, design basis event analysis, no longer takes credit for the Isolation Condenser it is very important for post-accident heat removal and mitigation of event consequences. It ranks very high on the probabilistic risk worth for this reason. Because of its risk importance the inspectors reviewed the aging management programs given credit for managing the affects of aging in the system.

The Isolation Condenser System contains safety-related components relied upon to remain functional during and following design basis events. For example the primary coolant boundary must be maintained through the condenser. Additionally the failure of nonsafety-related structures and components in the Isolation Condenser System could potentially prevent the satisfactory accomplishment of a safety-related function. The isolation condenser also performs functions that support fire protection and station blackout.

AmerGen is committed, in their application documents, to maintaining the water environment of the secondary side because the integrity of the heat exchanger tubes can be affected from both the inside and outside. Additionally, the heat exchanger shell, and therefore, the secondary water environment, is part of the One-Time Aging Management program because it is required to maintain structural integrity during a design basis earthquake to support the heat exchanger tubing and the attached reactor coolant/steam line piping.

The applicant proposed using the ASME Section XI In-service Inspection, Subsections IWB, IWC, and IWD aging management program with the water chemistry aging management program to manage loss of material of the Isolation Condensers. The inspectors reviewed the Oyster Creek in-service inspection program procedure ER-OC-330-1001, ISI Program Plan Fourth Ten-Year Inspection Interval to verify that it was modified to include inspections of the isolation condenser tube side components, eddy current testing of the tubes, and inspection (VT or UT) of the tube sheet and channel head to ensure that degradation is not occurring and the components intended function will be maintained. The inspectors reviewed selected NDE reports of isolation condenser system piping and components, where degradation would result, to verify compliance with ASME Section XI Code.

The inspectors reviewed the UT wall thickness data sheet 96-023-03 from 1R16 refueling outage which documented shell thickness measurements of the "B" Isolation Condenser. The UT results indicate that the shell thickness was over 0.350 inches in most cases with one reading at 0.312 inches. The vendor drawing 1691-655-20 indicates the shell thickness is 0.375 inches with 0.100 inches corrosion allowance with a minimum of 0.275 inches. Therefore, the "B" Isolation Condenser meets the original design specifications. The inspectors noted that coating inspections performed by the applicant of the inside shell surface of the "B" Isolation Condenser during 1R16 in 1996 blistering of the coating was observed in most of the submerged sections. The coating on the inside shell of the isolation condensers is not credited in the aging management programs.

Based on discussions with applicant personnel and reviews of the One-time Aging Management Program basis documents, the inspectors determined the applicant has elected to perform a one-time aging management program inspection of the shell of isolation condenser prior to entering extended plant operations.

b. Observation

The inspectors identified an observation related to the monitoring of liquid leakage from the former drywell sand bed region related to the current operating license period. This observation was determined not to be safety significant and has been entered into the applicant's ongoing corrective action system.

A current commitment for monitoring the sand bed drains is in a staff Safety Evaluation Report transmitted by letter November 1, 1995. This Safety Evaluation Report requested a commitment to perform inspections "3 months after the discovery of any water leakage". Subsequent correspondence from General Public Utilities Nuclear Corporation the licensee, at the time, clarified the commitment after discussions with the staff. The commitment made and accepted by the staff in a February 15, 1996, letter was to perform an evaluation of the impact of any leakage during power operations and conduct additional inspections of the drywell approximately 3 months after discovery of the water leakage if the evaluation determines that it is warranted. This commitment was not meant to apply to minor leakage from normal refueling activities.

During the inspection, the NRC team requested a walkdown of the torus room. AmerGen staff walked down the torus room prior to the NRC team making entry. Water collection jugs, fed by tubing from the former drywell sand bed drains, were emptied prior to the NRC's walkdown, without taking samples of the water in the jugs or recording water levels. The fact that water was present in the jugs meant that leakage had been occurring. The applicant informed the NRC team that the bottles had been improperly emptied without measurement or analysis. Upon further investigation, the applicant could not find documentation that showed prior surveillance of the water drains had been completed. AmerGen staff also could not find documented evidence that strippable coating of the refueling channel had been applied. This strippable coating is used as a measure to limit or prevent water leakage during refueling operations.

The applicant stated that, although there was no formal leakage monitoring in place, there has been no previous reported evidence of leakage from the former sand bed drains. Issue Report #348545 was submitted into the corrective action process when the missed commitment and the improper emptying of the bottles were discovered. This corrective action will capture the commitment in the applicant's computerized scheduling process so that the required actions will be automatically prompted. Because there was no previously reported leakage, the applicant did not investigate the source of leakage, take corrective actions, evaluate the impact of leakage, or perform additional drywell inspections.

The applicant further stated that a number of actions had been taken to alleviate the previous water leakage problem since discovery of the consequent drywell shell corrosion in the early 1990's. Some of the significant actions consisted of inspections of the reactor cavity wall, remote visual inspection of the trough area below the reactor cavity bellows seal area, and subsequent repair of the trough area and clearing of its drain. Clearing of the trough drain and repair of the trough routed any leakage away from the drywell shell. In addition, AmerGen believes that the strippable coating was applied to the reactor cavity walls before the reactor cavity is filled with water as part of refueling activities to minimize the likelihood of leakage into the trough area.

The license renewal application does not take credit for the use of the strippable coating, or the monitoring of the water leakage in managing the aging affects on the liner. As long as the coating of the exterior surface of the former sand bed area is maintained, any amount of water can be present and have no affect on the corrosion rate. The thickness of the cylindrical portion of the liner is managed using ultrasonic testing and this program will capture any changes in corrosion rate due to water in the liner gap. AmerGen has taken corrective actions to ensure, in the future, the drains are monitored, and the strippable coating is applied.

c. Overall Findings

The inspection verified that there is an adequate approach to monitor and control the effects of aging so that the intended function(s) of systems, structures, and components, for which an aging management review is required, will be maintained consistent with the current licensing basis during the period of extended operation. The inspection verified documentation, procedures, guidance, and personnel, appropriately supported the license renewal application.

40A6 Meetings, Including Exit

The inspectors presented the inspection results to Mr. T. Rausch, Oyster Creek Generating Station Vice President, and other members of the licensee's staff in a meeting that was open for public observation on September 13, 2006. The licensee had no objections to the NRC observations. No proprietary information was provided to the inspectors during this inspection. The State of New Jersey, Department of Environmental Protection attended the exit meeting, and made a statement at the meeting concerning the observation associated with the monitoring of liquid leakage from the former drywell sandbed drains. In addition, they stated that a letter concerning this issue was sent to the NRC Region I Regional Administrator dated September 13, 2006. A copy of this letter is available in the NRC ADAMS document management system under ML062630218.

ATTACHMENT

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

J. Camire	System Manager
L. Corsi	Mechanical Engineer, LR Project
M. Gallagher	Vice-President, License Renewal
J. Hufnagel	Licensing Lead, LR Project
K. Muggleston	Mechanical Engineer, LR Project
A. Ouaou	Civil Engineer, LR Project
F. Polaski	License Renewal Manager
T. Quintenz	Site Lead Engineer, LR Project
D. Warfel	Technical Lead, LR Project
R. Francis	App J. Program Manager
K. Muggleston	Licensing
S. Getz,	License Renewal
L. Corsi	License Renewal
R. Gayley	FAC Program Manager
J. Watley	CCCW System Engineer
C. Roth	TBCCW System Engineer
R. Artz	Chemist
M. Miller	License Renewal
T. Trettel	Fire Protection System Engineer
J. Yuen	System Engineer - Ventilation
C. Micklo	License Renewal
J. Esch	FirstEnergy Engineer
R. Bonelli	FirstEnergy Engineer
R. Skelskey	System Engineer - FRCT
M. Filippone	System Manager
E. Johnson	System Manager
R. Pruthi	System Manager
S. Schwartz	System Manager
D. Spamer	Senior Engineer, Electrical

LIST OF DOCUMENTS REVIEWED

Drawings

Complete Set of License Renewal Drawings:

LR-BR-2002, Rev. 0
LR-BR-2003, Rev. 0
LR-BR-2004, Rev. 0
LR-BR-2005, Rev. 0
LR-BR-2006, Rev. 0
LR-BR-2007, Rev. 0
LR-BR-2008, Rev. 0
LR-BR-2009, Rev. 0
LR-BR-2010, Rev. 0
LR-BR-2011, Rev. 0
LR-BR-2012, Rev. 0
LR-BR-2013, Rev. 0
LR-BR-2014, Rev. 0
LR-BR-2015, Rev. 0
LR-BR-M0012, Rev. 0
LR-FP-SE-5419, Rev. 0
LR-GE-107C5339, Rev. 0
LR-GE-148F262, Rev. 0
LR-GE-148F437, Rev. 0
LR-GE-148F444, Rev. 0
LR-GE-148F711 Rev. 0
LR-GE-148F712, Rev. 0
LR-GE-148F723, Rev. 0
LR-GE-148F740, Rev. 0
LR-GE-197E871, Rev. 0
LR-GE-234R166, Rev. 0
LR-GE-237E487, Rev. 0
LR-GE-237E756, Rev. 0
LR-GE-237E798, Rev. 0
LR-GE-713E802, Rev. 0
LR-GE-865D741, Rev. 0
LR-GE-885D781 Rev. 0
LR-GU-3E-243-21-1000, Rev. 0
LR-GU-3E-551-21-1000, Rev. 0
LR-GU-3E-551-21-1001, Rev. 0
LR-GU-3E-666-21-1000, Rev. 0
LR-GU-3E-822-21-1000, Rev. 0
LR-GU-3E-861-21-1000, Rev. 0
LR-GU-3E-861-21-1001, Rev. 0
LR-GU-3E-861-21-1002, Rev. 0
LR-GU-3E-862-21-1000, Rev. 0

LR-GU-3E-871-21-1000, Rev. 0
LR-JC-147434, Rev. 0
LR-JC-19479, Rev. 0
LR-JC-19616, Rev. 0
LR-JC-19629, Rev. 0
LR-OC-010520, Rev. 0
LR-SN-13432.19, Rev. 0

Other Drawings

Drawing 4059-2, Sheet 2 or 3, Reactor Bldg. First Floor At Elev. 23' 6", Sections & Details - SH.2
Drawing 3E-SK-5-85, 1986 Drywell Data UT Location Plan
Drawing BE-SK-S-89, Revision 0, 10/16/89; Ultrasonic Testing Drywell Level 30'2" - 67'5"
M0123, Post Accident Sampling Isometric, Rev. 2
M0124, Post Accident Sampling Isometric, Rev. 2
M0278, Diesel Fuel Oil Storage Tank Isometric, Rev. 0

GU 3E-000-A3-002, Sheet 7, Rev.1, Isometric Composite Various Systems IGSCC Weld History
Foster Wheeler Drawing1691-655-20, Outline & Section of Emergency Condenser, Rev. F
Drawing 4059-2, Sheet 2 or 3, Reactor Bldg. First Floor At Elev. 23' 6", Sections & Details - SH.2
Drawing 3E-SK-5-85, 1986 Drywell Data UT Location Plan
Drawing BE-SK-S-89, Revision 0, 10/16/89; Ultrasonic Testing Drywell Level 30'2" - 67'5"

Procedures

MA-AA-723-500, Inspection of Non-EQ Cables and Connections for Managing Adverse Localized Environments, Draft Rev 2A.
621.3.005, High Radiation Monitor Calibration, Draft Rev 48A.
621.3.002, Air Ejector Off Gas Radiation Monitor Check Source Functional Test, Draft Rev 26A.
2400-SMI-3623.09, Calibration and Operation of the LPRM Diagnostic System, Rev 11.
2400-SMI-3623.08, IRM Detector Current-Voltage (I/V) Testing, Rev 6.
2400-SMI-3623.03, IRM, SRM, LPRM, Characterization Trending and Diagnostics, Rev 7.
2400-SME-3780.05, Power Factor Testing of 5kV Cables, Rev 2.
2400-SME-3780.06, Dielectric Testing for 2.3kV and 5kV Cables and Equipment, Rev 8.
Exelon Technical Specification for Distribution System Wood Pole Inspection and Remediation, Dated 1/1/05.
ECR OC 05-00275-00: Revise C-1302-187-E310-037, Revision 2
ER-AA-330-007
ER-AA-335-018
ER-AA-330, Revision 3: Conduct Of Inservice Inspection Activities
ER-AA-330-007, Revision 3: Visual Examination Of Section XI Class MC Surfaces And Class CC Liners

ER-AA-330-018, Revision 2: General, VT-1, VT-1C, VT-3 And VT-3C, Visual Examination Of ASME Class MC And CC Containment Surfaces And Components
2400-GMM-3900.52, Revision 3: Inspection And Torquing Of Bolted Connections
SM-AA-300, Revision 0; Procurement Engineering Support Activities
WO R2064827-06, Disassemble Reactor Vessel For Refuel Outage, Prepare Areas & Apply Cavity Coating, 11/1/06
WO R2068582-03, Perform Reactor Vessel Reassembly, Remove Cavity Coating And Decon Cavity, 11/1/06
Procedure No. 666.5.007, Revision 16; Primary Containment Integrated Leak Rate Test
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00470325
00472090
00461639
00348545*

*As a result of this inspection

WO R0806127-01
WO R2027889-01
WO R2071967-06

CAP O2003-2586, #2 Diesel-Driven Fire Pump Cooling Water Line pin-hole leaks
CAP O2002-0916, Water leakage from base of Fire Hydrant #9
CAP O2005-2288, Debris in SGTS #1 Filter Train

Aging Management Programs

PBD-AMP-B.1.27, Revision 0, ASME Section XI, Subsection IWE
PBD-AMP-B.1.33, Revision 0, Protective Coating Monitoring And Maintenance Program
PBD-AMP-B.1.21, Aboveground Steel Tanks
PBD-AMP-B.1.34, Electrical Cables and Connections not Subject to 10 CFR 50.49 Environmental Qualification Requirements, Rev 0
PBD-AMP-B.2.01, Periodic Testing of Containment Spray Nozzles, Rev 0
PBD-AMP-B.1.35, Electrical Cables and Connections not Subject to 10 CFR 50.49 Environmental Qualification Requirements used in Instrumentation Circuits, Rev 0

PBD-AMP-B.1.36, Inaccessible Medium-Voltage Cables not Subject to 10 CFR 50.49 Environmental Qualification Requirements, Rev 0
PBD-AMP-B.2.06, Wooden Utility Pole Program, Rev 0

License Renewal Change Requests

LRCR 291
LRCR 290
LRCR 289

LIST OF ACRONYMS

ADAMS	Agency-wide Documents Access and Management System
ASME	American Society Mechanical Engineers
PARS	Publicly Available Records
GALL	Generic Aging Lessons Learned Report

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

ACRSR-2233

February 8, 2007

The Honorable Dale E. Klein
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: REPORT ON THE SAFETY ASPECTS OF THE LICENSE RENEWAL APPLICATION FOR THE OYSTER CREEK GENERATING STATION

Dear Chairman Klein:

During the 539th meeting of the Advisory Committee on Reactor Safeguards, February 1-3, 2007, we completed our review of the license renewal application for the Oyster Creek Generating Station (OCGS) and the updated Safety Evaluation Report (SER) prepared by the NRC staff. Our Plant License Renewal Subcommittee also reviewed this matter during meetings on October 3, 2006 and January 18, 2007. During these reviews, we had the benefit of discussions with representatives of the NRC staff and its contractor Sandia National Laboratories (SNL), members of the public, and AmerGen Energy Company, LLC (AmerGen) and its contractors. We also had the benefit of the documents referenced. This report fulfills the requirements of 10 CFR 54.25 that the ACRS review and report on all license renewal applications.

RECOMMENDATIONS

1. With the incorporation of the conditions described in Recommendations 2, 3, and 4, the application for license renewal for OCGS should be approved.

2. We concur with the staff's proposal to impose license conditions to increase the frequency of the drywell inspections and to monitor the two drywell trenches to ensure that the sources of water are identified and eliminated.

3. The staff should add a license condition to ensure that the applicant fulfills its commitment to perform an engineering study prior to the period of extended operation to identify options to eliminate or reduce the leakage in the OCGS refueling cavity liner.

4. The staff should add a license condition to ensure that the applicant fulfills its commitment to perform a 3-D (dimensional) finite-element analysis of the drywell shell prior to entering the period of extended operation.

DISCUSSION

The Oyster Creek Generating Station is located in Lacey Township, Ocean County, New Jersey, approximately 2 miles south of the community of Forked River, 2 miles inland from the shore of Barnegat Bay, and 9 miles south of Toms River, New Jersey. The NRC issued the provisional operating license for OCGS on April 9, 1969 and the operating license on July 2,

1991. OCGS is a single unit facility with a single cycle, forced circulation boiling water reactor (BWR)-2 with a Mark 1 containment. The nuclear steam supply system was furnished by General Electric and the balance of the plant was originally designed and constructed by Burns & Roe. The licensed power output is 1930 MWt with a design electrical output of approximately 650 MWe. The applicant, AmerGen requested renewal of the OCGS operating license for 20 years beyond the current license term, which expires on April 9, 2009.

During the 1980s, the licensee discovered corrosion on the outside wall of the OCGS drywell shell. Although some corrosion had occurred in the upper shell region, the majority had occurred in a region near the base of the shell where the shell was partially supported by a sand bed. The licensee determined that water had been leaking through flaws in the refueling cavity liner during refueling operations. This water had migrated down the outside of the drywell shell and into the sand bed. As part of the corrective actions, the licensee removed the sand and applied an epoxy coating to the outside of the shell in the sand bed region. In addition, repairs were made to the refueling pool liner and the concrete drain trough under the refueling seal. These repairs reduced the leakage and routed any leakage to a drain line rather than down the outside of the drywell shell. To further reduce leakage, the licensee applied strippable coatings to the liner during all but one of the subsequent refueling outages. The licensee performed ultrasonic testing (UT) to determine the as-found condition of the drywell shell and performed a structural analysis in 1992 to demonstrate acceptability of the containment in the degraded condition.

The 1992 structural analysis was reviewed and approved by the NRC staff. This analysis included a determination of the stresses in the thinned region under the design pressure loads and an evaluation of the potential for buckling during normal operations and postulated accident conditions. The buckling analysis utilized American Society of Mechanical Engineers (ASME) Code Case N-284, Revision 1. The staff accepted the use of this Code Case in the 1992 analysis. In support of the review of the OCGS license renewal application, the staff had SNL perform a confirmatory structural analysis. Both analyses demonstrated that the drywell shell met the minimum ASME Code requirements for buckling. However, the amount of margin above the Code minimum depended on the applicability of the increase in the buckling capacity due to tensile stresses orthogonal to the applied compressive stresses computed according to the Code Case. During the January 18, 2007 meeting, the Subcommittee requested additional justification for using the increased capacity factor. At our February meeting, Dr. C. Miller, the author of the ASME Code Case, described the technical basis for the Code Case and presented test results to demonstrate that the increased capacity factor was applicable to OCGS. The increased capacity factor used in the 1992 analysis provided by the applicant was based on results for metal cylinders. Dr. Miller showed results of tests conducted on metal spheres which demonstrated that the results for cylinders were conservative for spherical shells. The staff reaffirmed its position that the use of the increased capacity factor is appropriate for the analysis of the OCGS drywell shell. We concur with this position.

The 1992 structural analysis was based on the assumption that the shell is uniformly thinned in the sand bed region. The applicant has committed to perform a 3-D finite-element analysis of the OGCS drywell to determine the margin of the shell in the as-found condition using modern methods. This analysis will provide a more accurate quantification of the margin above the Code required minimum for buckling. The applicant has committed to complete the analysis prior to the period of extended operation. We commend the applicant for this action and would like to be briefed by the staff on the results when they become available. Although it is anticipated that the analysis will demonstrate additional margin above the Code required minimum, the applicant should complete this analysis in a timely manner prior to entering the period of extended operation in order to identify and resolve any unexpected results. The analysis should include sensitivity studies to determine the degree to which uncertainties in the size of thinned areas affect the Code margins. The staff should impose a license condition to ensure that the applicant completes the analysis prior to entering the period of extended operation.

In 2006, the applicant performed additional UT and visual inspections of the drywell shell. When compared to the previous UT, the 2006 results confirmed that the corrective actions taken in the sand bed region had been effective and that the corrosion had been arrested or at least that the corrosion rates were very low (i.e., within the data scatter). The epoxy coating appeared in very good condition with no evidence of degradation which is also consistent with the conclusion that the corrosion has been effectively arrested. These examinations also demonstrated that the corrosion rate in the upper shell region and the embedded floor regions remained sufficiently low to demonstrate structural integrity during the period of extended operation. The applicant has committed to perform UT and visual inspections of the drywell shell during the period of extended operation. Because of the relatively small margin above the Code minimum against buckling in the sand bed region shown by current analyses, the staff is proposing a license condition to increase the frequency of drywell inspections and UT in the sand bed region to all 10 bays every other refueling outage for the extended period of operation. Increased inspections will result in additional radiation exposure to personnel involved in the inspections. Therefore, the applicant should be allowed to increase the period between inspections if it demonstrates increased margin through analysis or if the ongoing inspections continue to demonstrate that the corrosion has been sufficiently arrested. With this provision, we agree with this license condition.

The 2006 examinations revealed that when the cavity was flooded for refueling, water leakage was still occurring. This leakage of approximately 1 gallon per minute is well within the capacity of the drain as long as the drain system is working properly. The purpose of the drain system is to catch water that may leak past a failed refueling seal or liner and divert the water to sumps, and prevent it from coming into contact with the outside of the drywell shell. Leakage is not expected to occur as part of normal operation with properly maintained equipment and structures. The applicant has committed to continue monitoring for leakage of the refueling cavity liner and other water sources associated with the drywell. The applicant has also committed to complete an engineering study to identify cost-effective repair or replacement options to eliminate the refueling cavity liner leakage. The engineering study will be completed prior to entering the period of extended operation. We agree that efforts should be made to eliminate routine leakage in order to provide increased protection against further degradation. The staff should impose a license condition to ensure the study is completed by the applicant prior to the period of extended operation.

During the 2006 refueling outage, the applicant discovered water in two trenches that had been previously excavated to allow access to and inspection of the inside of the shell in the embedded region. The applicant determined that the water had come from normal operation and maintenance activities. The water had migrated to the trenches due to a blocked drain tube in the sub-pile area and the lack of a seal between the shell and concrete curb. The applicant repaired the drain tube and installed a seal in the gap between the shell and concrete curb. The applicant intends to fill these trenches after two consecutive outages in which no water is observed. Having the trenches open is beneficial for identifying drainage issues, but it increases the risk of additional corrosion because it provides an open area in which water can be trapped against the shell. The staff is proposing a license condition that would require the applicant to leave the trenches open and monitor them during each refueling outage until such time that the applicant can demonstrate that the water sources have been identified and eliminated. We agree with the monitoring of the trenches to ensure the elimination of the sources of water. However, leaving the trenches open longer than necessary increases the risk of future corrosion. Therefore, the applicant should not be unnecessarily delayed in repairing the trenches. With this provision, we agree with the license condition proposed by the staff.

In the updated SER, the staff documents its review of the license renewal application and other information submitted by AmerGen and obtained during an audit and inspections conducted at the plant site. The staff reviewed the completeness of the applicant's identification of structures, systems, and components (SSCs) that are within the scope of license renewal; the integrated plant assessment process; the applicant's identification of the plausible aging mechanisms associated with passive, long-lived components; the adequacy of the applicant's aging management programs (AMPs); and the identification and assessment of time-limited aging analyses (TLAAs) requiring review.

The OCGS application either demonstrates consistency with the Generic Aging Lessons Learned (GALL) Report or documents deviations from the approaches specified in the GALL Report. The staff reviewed this application in accordance with NUREG-1800, the "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants."

The applicant identified those SSCs that fall within the scope of license renewal. For these SSCs, the applicant performed a comprehensive aging management review. Based on the results of this review, the applicant will implement 57 AMPs for license renewal including existing, enhanced, and new programs. In the SER, the staff concludes that the applicant has appropriately identified SSCs within the scope of license renewal and that the AMPs described by the applicant are appropriate and sufficient to manage aging of long-lived passive components that are within the scope of license renewal. With the incorporation of the license conditions described in Recommendations 2, 3 and 4, we agree with this conclusion.

The staff conducted inspections and an audit of the license renewal application. The purpose of the inspections was to verify that the scoping and screening methodologies are consistent with the regulations and are adequately reflected in the application. In addition, the inspectors personally examined selected areas of the sand bed region to verify the condition of the epoxy coating. The audit confirmed the appropriateness of the AMPs and the aging management reviews. Based on the inspections and audit, the staff concluded that these programs are consistent with the descriptions contained in the OCGS license renewal application. The staff also concluded that the existing programs, to be credited as AMPs for license renewal, are generally functioning well and that the applicant has established an implementation plan in its commitment tracking system to ensure timely completion of the license renewal commitments.

The applicant identified those systems and components requiring TLAAs and reevaluated them for 20 more years of operation. Affected TLAAs include those associated with neutron embrittlement, metal fatigue, irradiation-assisted stress corrosion cracking, environmental qualification of electrical equipment, and stress relaxation of hold-down bolts. The staff concluded that the applicant has provided an adequate list of TLAAs. Further, the staff concluded that in all cases the applicant has met the requirements of the license renewal rule by demonstrating that the TLAAs will remain valid for the period of extended operation, or that the TLAAs have been projected to the end of the period of extended operation, or that the aging effects will be adequately managed for the period of extended operation. With the incorporation of the license conditions described in Recommendations 2, 3 and 4, we concur with the staff that OCGS TLAAs have been properly identified and that criteria supporting 20 more years of operation have been met.

With the incorporation of the license conditions described in Recommendations 2, 3, and 4, no issues related to the matters described in 10 CFR 54.29(a)(1) and (a)(2) preclude renewal of the operating license for OCGS. The programs established and committed to by AmerGen provide reasonable assurance that OCGS can be operated in accordance with its current licensing basis for the period of extended operation without undue risk to the health and safety of the public and the NRC should approve the AmerGen application for renewal of the operating license for OCGS.

Sincerely,

/RA/

William J. Shack
Chairman

References:

1. Updated Safety Evaluation Report Related to the License Renewal of Oyster Creek Generating Station, December 29, 2006.

2. Safety Evaluation Report with Open Items Related to the License Renewal of the Oyster Creek Generating Station, August 18, 2006.

3. Oyster Creek Generating Station- Application for Renewed Operating Licenses, July 22, 2005.

4. Supplemental Information Related to the Aging Management Program for the Oyster Creek Drywell Shell, Associated with AmerGen's License Renewal Application, June 20, 2006.

5. Audit and Review Report for Plant Aging Management Reviews and Programs- Oyster Creek Generating Station August 18, 2006.

6. Supplemental Response to NRC Request for Additional Information (RAI 2.5.1.19-1), dated September 28, 2005, Related to Oyster Creek Generating Station License Renewal Application, November 11, 2005.

7. Oyster Creek Generating Station - NRC License Renewal Inspection Report 05000219/2006007, September 21, 2006

8. Memorandum dated December 14, 2006 from Louise Lund to John Larkins, Subject: Review Background Materials for the Meeting of the License Renewal Subcommittee Scheduled on January 18, 2007, Related to the Interim Review of the License Renewal of the Oyster Creek Generating Station. ML063470557

9. Memorandum date December 8, 2006 from Michael P. Gallagher to the U.S. Nuclear Regulatory Commission, Subject: Submittal of Information to ACRS Plant License Renewal Subcommittee Related to AmerGen's Application for Renewed Operating License for Oyster Creek Generating Station. ML063470532

10. Sandia National Laboratories Report "Structural Integrity Analysis of the Degraded Drywell Containment at the Oyster Creek Nuclear Generating Station," January 2007

11. ASME Code Case N-284-1, "Metal Containment Shell Buckling Design Methods, Class MC, Section III, Division one, March 14, 1995."

12. Letter dated January 31, 2007, from Senator Frank Lautenberg, Senator Robert Menendez, Representative Christopher H. Smith, and Representative Jim Saxton to The ACRS.

13. Letter dated January 31, 2007 from Richard Webster, Rutgers Environmental Law Clinic to the ACRS, regarding the Safety Evaluation Report for Oyster Creek Nuclear Power Plant.

14. Oyster Creek Generating Station-NRC In-Service Inspection and License Renewal Commitment Followup Inspection Report 0500021/2006013, January 17, 2007.

embrittlement, metal fatigue, irradiation-assisted stress corrosion cracking, environmental qualification of electrical equipment, and stress relaxation of hold-down bolts. The staff concluded that the applicant has provided an adequate list of TLAAs. Further, the staff concluded that in all cases the applicant has met the requirements of the license renewal rule by demonstrating that the TLAAs will remain valid for the period of extended operation, or that the TLAAs have been projected to the end of the period of extended operation, or that the aging effects will be adequately managed for the period of extended operation. With the incorporation of the license conditions described in Recommendations 2, 3 and 4, we concur with the staff that OCGS TLAAs have been properly identified and that criteria supporting 20 more years of operation have been met.

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/RA/

William J. Shack
Chairman

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March 8, 2007

Dr. William J. Shack, Chairman
Advisory Committee on Reactor Safeguards
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: RESPONSE TO ADVISORY COMMITTEE ON REACTOR SAFEGUARDS REPORT ON THE SAFETY ASPECTS OF THE LICENSE RENEWAL APPLICATION FOR THE OYSTER CREEK GENERATING STATION

Dear Dr. Shack:

During the 539th meeting of the Advisory Committee on Reactor Safeguards (ACRS or the Committee) held on February 1–3, 2007, the ACRS completed its review of the license renewal application (LRA) for the Oyster Creek Generating Station (OCGS) and the associated final safety evaluation report (SER) prepared by the U.S. Nuclear Regulatory Commission (NRC) staff. In its final report, the Committee recommends renewal of the OCGS operating license in conjunction with the recommendations discussed in your letter dated February 8, 2007. The staff appreciates the Committee's expeditious, objective, and in-depth review of the LRA and the staff's final SER. The staff agrees with the Committee's recommendations:

1. The staff will impose a license condition to increase the frequency of the drywell inspections and to monitor the two drywell trenches to ensure that the sources of water are identified and eliminated.

2. The staff will ensure that the applicant fulfills its commitment to (a) perform an engineering study prior to the period of extended operation to identify options to eliminate or reduce the leakage in the OCGS refueling cavity liner, and (b) perform a 3-D (dimensional) finite-element analysis of the drywell shell prior to entering the period of extended operation.

The staff recognizes the ACRS's commitment to safety and appreciates the Committee's continued support of the license renewal process.

Sincerely,

/RA/

Luis A. Reyes
Executive Director
for Operations

cc: Chairman Klein
Commissioner McGaffigan
Commissioner Merrifield
Commissioner Jaczko
Commissioner Lyons
SECY

March 8, 2007

Dr. William J. Shack, Chairman
Advisory Committee on Reactor Safeguards
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

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Sincerely,

/RA/

Luis A. Reyes
Executive Director
for Operations

cc: Chairman Klein
Commissioner McGaffigan
Commissioner Merrifield
Commissioner Jaczko
Commissioner Lyons
SECY

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Letter to W. Shack, from L. Reyes, dated: March 8, 2007

SUBJECT: RESPONSE TO ADVISORY COMMITTEE ON REACTOR SAFEGUARDS REPORT ON THE SAFETY ASPECTS OF THE LICENSE RENEWAL APPLICATION FOR OYSTER CREEK GENERATING STATION

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