October 27, 2009

The Honorable Christopher H. Smith
United States House of Representatives
Washington, D.C. 20515

Dear Congressman Smith:

On behalf of the Commission, I am responding to your letter of May 18, 2009, in which you expressed concern regarding the U.S. Nuclear Regulatory Commission (NRC) staff's review of the 3-D finite element analysis (3-D FEA) of the Oyster Creek (OC) drywell shell. I want to assure you that safe and secure operations of the Oyster Creek nuclear power plant are of primary importance to the NRC. The concerns you raise are important and have been addressed in our review. The Commission has considered your request to have Sandia National Laboratories (Sandia) review the 3-D FEA and has concluded that substantial and sufficient analyses and reviews have been performed of the drywell shell. Sandia has previously analyzed the drywell shell and concluded that the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code requirements were met. Three separate analyses of the drywell shell and independent reviews of the analyses have concluded that there is sufficient margin and confirmed that the ASME Code requirements have been met, such that the plant can be operated safely for the additional 20-year license period. Under these circumstances, we do not believe that a review of the 3-D FEA by Sandia is necessary to ensure public safety.

As noted in the enclosure (Enclosure 1), the 3-D FEA performed by Structural Integrity Associates (SIA) is one of three separate analyses that have been performed for OC's drywell shell. The analysis of record for the drywell, as referenced in OC's Final Safety Analysis Report, was conducted by General Electric in the early 1990's timeframe and was independently reviewed by structural engineering experts at Brookhaven National Laboratory. The analysis performed by Sandia in the 2006 timeframe was requested by NRC staff as part of the OC license renewal review and was reviewed by the Advisory Committee on Reactor Safeguards (ACRS), the NRC's independent oversight panel for reactor safety issues, which reviews all license renewal applications. The 3-D FEA was performed by SIA, a contractor to the licensee (Exelon), in the 2008 timeframe, and was submitted by Exelon in response to comments from the ACRS. The 3-D FEA, besides being reviewed by the NRC staff, was also independently reviewed by Becht Nuclear Services (Becht), a contractor engaged by the State of New Jersey. Becht's review of the 3-D FEA concluded that OC's drywell shell met the ASME Code requirements.

The three separate analyses used different approaches to model the current state of the drywell and concluded that OC's drywell shell meets the ASME Code requirements, ensuring public safety. The modification to the capacity reduction factor (CRF), as used by General Electric and SIA, was supported by extensive testing by a structural engineering expert who was working at the vendor that designed and built OC's drywell shell. His approach for modifying the CRF has been well-vetted through the ASME consensus Code process. The ASME Code committee is an independent body that uses a consensus process by gathering structural engineering experts from multiple organizations, including the national laboratories, to determine appropriate engineering practices. The results of their reviews are compiled into a code and code cases, which are used internationally. The ASME approach for modifying the CRF used by General Electric and SIA has been reviewed and codified into three separate ASME Code cases. The three separate analyses and the rigorous CRF review by ASME provide confidence that the drywell shell will maintain structural integrity during the 20-year license period.

The Commission instructed the NRC staff, as informed by the recommendations in the Atomic Safety and Licensing Board's (ASLB) advisory opinion, to use its expertise and engineering judgment to scrutinize the 3-D FEA carefully. Registered professional structural engineers on the NRC staff with appropriate expertise performed the review. One of the engineers has 35 years of structural engineering experience in nuclear power plant applications and the other engineer has 40 years of experience, including 37 years in nuclear power plant applications. Both engineers represent the NRC in a number of organizations that develop standards, including the ASME, the American Concrete Institute (ACI), and the American Institute of Steel Construction. One of the engineers is a Fellow of the ACI and the American Society of Civil Engineers. Consistent with the Commission's April 1, 2009, Order, the staff performed a thorough review of the 3-D FEA that included an examination of the supporting documentation for the analysis and discussions with SIA.

Although the Commission adopted the recommendation in ASLB Judge Abramson's separate advisory opinion that the staff "engage appropriate expertise to conduct a thorough examination of the analysis once submitted," the Commission did not direct the staff specifically to have Sandia review the 3-D FEA. The staff was directed to suitably and appropriately inform its review by the recommendations in the ASLB Advisory Opinion. As discussed in Enclosure 1, the staff considered the ASLB recommendations as part of its review. It is important to note that Sandia's analysis (performed in 2006 as part of NRC's license renewal review) already concluded that the ASME Code margins were met. Since the time that Sandia conducted its analysis, no new technical information has been identified that would change those conclusions.

Based on the comprehensive analyses and reviews performed to date, the NRC has concluded that the ASME Code requirements are met for OC's drywell shell. Additionally, the aging management programs that Exelon is implementing provide reasonable assurance that the OC containment structure will continue to satisfy its safety requirements throughout the additional 20-year license period.

Throughout the review of the OC license renewal application, the NRC has been committed to ensuring dissemination of public information on renewal issues and opportunities for public involvement. Our decision has been informed by the participation of interested parties in the adjudicatory process before the ASLB and the Commission. The agency has responded to correspondence expressing concerns about the drywell issue and has engaged in other public outreach to address the evaluations of the drywell shell. This involvement in agency adjudication and outreach is an important part of our commitment to fair and transparent regulation.

In your letter, you requested an outside review by Sandia of the OC analysis. For the reasons stated above, after considering the factors prescribed by the Commission, the staff determined that a review by Sandia was not needed. However, the ACRS was planning on conducting an information briefing and expanded it to complete an evaluation of the OC analysis. On September 23, 2009, the ACRS Subcommittee on Materials, Metallurgy, and Reactor Fuels held a public meeting to review the Oyster Creek 3-D FEA of the drywell shell. This meeting was noticed in the *Federal Register* and stakeholders who had previously expressed an interest in this matter were notified of the meeting. The NRC staff, Exelon, and external stakeholders presented information on the analysis to the ACRS Subcommittee for its consideration. Subsequently, on October 8, 2009, the ACRS Full Committee held a public meeting to review the SIA 3-D FEA. This meeting was also noticed in the *Federal Register*, and the NRC staff and Exelon presented information to the Full Committee on the 3-D FEA. The ACRS assessed the 3-D FEA and the information provided by the NRC staff, licensee, and external stakeholders.

Based on its review, the ACRS issued a letter dated October 16, 2009 (Enclosure 2), which provides its findings and conclusion. The ACRS concluded that the analysis presented by Exelon fulfilled its commitment to provide a modern, realistic, 3-D finite element analysis that better qualifies the available safety margin for the current configuration of the OC drywell shell. The ACRS also concluded that the analysis confirms that the OC drywell shell complies with its current licensing basis for design basis accidents with margin, and that the analysis was performed using good engineering practices and judgment and used conservatively biased realistic assumptions. I believe the review conducted by the ACRS is responsive to your request, and hope that it facilitates a greater understanding of the 3-D FEA and basis for acceptability.

Thank you for your interest in this matter and I appreciate your taking the time to discuss this and other relevant matters.

Sincerely,

/RA/

Gregory B. Jaczko

Enclosures:
1. Staff Assessment of the Oyster Creek 3-D Finite Element Analysis of the Drywell Shell
2. Letter from Mario Bonaca, ACRS, dated October 16, 2009, “Report on the 3-Dimensional Finite Element Analysis of the Oyster Creek Nuclear Generating Station Drywell Shell”

STAFF ASSESSMENT OF THE OYSTER CREEK 3-D FINITE ELEMENT ANALYSIS OF THE DRYWELL SHELL

INTRODUCTION

The NRC Staff ("Staff") completed a comprehensive review using our expertise and engineering judgment to scrutinize carefully Oyster Creek Nuclear Generating Station's ("Oyster Creek") three dimensional finite element analysis ("3-D FEA"). Based on the Staff's review, the 3-D FEA provides a conservatively-biased, realistic analysis of the available ASME code margins utilizing a more modern computer analysis and sensitivity analyses to account for uncertainties in the extent of degradation between the measured locations in the drywell shell. Based on the Staff's comprehensive review, Oyster Creek's 3-D FEA shows that the drywell shell meets the ASME code margins under all the postulated conditions, as is discussed in more detail below.

I. Background

The Staff completed a comprehensive review of the Oyster Creek license renewal application ("LRA") in April 2007. Oyster Creek is currently owned and operated by Exelon Generation Company, LLC (Exelon or the applicant).[1] The Staff determined that Exelon demonstrated that aging effects will be adequately managed to support operation for an additional 20 years and issued a renewed license on April 8, 2009.[2]

As part of the LRA review, the Staff reviewed the aging management program for the drywell shell. The plant contains a drywell shell constructed of carbon steel, which is approximately 100 feet tall and is shaped like an inverted light bulb (Figure 1). It measures approximately 70 feet in diameter at the spherical base. At an elevation of 71 feet 6 inches, it transitions from a spherical shape to a cylindrical shape that is approximately 33 feet in diameter. In the late 1980s, corrosion was discovered in an area in the spherical section of the shell marked in Figure 1 as the sandbed region, which is divided into 10 bays each designated with an odd number from 1 through 19. After removing the sand from this region, it was found that corrosion was unevenly distributed within the ten bays of the drywell shell, with Bays 1, 11, 13, 17, and 19 being the most affected bays. An analysis was performed by General Electric ("GE") during that period, which is the analysis of record and Oyster Creek's current licensing basis, to confirm that the drywell shell had sufficient structural integrity for continued operations. To support that analysis, the applicant evaluated the degradation of the shell, taking approximately 1,000 ultrasonic ("UT") measurements. Based on the UT measurements, areas on the external side of the shell that were considered the most thinned were ground flat for UT monitoring and grids were established for UT monitoring on the inside of the shell. The Staff reviewed the GE analysis and documented its review in the "Evaluation Report on the Structural Integrity of the Oyster Creek Drywell," dated April 24, 1992. The sandbed region was subsequently coated to protect the shell from further corrosion. Since discovery of the corrosion, monitoring of the drywell shell by UT measurements has continued, with the most recent measurements taken in 2008.

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[1] Oyster Creek was previously owned by AmerGen Energy Company ("AmerGen") and the license renewal application was filed by AmerGen. For the purposes of this report, Exelon will be used for consistency.

[2] Agency Wide Document Access and Management System ("ADAMS") Accession No. ML080280440.

ENCLOSURE

During the review of the drywell shell, the staff evaluated the information in the license renewal application including: (1) the analysis of record, which was performed by General Electric ("GE") (References 1 and 2); (2) the confirmatory analysis performed for the NRC by Sandia National Laboratories ("Sandia") (Reference 3), and (3) the applicant's commitments (see Commitment 27 in Appendix A of the Staff's safety evaluation report ("SER")).[3] Based on the analysis of record and the Sandia confirmatory analysis, commitments, along with the other provisions of the aging management program, the Staff concluded that reasonable assurance exists that the drywell shell will withstand the postulated conditions, without exceeding the ASME code margins, during the period of extended operation.

During the review of Oyster Creek's license renewal application, the Advisory Committee on Reactor Safeguards ("ACRS") met on February 1, 2007. During the February meeting, Exelon committed to perform a 3-D FEA of the Oyster Creek drywell shell in the as-found condition using more modern computational methods. The analysis of record performed by GE used many conservatisms due to the limitations of computer modeling, and a more modern 3-D FEA was intended to provide a more realistic assessment of the actual factor of safety for the drywell shell. Exelon's commitment to perform the 3-D FEA was incorporated as Item 18 of Commitment 27.[4] The commitment was also entered as a proposed license condition in the Staff's SER and is a current condition of the renewed license. The 3-D FEA provides a conservatively-biased but realistic quantification of the available margins above the ASME code minimum for buckling. It also adequately bounds the effect of the measurement uncertainties in the size of the thinned areas on the ASME code margins based.

Exelon submitted a Summary Report (Reference 4), describing the 3-D FEA, in fulfillment of its commitment, by letter dated January 22, 2009. As part of the Staff's review of the 3-D FEA, supporting reports and calculations were reviewed as part of the inspection completed in March 2009. Reference 4 presents the results of a baseline analysis of the postulated conditions and sensitivity analysis to account for measurement uncertainties. Enclosure 1 of the summary report, referred to as the Baseline Analysis, summarizes the analysis used to define the baseline degradation of the drywell shell, the 3-D FEA model, loads and load combinations imposed on the shell, the analysis results, the buckling evaluation, and Exelon's evaluation of the available margin with respect to the ASME Section III, Subsection NE acceptance criteria. Enclosure 2 of the report, referred to as the Sensitivity Analysis, utilizes the same inputs and processes as in the Baseline Analysis with the exception that the baseline thicknesses are reduced in order to capture measurement uncertainties in existing thickness measurements of the sandbed.

In Reference 4, Exelon determined that for the two sensitivity analyses, the available safety factors exceed the ASME Section III, Subsection NE acceptance criteria for buckling. The applicant indicated that the 3-D FEA incorporates data on the drywell shell thicknesses through the applicant's 2008 refueling outage inspection.

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[3] Safety Evaluation Report Related to the License Renewal of Oyster Creek Generating Station, Docket No. 50-219, March 2007.

[4] *Id.* The ACRS discussed Exelon's commitment in its letter dated February 8, 2007, to Chairman Klein.

The Staff's assessment of the 3-D FEA consists of an evaluation of Exelon's Baseline Analysis, Exelon's Sensitivity Analysis, an assessment of the Exelon's Conclusions, and the Staff's Conclusion.

Prior Analyses of the Drywell Shell

Three independent analyses of Oyster Creek's drywell shell have been completed by GE, Sandia, and Structural Integrity Associates ("SIA"), who performed the 3-D FEA. In addition, the 3-D FEA performed by SIA was independently reviewed by the Staff and Becht Nuclear Services ("Becht"), a contractor hired by the State of New Jersey.[5]

A. Analysis of Record

In the GE analyses, internal grid measurements of the drywell shell were compared against two criteria:[6] (a) a mean thicknesses greater than 736 mils in each bay, and (b) the lowest thickness in an isolated area[7] greater than 490 mils. For any external ultrasonic testing ("UT") measurements that were found to be less than 736 mils (also called the locally-thinned areas), the required thickness was 536 mils, in an area not to exceed 1 square foot. The actual measured UT thicknesses were all greater than the allowable criteria (i.e., mean thickness > 736 mils; locally thinned areas > 536 mils; pressure criterion > 490 mils). GE's analysis was intended to determine the limiting thickness acceptance criteria. In the GE analysis, the locally-thinned areas were conservatively located between the centerlines of the vents, which results in early drywell shell buckling modes. The conservatively modeled GE analysis, in general, met the safety factor of 2.0 against buckling under the most limiting load combination containing the refueling water load. The GE analysis contained many conservative assumptions, such as the location of the locally-thinned areas being more conservative than the actual location of the locally-thinned areas (near the vent lines), where considerably thicker reinforcing plates would preclude certain early buckling modes.

B. Sandia Analysis

The Staff-sponsored Sandia analysis (Reference 3) conservatively considered the average of the external UT measurements in each sandbed bay as the bay thickness used in the evaluation input. The locally-thinned areas, modeled as 30 inch by 18 inch rectangles in the two most degraded bays, were assigned the lowest measured thicknesses in those bays. Under the limiting load combination containing the refueling water load, the safety factor against buckling was calculated as 2.15. The Sandia analysis also confirmed that the Oyster Creek drywell shell meets the ASME code margins under all the postulated conditions.

II. Baseline Analysis and Three Dimensional Finite Element Model

The 3-D FEA, performed by SIA, is characterized as a conservatively-biased realistic analysis of the current condition of the drywell shell and evaluated the effects of the measurement uncertainties. The drywell shell's thickness was divided into two major sections based on

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[5] Becht's review is available at ADAMS Accession No. ML091040727.

[6] The GE analysis developed two acceptance criteria for drywell thickness requirements. The two criteria account for overpressure and buckling failure modes.

[7] The isolated area is defined to be a 2 inch diameter circle or smaller.

elevation. The degraded areas identified by the internal grid UT measurements are separated into measurements above an elevation of 11 feet 0 inches and below this elevation (see Figure 2). In general, the measured thicknesses above 11 feet 0 inches are higher than or equal to those below this elevation, except for the thicknesses in Bay 7. The Bay 7 thicknesses are close to the nominal thickness (1154 mils) of the drywell shell in the sandbed area because the bay experienced very little degradation.

The model representing the degradation also considered any locally-thinned areas identified during the 1992 inspection, prior to the application of the epoxy coating to the exterior of the sandbed region. During subsequent inspections, the licensee confirmed that these locally-thinned areas have not experienced significant additional corrosion. These areas were modeled in the 3-D FEA as locally-thinned circular areas (instead of the square areas used in GE analysis) to facilitate more modern computer modeling. These circular areas in the 3-D FEA completely circumscribed the square areas of the GE analysis. Therefore, this modeling results in larger thinned areas than analyzed by the GE analysis. The locally-thinned areas are placed at the locations where they were actually identified, resulting in more realistic failure modes than those characterized by GE.

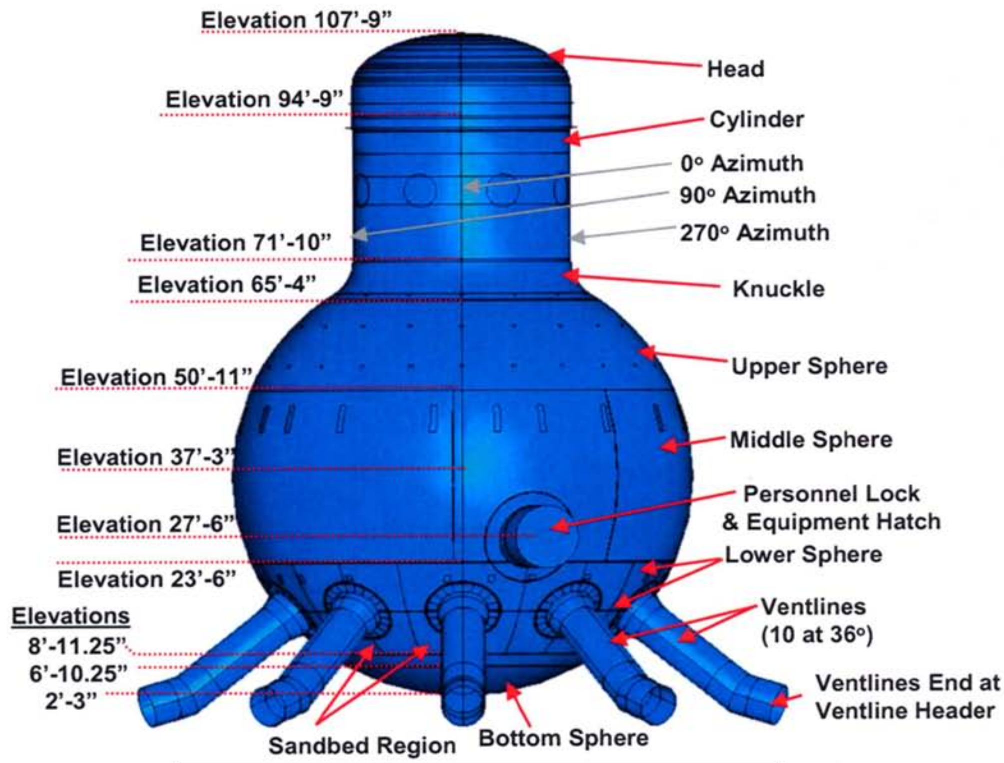

FIGURE 1 Oyster Creek Drywell Shell

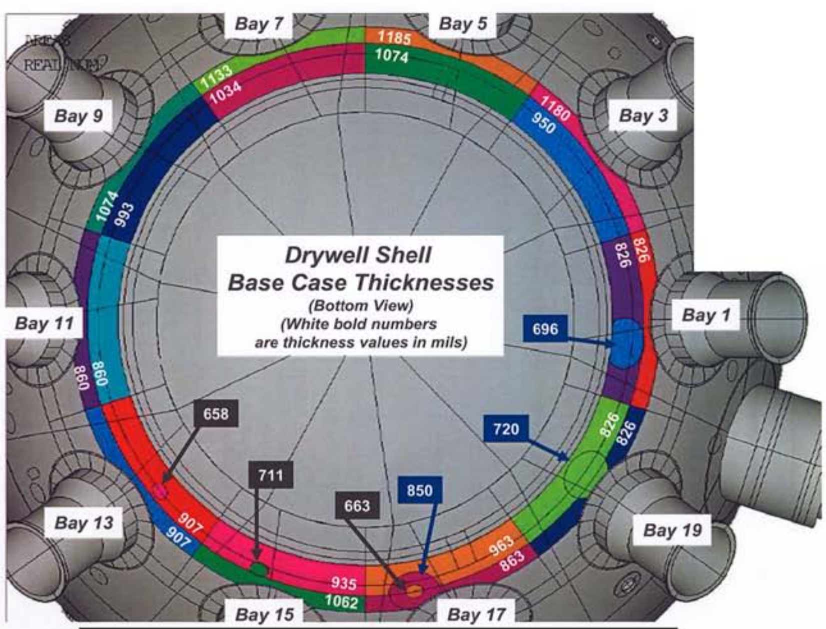

FIGURE 2 Schematic - Sand Bed Area

Table 1, which reproduces selected information from Table 3-4 of Reference 4, provides the diameters and thicknesses of the identified locally-thinned areas. This data was used to determine the modeled drywell shell thicknesses based on the internal grid UT measurements, and the external UT measurements.

Table 1: Local Thinned Area Modeling

Sandbed Region	Diameter (inches)	Thickness (mils)
Bay 1	51	696
Bay 13	18	658
Bay 15	18	711
Bay 17	18 (1) and 51(2)	663 (1) and 850 (2)
Bay 19	51	720

Notes: (1) Inner circle of locally thinned area.
(2) Outer ring of locally thinned area.

The assigned thicknesses in all bays in the sandbed area correspond to the UT measurements taken during the 2006 inspection.[8] The applicant confirmed that the thicknesses found during the 2008 inspection are consistent with those found during the 2006 inspection. The staff finds Exelon's modeling of the corroded areas in the sandbed region appropriate for a realistic analysis of the available margin to the ASME code limits.

A. Finite Element Model – Mesh Optimization

SIA used the public domain computer code ANSYS, Release 11.0, for performing the 3-D FEA. The entire geometry of the drywell shell is mapped using areas in three dimensional space at the mid-thickness location of the shell parts including the cylindrical, upper spherical, lower spherical, and the sandbed areas. Real, type, and material numbering in ANSYS were used to distinguish the different components. Real numbers are for the thicknesses, type numbers are for the assemblies (e.g. star truss assemblies), and material numbers are for the different materials associated with the drywell shell assembly.

The 3-D FEA incorporated all penetrations greater than 3 inches in diameter, the insert plates, the reinforcing plates, the star-truss assembly that stabilizes the cylindrical portion of the shell during an earthquake event, the vent lines including the vent headers and downcomers, the constraint imposed by the concrete floors, trenches, and the components that would significantly affect the analysis' results. The model also incorporates the general element size of 3 inches in the spherical and cylindrical regions of the drywell, 1.5 inches in the sandbed region, and 0.75 inches in the locally-thinned areas. The 3-D FEA contains approximately 400,000 elements.

SIA performed mesh sensitivity analyses using 2-inch elements in the spherical and cylindrical regions, 0.75-inch elements in the sandbed region, and 0.5 inch elements in the locally corroded area, constituting an additional 100,000 elements. The sensitivity analysis was performed using the load combination containing the 44 psi internal pressure. The stress comparison indicated that there was no significant difference in the results with the finer mesh element size.

[8] *See* Figure 2.

Based on the staff's review of the geometry description of the drywell shell, representation of various components in the model, and the results of the mesh size optimization process, the staff finds that the essential parameters of the drywell shell are appropriately and adequately modeled in the 3-D FEA.

B. Model Boundary Conditions

In the Baseline Analysis, SIA discussed the boundary conditions for three cases: (a) the bottom head thermal and structural boundary condition, (b) the star truss boundary condition in the drywell shell cylindrical region, and (c) the vent-header boundary condition.

The boundary condition for the bottom head of the spherical portion of the drywell shell is treated in two parts. In the first part, SIA considered the spherical drywell shell below the bottom of the sandbed region at an elevation[9] of 8 feet 11½ inches to the bottom of the sphere at an elevation of 2 feet 3 inches, which is embedded in concrete, as fixed in the radial direction. However, this constraint allows for movement in the meridional direction, assuming no bond exists between the drywell shell steel and the concrete. In the second part, for the spherical shell above the bottom of the sandbed region to the top of the inside concrete at an elevation of 10 feet 3 inches, SIA assumed the inside concrete does not restrain drywell shell movement in any direction.

There are 8 star truss assemblies (also called stabilizers) spaced evenly around the circumference in the cylindrical portion of the drywell shell, at an elevation of 82 feet 2 inches. Each star truss has two truss members, which consist of double extra strong 10 inch pipe. The truss members connect the biological shield wall to the drywell shell star truss reinforcing plate and are welded onto the top plate of the shield wall. On the outside of the star truss reinforcing plate, a male lug connects the slot plate embedded into the concrete wall of the reactor building. The star truss components transmit the forces from the drywell shell internals (i.e. reactor vessel) to the reactor building without loading the drywell shell. The star truss assemblies only impose a net force in the circumferential direction of the drywell shell. This circumferential force is resisted by the concrete wall of the reactor building through the male lug on the outside of the star truss reinforcing plate.

In the 3-D FEA model, drywell shell displacement is constrained in the circumferential direction at the male lug of the star truss. This constraint is imposed to simulate the drywell shell's actual movement restraint when the male lug on the outside of the star truss comes into contact with the reactor building concrete wall within the slot. No similar constraint was imposed on the welded connections between the truss members and the shield wall.

The vent headers and downcomers are supported by vertical columns located at the bottom of the stiffener attached to the vent header for each end of the vent header segment. The 3-D FEA model incorporates this constraint as a boundary condition at the vent-line. The model constraint limits displacement in the vertical direction.

The Staff finds that SIA has realistically implemented appropriate boundary conditions for the concrete in the spherical shell, the star truss assemblies in the cylindrical shell, and the supports for the vent-line headers and downcomers. The boundary constraints represent the physical geometry of the shell components, and are in accordance with good engineering practice.

[9] Elevations on the drywell shell are shown in Figure 1, above.

C. Loading Input

The primary loads considered in the analysis include: dead loads (including the water load during refueling), live loads, loss of coolant accident loads (internal pressure and associated temperature), pipe reaction loads, and earthquake loads (Operating Basis Earthquake ("OBE") and Safe Shutdown Earthquake ("SSE")) including anchor movements where applicable, external pressure, and post-accident flooding loads.

The original loading conditions and load combinations were defined in the Oyster Creek Final Safety Analysis Report (FSAR), upon granting the full power license to the plant. Since that time, there have been NRC approved updates and modifications to the original FSAR. The modeled loading conditions and combinations utilize the modified and updated information as reflected in the current FSAR. Table 2 shows the loads and load combinations utilized in the 3-D FEA. The loads and load combinations analyzed using the modified loads are shown in Table 7-6 of 403R, and are reproduced in Table 2 of this document.

Table 2: Load Combinations[10]

Load Combination	ASME Level	Condition	Load Cases
LC1	A	Design/Test	Prs1 + Grvty + OBE + SAM
LC2	A	Design/Test	Prs1 + Grvty – OBE – SAM
LC3	A/B	Normal	Prs2 + Grvty + Mch/Live + OBE + SAM + EPOBE + EPThrm + Thrm2
LC4	A/B	Normal	Prs2 + Grvty + Mch/Live – OBE – SAM – EPOBE + EPThrm + Thrm2
LC5	A/B	Refueling	Prs2 + Grvty + Mch/Live + OBE + SAM + EPOBE + EPThrm + Thrm2 + Refuel
LC6	A/B	Refueling	Prs2 + Grvty + Mch/Live – OBE – SAM – EPOBE + EPThrm + Thrm2 + Refuel
LC9	C	Post-Accident	Prs2 + Grvty + Mch/Live + SSE + SAM(SSE) + EPSSE + Flood
LC10	C	Post-Accident	Prs2 + Grvty + Mch/Live – SSE – SAM(SSE)– EPSSE + Flood

Notes:
SAM – Seismic Anchor Movement
Prs1 – Internal Pressure
Prs2 – External Pressure
Thrm2 – Operating Temperature
EP – External Piping
OBE – Operating Basis Earthquake
SSE – Safe Shutdown Earthquake

10 This information in this table is also available at Table 7-6 of Reference 4.

The Staff finds that the loads and load combinations modeled in the 3-D FEA are consistent with those in the current FSAR. The staff also finds that the ASME Service Levels assigned to the load combinations are in accordance with the prevailing practice, except for the use of Service Level D for LC9 and LC10. Since SIA applied a more restrictive standard, Level C for LC9 and LC10, the Staff finds this modeling to be conservative and consistent with good engineering practice.

D. Stress Evaluation

The original Code of Record for the design and construction of the Oyster Creek drywell shell was the ASME Boiler and Pressure Vessel (B&PV) Code, Section VIII, 1962 Edition, with Nuclear Case Interpretations 1270 N-5, 1274 N-5 and 1272 N-5. The original Code of Record and Nuclear Case Interpretations do not provide stress intensity limits at different operating conditions (i.e. post-accident conditions) or limits for the local membrane stress due to the thickness reductions from local to general corrosion effects.

To address the need for guidance for the above areas, the code stress intensity requirements and limits from the ASME B&PV Code, Section III, Subsection NE, Class MC Components 1989 Edition including 1991 Winter Addenda were used. This ASME B&PV Code, Section III was used in previous re-evaluations of the Oyster Creek drywell stress analyses.[11] The stress intensity limits, as specified in Subarticle NE-3320, must also be satisfied.

The applicable allowable stress intensity, S_{mc}, for the Oyster Creek drywell materials are presented in Table 3, below. If the allowable stress intensity was not available for a material, the allowable value was appropriately determined by SIA based on the rules provided in Appendix III of the ASME B&PV Code, Section III. SIA also explained the use additional Subsection NE provisions as follows:

(a) Per NE-3213.9, the bending stress at a gross structural discontinuity is classified as secondary stress. This requirement includes bending stresses due to internal pressure and external loads or moments per Table NE-3217-1, and for stresses near the nozzle, other openings, junctions to heads or flanges, knuckles or junctions to the shell.

(b) Per NE-3213.10, a stress region may be considered local, if the distance over which the membrane stress intensity exceeds 1.1 S_{mc}, does not extend in the meridional direction more than $\sqrt{(Rt)}$, where R is the minimum mid-surface radius of curvature and t is the minimum thickness in the region considered.

(c) Per NE-3221.3, the primary general or local membrane plus primary bend stress intensity is derived from the highest value across the thickness of a section. In addition, it is stated in Note (2) of Figures NE-3221-1 and NE3221-3 that the provisions apply to the solid rectangular section.[12]

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[11] *See* References 2 and 3.

[12] Reference 4.

Among the load combinations presented in Table 2, the refueling cases (LC5 and LC6) are the limiting cases in Levels A and B service conditions, because of the additional water load during refueling. For Level C, the post-accident flooding cases (LC9 and LC10) are the limiting cases. These load cases were selected as the bounding cases for stress evaluation.

The materials used in the Oyster Creek drywell shell construction and the ASME Code allowable stress intensities are provided in Table 7-2 of Enclosure 1[13] and are reproduced in Table 3 of this document.

Table 3: Basic Code Allowable Stress Intensities[14]

Material	Allowable Stress Intensities, S_{mc} (ksi)		
	100 °F	200 °F	300 °F
SA-516, Grade 70	19.3	19.3	19.3
SA-516, Grade 60	16.5	16.5	16.5
SA-320, L7(4)	27.5	27.5	27.5
USS-T1(1) (SA-514, Grade F)	27.5	27.5	27.5
SA-333, Grade O(2)	15.1	15.1	15.1
SA-333, Grade 1	15.1	15.1	15.1
SA-312, Type 304	18.8	17.8	16.6
SA-276, Type 304(1, 3)	18.8	17.8	16.6

Notes: (1) Stress intensity allowable is based on Appendix III, Article III-3000, using tensile strength of 110 ksi at room temperature.
(2) Stress intensity allowable not available, uses the allowable of SA-333, Grade 1.
(3) Use the allowable of SA-312, Type 304.
(4) Use the allowable of SA-193 Grade B7.

For each of the ASME Service Levels, SIA has determined the allowable limits as a function of the basic allowable stress intensities shown in Table 3. SIA points out that for LC9 and LC10 (Post-Accident Flooding and SSE) the current practice permits the use of Service Level D allowable limits, which are significantly higher than the Applied Service Level C allowable limits. However, Oyster Creek's FSAR utilizes Service Level C, for these load combinations. SIA used Service Level C allowable limits for the 3-D FEA. The stresses generated in the drywell shell components from the analysis, with respect to the ASME III, Subsection NE allowable limits demonstrated that:

1) Under load combinations LC5 and LC6, incorporating the refueling load, the stresses in the cylindrical shell, stiffeners and in the star truss assemblies are below service Level B allowable limits,
2) Under LC5, the stresses in the sandbed bays and the vent pipes are comparatively low (i.e., <35% of the service Level B allowable limits), and

[13] Reference 4.

[14] *See also* Table 7-2 of Reference 4.

3) Under load combinations LC9 and LC10 and incorporating the post-accident flooding load and SSE, the stresses in the cylindrical and sandbed region are less than 80% of the Level C allowable limits.

SIA also performed a fatigue evaluation of the drywell shell material for the six stipulated conditions in NE-3221.5(d) of the ASME Code. The evaluation was based on the following assumptions, which have been and are being monitored under the current licensing basis:

1) The average temperature of the drywell is 150º F,
2) The number of significant pressure-temperature fluctuations is 200,
3) The startup-shutdown cycles are likely to be no more than one per year,
4) The leak test cycle is one every 10 years, and
5) An average (rather than instantaneous) coefficient of thermal expansion of the materials is used.

These assumptions fairly represent the plant characteristics, are appropriate for performing the fatigue evaluation, and utilize good engineering practice. The evaluation considered assumed cycles for each drywell shell material for the six stipulated conditions and showed that a detailed fatigue analysis is not required.

The staff finds that the stresses generated from the analysis are below the allowable limits for the critical load combinations, the required ASME code fatigue evaluation, and show no adverse effects on the drywell shell components.

E. <u>Buckling Evaluation</u>

The basic compressive allowable stress values, referred to in ASME Code Section III NE-3222.1, correspond to a factor of safety of 2 in Code Case N-284-1. This factor is applied to the buckling stress values that are determined by classical analysis and reduced by an appropriate capacity reduction factor ("CRF").[15] The CRF accounts for (1) the effects of imperfections and non-linearity in geometry and boundary conditions and (2) the plasticity reduction factors which account for nonlinearity in material properties. The stability stress limits in NE-3222.2 correspond to the following factors of safety (FS):

1) FS = 2.0 for the Level A/B Service Limits would apply to LC-3 to LC-6,
2) FS = 1.67 for Service Limit C would apply to LC-9 and LC-10, and
3) FS = 1.34 for Service Limit D – Not applicable for Oyster Creek.

The buckling evaluation of the drywell shell components subject to meridional compressive stresses requires a geometric description as well as the basic material properties of the components. The buckling evaluation consists of a determination of: (1) load factors from the buckling mode shapes, (2) CRF, modified by the hoop tensile stresses (if applicable), (3) plasticity reduction factor (if applicable), and (4) calculation of safety factors as the theoretical buckling stress divided by the meridional compressive stress.

[15] The selection of use of the CRF is discussed below in a separate section.

1. Capacity Reduction Factor

For the load combination consisting of refueling load and external pressure, the Sandia analysis did not give credit to the hoop tensile stresses generated due to gravity load, and utilized unmodified capacity reduction factor ("CRF") given in the ASME Code Case N-284. The GE analysis, which is the analysis of record, and the SIA 3-D FEA gave credit to the tensile stresses generated by the refueling load. GE and SIA modified the CRF accordingly. As such, the CRF has been the subject of considerable interest and differing views amongst the parties to the license renewal hearing process. The Staff notes that for the SIA analysis, the refueling loading case was not the limiting case for buckling, but the post-accident flooding which used the same CRF as the GE analysis and Sandia analysis.

The analysis of record for Oyster Creek's current licensing basis was performed by GE in 1992. GE used extremely conservative inputs in accordance with the modeling capabilities that were in use at that time. As discussed during the license renewal hearings, GE made very conservative assumptions regarding the physical state of the sandbed region. GE uniformly thinned the drywell shell in the sandbed region in its analysis even though large areas of the sandbed region experienced little corrosion.

In GE's analysis, a modified CRF was applied to the drywell shell based on tests on metal spheres[16] performed by Dr. Clarence Miller. Code Case N-284, which was used for this analysis, allows higher values of CRF when appropriate justification is provided.

In 10 CFR 50.55a, the regulations state that nuclear power plants must meet the requirements of the ASME Boiler and Pressure Vessel Code, which is formulated and approved through a consensus process with experts. The NRC endorses the code and its code cases through the rulemaking process, or through approving its use for certain applications. In this case, the Staff evaluated this approach in its review of the GE analysis in 1994. In conjunction with the 1994 review, the Staff had structural engineering experts at Brookhaven National Laboratory ("BNL") review Oyster Creek's use of the modified CRF. Considering the input from the technical evaluation from BNL,[17] which explicitly discussed the use of the modified CRF for Oyster Creek's drywell shell, the staff found the modification to the CRF acceptable.[18]

a) Advisory Committee on Reactor Safeguards

At the February 8, 2007, meeting of the Advisory Committee on Reactor Safeguards ("ACRS") on the Oyster Creek license renewal application, the technical basis for the Code Case and test results were presented regarding the modified CRF use for analyzing Oyster Creek's drywell shell. In its February 8, 2007, letter to the Staff, the ACRS concurred with the Staff's position that found the modified CRF acceptable.

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[16] The sandbed bay has the same geometry as a sphere.

[17] The BNL evaluation of the modified capacity reduction factor is available at "Evaluation Report on the Structural Integrity of the Oyster Creek Drywell," dated April 24, 1992.

[18] *Id.*

b) Sandia Analysis

During the staff's review of Oyster Creek's license renewal application, the staff's structural engineering expert asked Sandia to perform a confirmatory analysis of the drywell shell using a more modern finite element analysis approach. Sandia made significant conservative assumptions and concluded that the drywell shell met ASME code margins even without using a modified CRF. The Staff notified the ACRS in a March 8, 2007, memorandum that Sandia did not use a modified CRF because they did not have the results from Dr. Miller's tests available to them.[19] Therefore, Sandia was unable to justify the modified CRF without confirming Dr. Miller's tests. Because Sandia's analysis found that the drywell shell met ASME Code margins without modifying the CRF, the Staff determined that the application of the modified CRF would only increase the available margin already shown to exist.

c) SIA 3-D FEA

Exelon committed during the February 8, 2007, ACRS meeting to perform a 3-D finite element analysis of the Oyster Creek drywell, to better quantify the existing margin above the minimum required ASME code factor of safety in the as-found drywell shell using more modern methods. Exelon submitted a summary report of the analysis, performed by SIA, to the NRC Staff on January 22, 2009, which is the subject of this report. This analysis used the modification to the CRF supported by Dr. Miller's tests.

d) Becht Review

The State of New Jersey ("New Jersey") engaged a contractor, Becht Nuclear Services ("Becht"), to perform an independent review of the SIA analysis of the drywell shell. Becht's review of the 3-D FEA concentrated mainly on the SIA analysis and the Sandia analysis. Becht concluded that "[t]he Code requirements are satisfied for the drywell in its current (2006-2008) state of degradation …."[20] Becht also concluded that "the required code buckling factor of safety (FS) is acceptable without use of Miller's modified capacity reduction factor."[21] Finally, Becht "believes that the uncertainty associated with the wall thinning measurements has been treated adequately for the measurements provided to-date, and as evaluated in two sensitivity cases in SIA's [analysis]."[22]

Although Becht concluded that the use of the modified CRF negatively impacted how conservative the 3-D FEA was when taken alone, SIA's conservative treatment of the "locally high theoretical buckling stress would offset any negative conservatism arising from the modified CRF." Becht also states that none of the negative conservatisms are singly or in-combination significant enough to warrant further action. These conservatisms are of the types covered by margins included in the ASME Codes and Standards.

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[19] Memo from P. Kuo, Director of DLR/NRR to F. Gillespie, ACRS, "Re:ACRS Review of Oyster Creek SER" (Mar. 8, 2007) (ADAMS Accession No. ML070650376).

[20] Becht Report at 1.

[21] *Id.*

[22] *Id.*

e) Staff Conclusion

In reviewing the 3-D FEA, other analyses of the drywell shell, and independent reviews of the analyses, the Staff notes that numerous analytical approaches would accurately and adequately account for stresses present in any structure when utilized consistent with the ASME code by a well-qualified and experienced structural engineer. As such, every decision requires technical judgment and balancing on the part of the structural engineer. For example, SIA imposed conservative local stresses and a “less conservative” CRF, where Becht’s preferred methodology would have imposed less conservative local stresses and a more conservative CRF. As Becht concluded above, the ASME code factor of safety is met regardless of the CRF.

SIA’s use of the modified CRF is accepted by structural engineering experts in the ASME code community, who have formally endorsed this approach to modifying capacity reduction factors in a recently approved Code case.[23] Article 1500 of Code Case N-284-1 allows for the CRF to be increased for tensile stresses resulting from internal pressures. However, the Code Case does not address the effects of tensile stresses that result from other loadings (e.g., gravity loads). Based on the research performed by Dr. Miller, the applicant used the modified CRF for other loads resulting in tensile stresses.[24] The staff found the use of the modified CRF acceptable for use.

The methodology utilized by the SIA and as described by Dr. Miller for modification of the CRF was formally published in ASME Code Case N-759 (2006), “Alternative Rules for Determining Allowable External Pressure and Compressive Stresses for Cylinders, Cones, Spheres, and Formed Heads.” The staff is in the process of reviewing the Code case for endorsement in Revision 35 of Regulatory Guide 1.84 for, “Design, Fabrication, and Materials Code Case Acceptability, ASME Section III.”

When this methodology is applied to a shell with a non-uniform wall thickness, as in the case of Oyster Creek drywell shell, where the calculated stresses in both the hoop (tensile) and meridional (compressive) directions are not uniform among different elements, certain approximations have to be made to compute the magnitude of the modified CRF. The staff recognizes the complexity involved in developing a reasonable estimate of the CRF due to the variation in the ratio of tensile hoop stresses to compressive meridional stresses among different elements from the finite element model. The Staff finds the approach used by SIA in developing the modified CRF to be realistic and reasonable for this purpose. The Staff also notes that even if SIA had used an unmodified CRF, the analysis indicates that Oyster Creek’s drywell shell exceeds the minimum safety factors as prescribed by the ASME code. Thus, the Staff recognizes that the modification of the capacity reduction factor requires expert structural engineering judgment due to the complexity of the analysis, but does not agree that its use leads to a non-conservative analysis for the reasons discussed above.

2. Buckling Modes and Safety Factors

The controlling load cases considered for the buckling analysis are (1) the refueling load case and (2) the post-accident flooding load case. SIA performed bifurcation buckling analysis for each load case, and generated 200 eigenvalue buckling modes with corresponding load factors.

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[23] ASME Code Case N-759 (2006), “Alternative Rules for Determining Allowable External Pressure and Compressive Stresses for Cylinders, Cones, Spheres, and Formed Heads.”

[24] Reference 2 at Section 2.

After a review of the buckling modes, corresponding load factors, and displacements, the following buckling modes governed each section of the drywell shell.

1) For the refueling load case, the first mode governs the cylindrical portion of the drywell, while the postulated buckling of the upper, middle and lower spherical region of the drywell occurs in the 15th, 25th, and 32nd modes. The postulated buckling of the sandbed region occurs at modes higher than the 45th mode with a relatively high load factor of 11.584 (high load factor means high factor of safety).

2) For the post-accident flooding case, the postulated buckling in the cylindrical region occurs at the 56th mode and that of the upper and middle spherical region occurs at the 200th mode. The postulated buckling of the lower spherical region occurs at the 15th mode with a load factor of 7.344. In the sandbed region, a very small amount of displacement (0.01 inch) occurs in the 11th mode in Bay 1. SIA has used the 11th mode load factor of 7.162 for calculating the buckling safety factor calculations.

A summary of the pertinent safety factors for each major component of the drywell are given in Table 4 and Table 5. The staff finds the process used in arriving at the load factors, theoretical buckling loads, and safety factors consistent with good engineering practice and ASME Section III requirements, as supplemented by Code Case N-284-1.

Table 4: Buckling Evaluation[25]

Loading ↓	Region →	Cylindrical	Spherical		
			Upper	Middle	Lower
Refueling	SF – Safety Factor Min. Reqd – 2.00	3.39	4.27	3.60	3.60
Flooding	SF – Safety Factor Min. Reqd –1.67	3.46	7.57	4.76	2.22

[25] Reference 4.

Table 5: Buckling Evaluation, Sandbed Region[26]

Loading	**Sandbed Region**	**Bays**									
		1	3	5	7	9	11	13	15	17	19
Refueling	Safety Factor Min. Reqd – 2.00	3.70	3.54	3.56	3.57	3.56	3.69	3.78	3.69	3.67	3.74
Flooding	Safety Factor Min. Reqd – 1.67	2.08	2.36	2.25	2.24	2.23	2.12	2.13	2.29	2.14	2.02

[26] *Id.*

F. Other Conservative Assumptions

SIA has listed additional conservative assumptions and approaches included in its Baseline Analysis, as follows:

1) Seismic Response Spectra: The drywell is supported at two locations, at the 10 feet 3 inches base, and at the 82 feet 9 inches star truss location. The seismic spectra at the 82 feet 9 inches star truss support location are significantly higher than the spectra at the 10 feet 3 inches base. The bounding seismic response spectra are used as the seismic input for the drywell. This introduces significant conservatism in the seismic response of the structure.

2) Post Accident Flooding Case Contributing Water Mass: All of the water is included as added mass in the drywell in the seismic analysis. In a seismic event, a substantial portion of the water will be acting on the bioshield wall, which has been conservatively neglected in the Baseline Analysis. The assumption that the entire seismic energy of the water is absorbed by the drywell is conservative.

3) Welded Steel Structure Damping Ratio: As implemented by the SIA analysis, NRC Regulatory Guide 1.61 (issued in October 1973) recommends a damping value of 2% to be used with the Operating Basis Earthquake analysis for welded steel structures. Revision 1 of Regulatory Guide 1.61 (issued in March 2007) recommends a 3 percent damping value for welded structures. The use of an increased damping value will reduce the component stresses in the refueling case, in which the Operating Basis Earthquake is included as one of the loads.

4) Service Level C Limits versus Service Level D Limits: The Post Accident Flooding Case is evaluated using the Service Level C Limits, which are significantly lower than the Service Level D Limits. The Post Accident Flooding condition is a highly unlikely event and is generally categorized as a Service Level D event. Using the Service Level D Limits would have yielded additional margins on the analysis results for the Post Accident Flooding Load Case.

5) Support Provided by Star Truss/Bioshield Wall: There are a total of 8 star trusses connecting the drywell shell to the bioshield wall at an elevation of 82 feet 9 inches. The structural support provided by the Star Truss/Bioshield Wall has been conservatively neglected from the analyses. The structural support will take on some load, which is a conservatism in the analyses, especially in the buckling analysis.

6) Sizes of Locally Thinned Areas: The sizes of the locally thinned areas in the sandbed region have been conservatively mapped. An extended circular boundary has been drawn to enclose the postulated locally thinned areas. These conservatisms result in a larger, thinner area being modeled in the analysis, which produces conservative analysis in the sandbed region.

III. Sensitivity Analysis

Two Sensitivity Cases were considered as part of 3-D analysis of the Oyster Creek drywell shell. The summary analysis is provided in Reference 4. The two cases considered are:

> **Case 1 ("S1"):** Reduce the wall thickness of the defined locally thinned area in Bay 1 from 696 mils (see Figure 2) to 596 mils, keeping the general area of the thickness in Bay 1 constant (826 mils - as in 403R) with all other inputs the same as in the Base Case.
>
> **Case 2 ("S2"):** Reduce the wall thickness of the general area in Bay 19 from 826 mils (see Figure 2) to 776 mils, keeping the thickness of the locally thinned area constant (720 mils) and all other inputs the same as in the Base Case.

To capture the potential uncertainties in identifying the location and degradation of the locally thinned areas, the applicant has reduced the thickness of the 51-inch diameter circle (see Figure 2) by 100 mils in Bay 1, the bay most affected by corrosion. As the UT measurements at certain elevations are not available in certain bays, in addition to the approach taken by the applicant in the Baseline analysis, the applicant has reduced the shell thickness by 50 mils in Bay 19 in S2. This will quantify of the effects of thickness reduction in the sandbed region on the buckling safety factors. The staff finds the use of these sensitivity cases conservative and acceptable for considering the effect of the size and thickness of degraded areas on the available margin.

The stress analyses and buckling evaluation for the two sensitivity cases follow the same methodology used in the baseline analysis. The load combinations and ASME Code evaluation are performed in the same manner as those in the baseline analysis. The loading input, analysis procedures, 3-D FEA model, and the acceptance criteria for the sensitivity cases are the same as those used in the baseline analysis. The reduced wall thicknesses in a locally thinned area in Bay 1 and reduced thicknesses in Bay 19 are the only variations from the baseline analysis. The buckling results from these two sensitivity cases are compared to the results obtained in the baseline analysis.[27]

As the regions above the sandbed area are not affected in S1 and S2, the stresses in these regions are essentially the same as those in the baseline analysis. A summary of the stress intensity results from the Baseline analysis, S1, and S2 is provided in Reference 4. The analysis shows less than an eight percent increase in the stress intensities in Bays 1 and 19, for S1 and S2. These nominal increases remain well below the ASME code limits.

Using the same methodology as in the Baseline analysis, SIA developed the load factors and displacements for S1 and S2 for the controlling load combinations with the refueling load and post-accident flooding load. The affected region remains the sandbed region. A review of the

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[27] SIA's sensitivity analyses adequately vary the size of the thinned area and the depth of the thinned areas such that reasonable conclusions can be drawn regarding these effects and are adequately analyzed for these effects on the drywell shell. The three analyses performed by GE, Sandia, and SIA modeled the drywell shell degradation in different geometries. Based on the variation in analysis techniques, reasonable conclusions can be drawn about the effect of variations similar to performing additional sensitivity analyses.

safety factors in the sandbed region indicates that the safety factor in Bay 19 is decreased by about 9 percent (from 3.74 to 3.40) in S1, under the refueling load. In the post-accident flooding load case, the safety factors for S1 and S2 show minimal variations to the baseline analysis. In the load cases considered, the safety factors always exceed the ASME allowable minimums.

IV. Overall Staff Assessment

The staff's assessment indicates that the 3-D FEA of the Oyster Creek drywell shell has been performed using modern methods of analysis, using the actual geometry of the shell, with appropriate boundary conditions. The applicant's consultant, SIA, utilized UT thickness measurements taken during the 2006 outage. In the cover letter accompanying the Summary Report, dated January 22, 2009, the applicant confirms that the UT measurements taken during the 2008 outage correspond to the measurements taken during the 2006 outage. In Section 3.11 of the staff's Inspection Report (Reference 6), the staff noted that the technical evaluation also compared the 2008 data values to the corresponding values recorded by the 2006 UT examinations in the same locations and concluded that there were no significant differences in measured thicknesses and no observable widespread corrosion.

The applicant's analysis is performed for the two controlling cases (i.e., the refueling load case and the post-accident flooding load case). These cases have different ASME allowable safety factors (2.0 for the refueling load case and 1.67 for the post-accident flooding load case). Table 6, below, reflects the normalized safety factors.[28] A normalized safety factor greater than 1 shows that excess margin exists to the ASME code limits. In the table, only Bays 1 and 19 of the sandbed region, the most affected bays, are included. In the refueling load case, the effects of the sensitivity cases are not appreciable. The lowest calculated buckling margins in the sandbed region are controlled by the post-accident flooding load.

Table 6: Comparison of Normalized Safety Factors[29]

LOAD	REGION	BASELINE	S1	S2
Refueling	Cylindrical Shell	1.65	1.67	1.67
	Upper Spherical Shell	2.13	2.13	2.13
	Middle Spherical Shell	1.80	1.80	1.80
	Lower Spherical Shell	1.80	1.69	1.80
	Sandbed Bay 1	1.85	1.74	1.81
	Sandbed Bay 19	1.87	1.70	1.84
Flooding	Cylindrical Shell	2.07	2.07	2.07
	Upper Spherical Shell	4.53	4.54	4.53
	Middle Spherical Shell	2.85	2.84	2.84
	Lower Spherical Shell	1.33	1.31	1.31
	Sandbed Bay 1	1.25	1.26	1.23
	Sandbed Bay 19	1.21	1.20	1.19

[28] $SF_n = SF_{actual}/SF_{allowable}$

[29] The normalized safety factors shown in the table are used to ease comparison to the ASME code criteria for the various components of the drywell shell. Any number in excess of 1.0 shows available margin to the ASME code criteria.

The following observations can be made from the Table:

1) For all the cases, the normalized safety factors are well above 1.0 (a margin of 1.0 would indicate that the factor of safety is at the ASME Code allowable limit).

2) For the flooding load case, the lowest normalized safety factor of 1.19 is indicated in sandbed Bay 19.

3) For the refueling load case, the lowest normalized safety factor is indicated in the cylindrical shell.

4) The refueling load case affects the normalized safety factors in S1, but not in S2 in sandbed Bays 1 and 19.

5) Based on the comparison of the normalized safety factors in Table 6, it is apparent that no substantial difference exists in the normalized safety factors between the Base Case and Sensitivity Case 2, where the Bay 19 thickness was reduced by 50 mils. If the 50 mils reduction had been extended to the two adjacent Bays (i.e., Bay 19 and Bay 1), it is the staff's position that the normalized safety factors would have been, at a minimum, greater than 1.0.

Overall, the analysis demonstrated that all the components of the drywell shell show adequate margins against instability under refueling loads and post-accident flooding loads.

A review of the refueling load case stress evaluations performed for the Baseline analysis, S1, and S2, indicates that the stresses in the cylindrical shell, stiffeners/gussets and star truss stiffeners indicate stresses less than the allowable ASME code limits. It is the Staff's assessment that the sensitivity analyses and other analyses of the drywell shell adequately bound the intent of Judge Barrata's separate opinion [30] and demonstrate that additional sensitivity analyses will not result in significant new information. For the refueling load, the margin against buckling is the lowest in the cylindrical shell region, an area that has experienced little corrosion.

V. <u>Concluding Remarks</u>

The staff has performed an in depth review of Oyster Creek's 3-D FEA of the drywell shell. Based on that review, the Staff concludes that the 3-D FEA fulfilled Oyster Creek's commitment to provide a modern analysis that quantifies the available safety margin including performing sensitivity analyses. The 3-D FEA shows that Oyster Creek's drywell shell complies with the ASME code limits for the postulated conditions. This analysis was performed utilizing good engineering practices and judgment and applied conservatively-biased realistic assumptions. The sensitivity analyses performed by Oyster Creek also adequately bound the intent of Judge Barrata's separate opinion. Although this 3-D FEA is not the analysis of record for Oyster Creek's current licensing basis, it also demonstrates that Oyster Creek's drywell shell meets the ASME code limits even when accounting for measurement uncertainties in the thicknesses of the shell.

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[30] *See Amergen Energy Co., LLC* (Oyster Creek Nuclear Generating Station) LBP-07-17, 66 NRC 327,373-76 (2007)

VI. References

1. An ASME Section VIII Evaluation of Oyster Creek Drywell for Without Sand Case, Part I, Stress Analysis," GE Report, Index No. 9-3, Revision 0, DRF # 00664, February 1991, prepared by GE Nuclear Energy, San Jose, California.

2. An ASME Section VIII Evaluation of Oyster Creek Drywell for Without Sand Case, Part II, Stability Analysis, GE Report, Index No. 9-4, Revision 2, DRF # 00664, November 1992, prepared by GE Nuclear Energy, San Jose, California.

3. Structural Integrity Analysis of the Degraded Drywell Containment – Oyster Creek Nuclear Generating Station, Sandia National Laboratories, January 2007, ML070120395.

4. Exelon Nuclear, "Structural Evaluation of the Oyster Creek Drywell – Summary Report," prepared by Structural Integrity Associates, San Jose, California, January 2009, ML090270871, ML090270872, and ML090270873.

5. Miller, C.D., "Applicability of ASME Code Case N-284-1 to Buckling Analysis of Drywell Shell," June 2006.

6. Oyster Creek Generating Station – NRC License Renewal Follow-up Inspection Report 05000219/2008007, Dated January 21, 2009, ML090210106.

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

October 16, 2009

The Honorable Gregory B. Jaczko
Chairman
U. S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: REPORT ON THE 3-DIMENSIONAL FINITE ELEMENT ANALYSIS OF THE OYSTER CREEK NUCLEAR GENERATING STATION DRYWELL SHELL

Dear Chairman Jaczko:

During the 566th meeting of the Advisory Committee on Reactor Safeguards (ACRS), October 8 -10, 2009, we reviewed the 3-dimensional (3-D) finite element analysis (FEA) of the Oyster Creek Nuclear Generating Station (Oyster Creek) drywell shell and the associated assessment prepared by the NRC staff. Our Materials, Metallurgy, and Reactor Fuels Subcommittee also reviewed this matter during its meeting on September 23, 2009. During these reviews, we had the benefit of discussions with representatives of the NRC staff, Exelon Nuclear Generation Company (Exelon) and its contractors, and members of the public. We also had the benefit of the documents referenced.

CONCLUSION

The analysis presented by Exelon fulfills its commitment to provide a modern, realistic, 3-D FEA that better quantifies the available safety margin for the current drywell shell configuration of Oyster Creek. The analysis confirms that Oyster Creek's drywell shell complies with its current licensing basis for design basis accidents with margin. This analysis was performed using good engineering practices and judgment and used conservatively biased realistic assumptions.

BACKGROUND

During the 1980s, the licensee discovered corrosion on the outside wall of the Oyster Creek 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 it was partially supported by a sandbed. 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 sandbed. As part of the corrective actions, the licensee removed the sand and applied an epoxy coating to the outside of the shell in the sandbed region. In addition, repairs were made to the refueling cavity 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.

In 1992, the licensee performed ultrasonic testing to determine the as-found condition of the drywell shell and also performed a structural analysis to demonstrate acceptability of the containment in the degraded condition. The 1992 structural analysis was reviewed and approved by the NRC staff and remains the licensing basis analysis for the drywell shell. 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 1992 structural analysis was based on the assumption that the shell is uniformly thinned in the sandbed region to a thickness of 0.736 inch. The analysis showed that the shell met the allowable stress values for buckling per Section NE-3222 of the 1989 Edition of the ASME Code, Section III, Division 1, Subsection NE, Class MC Components. Since the average thickness of the shell is greater than 0.736 inch, the actual factor of safety exceeds the Code minimums. But, based on the licensing basis analysis, it is not possible to get a good estimate of the actual margins.

During our February 1, 2007 meeting regarding the license renewal application for Oyster Creek, Exelon committed to perform a 3-D FEA of the Oyster Creek drywell shell in the as-found degraded condition using more modern methods. The basic purpose of the analysis was to provide a more accurate quantification of the actual margins above the ASME Code required minimums.

In our February 8, 2007 report on the Oyster Creek license renewal application, we recommended that the staff add a license condition to ensure that the applicant fulfilled this commitment to perform a 3-D FEA of the Oyster Creek drywell shell prior to entering the period of extended operation, and requested a briefing on the results of the analysis when they became available. Consequently, this commitment was entered as a license condition in the staff's final Safety Evaluation Report. By letter dated January 22, 2009, Exelon submitted the results of the 3-D FEA of the Oyster Creek drywell shell.

DISCUSSION

The 3-D FEA submitted by Exelon to meet its commitment was performed by Structural Integrity Associates (SIA). SIA had access to proprietary design data for the drywell shell and were able to develop a very detailed structural model including all penetrations over 3 inches. Penetrations that are 3 inches or smaller are not specifically modeled. Instead, only their reinforcing plates or insert plates are modeled to account for the added stiffness of the plates. The vent pipes/header are modeled to account for the effect of their stiffness on the rest of the drywell. The base model includes approximately 406,000 shell elements ranging in size from 0.75 inch in locally thinned areas, 1.5 inches in the bulk of the sandbed region, 3.0 inches in most of the cylindrical and spherical shell, and up to 12.0 inches in the bottom spherical shell within the concrete. Mesh convergence was studied by considering meshes with up to 1,000,000 elements. Since the radius-to-thickness (R/t) ratio of the cylindrical and spherical shells ranges from 300-600, the use of shell elements, which assume a linear variation of the stress across the thickness is appropriate.

The SIA analysis has been reviewed by the staff, the ACRS and its consultants, Dr. Gery Wilkowski and Professor John Hutchinson, as well as the consultant for the State of New Jersey, Becht Nuclear Services. There is general agreement that this is a modern structural analysis performed utilizing good engineering practices and judgment. The primary sources of uncertainty are the characterization of the thickness of the sandbed region and the calculation of the capacity reduction factors, which account for the reduction in buckling loads of shells due to their sensitivity to deviations from perfect geometry.

Ultrasonic thickness measurements, which are the most accurate way to measure the remaining thickness of the shell, are available only for a small fraction of the sandbed region. Two types of measurements have been made, those based on 7 x 7 or 1 x 7 grids with 1-inch spacing between transducers, and individual ultrasonic thickness measurements at selected locations. Except for the grids in the trenches in Bays 5 and 17, all the grid measurements are made at Elevation 11' 3”. The grid locations were chosen as the thinnest locations at that elevation.

The locations for the individual ultrasonic thickness measurements were selected by visual examination to be the areas of greatest local thinning over a roughly 2.5 inch diameter region. The areas selected for examination were ground to ensure flat contact of the probe with the surface. Micrometer measurements showed that this grinding further reduced the local thickness by about 0.10 inch.

The licensee used visual inspection, judgment, the results from the grids at Elevation 11' 3", supplemented by the grids in the trenches in Bays 5 and 17 to estimate the average thicknesses in each of the Bays. The licensee used the results of the individual ultrasonic thickness measurements primarily to define thinned local regions.

The selection of the locations for the grid and the locations of the individual ultrasonic thickness measurements have been inspected and reviewed by the staff. They have been found to characterize the thickness conservatively for licensing basis analyses. The staff also finds Exelon’s modeling of the corroded areas in the sandbed region acceptable for a realistic analysis of the available margin to the ASME Code limits. We concur with the staff’s conclusion.

In the analysis performed by the Sandia National Laboratories (SNL) to support the review of the Oyster Creek license renewal, the individual ultrasonic thickness measurements were used to estimate the average thickness of the sandbed region. Such an approach is conservatively biased, since the thinnest regions were selected, and the thicknesses were further reduced by grinding. Not surprisingly, SNL obtained average thicknesses in the bays that are typically less than those estimated by the licensee. The average difference over all the bays is -0.068 inch. For the most severely corroded Bays 1 and 19, the licensee’s estimates are actually somewhat less than the SNL estimates. Based on the conservatisms inherent in using the individual ultrasonic thickness measurements to estimate remaining thickness, the SNL results support the conclusion that at least the average thickness of the sandbed region used by the licensee is appropriate for a realistic analysis of the margins in the drywell shell.

The licensee increased the size of the locally thinned zones in Bays 1, 13, 15, 17, and 19 compared to those used in the licensing basis analysis. The size and remaining thickness assigned to these local areas are conservative compared to the data, and the sizes are larger than those used by SNL.

A variety of load cases were studied by SIA. The limiting cases for buckling were the refueling case with the dead weight of the reactor cavity water and the post-accident case with seismic load and flooding. The minimum required safety factor in the refueling case for the current licensing basis is 2.0. The computed minimum value in the drywell occurs in the upper cylindrical shell and is 3.39. The minimum value in the sandbed region occurs in Bay 3 and is

3.54. The minimum required safety factor in the flooding case for the current licensing basis is 1.67. The computed minimum value is 2.02 in Bay 19.

In addition to the base case with the thicknesses selected as described above, the licensee also considered two sensitivity cases. In the first, the wall thickness of the 51-inch diameter locally thinned area in Bay 1 was reduced by an additional 0.10 inch, keeping the thickness in the un-thinned portion of Bay 1 constant. In the second, the thickness in the un-thinned portion of Bay 19 was reduced by 0.05 inch, keeping the locally thinned area in the bay constant.

In the first case, for refueling, the computed minimum safety factor occurs in the sandbed, Bay 3, rather than the upper cylinder and is 3.21 (versus 3.39). For flooding, the minimum safety factor still occurs in Bay 19 and is 1.98 (versus 2.02).

In all the solutions, although the thicknesses of the bays vary from 0.826 inch to 1.13 inches and some bays have locally thinned areas and others do not, the safety factor varies by less than +/- 8%. The safety factor associated with a bay does not correlate with the thickness of the bay. The 3-D FEA shows that loads redistribute from thinner regions to thicker regions. This suggests that the uncertainties in the thicknesses of the individual bays have relatively small effects on the safety factors, unless the average thickness of the entire sandbed region has been significantly overestimated. A comparison of the results of the licensee analysis with those of the SNL analysis of the thickness, based on the individual ultrasonic thickness measurements, reveals overestimation of the average thickness is unlikely.

The FEA performed by SIA compute buckling loads based on perfect geometries. For some structures such as beams and thick-walled cylinders, the observed buckling loads are close to those predicted based on "perfect" geometry. For thin cylindrical and spherical shells, the observed buckling load can be a small fraction (as low as 0.2) of that predicted for the perfect structure. To account for this imperfection sensitivity, capacity reduction factors, i.e., multipliers on the predicted loads, are introduced.

In the 1992 structural analysis reviewed and approved by the NRC staff, the buckling analysis used ASME Code Case N-284, Revision 1 to compute the capacity reduction factors. The staff accepted the use of this Code Case in the 1992 analysis. 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. At our February 1, 2007 meeting, Dr. C. Miller, the author of the ASME Code Case, described the technical basis for the Code Case and presented experimental results to demonstrate that the increased capacity factor was applicable. The increased capacity factor used in the 1992 analysis provided by the licensee was based on results for cylindrical shells. Dr. Miller showed results of tests conducted on spherical shells which demonstrated that the results for cylindrical shells were conservative for spherical shells. The staff reaffirmed its position that the use of the increased capacity factor was acceptable for the analysis of the Oyster Creek drywell shell. We concurred with this position.

In our assessment of the current analysis, we sought additional input on the capacity reduction factors computed from the Code Case and used by SIA. Our consultant, Professor John Hutchinson of Harvard, is known for his analytical studies of the effect of small geometric imperfections on the buckling loads of cylindrical and spherical shells. He performed an analysis to get an independent estimate of the capacity factor for a spherical shell under biaxial loads such as those that occur in the analysis of the drywell shell. The Code Case results based on the empirical formulas developed by Dr. Miller are slightly more conservative than the results of Professor Hutchinson's analysis. Thus, we remain convinced that the use of the modified capacity factors by SIA is appropriate.

Dr. Sam Armijo did not participate in the Committee's deliberations regarding this matter.

Sincerely,

Mario V. Bonaca

REFERENCES

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2. Report from W. J. Shack, Chairman, ACRS, to D. E. Klein, Chairman, NRC, "Report on the Safety Aspects of the License Renewal Application for the Oyster Creek Generating Station," 02/08/2007 (ML070390474)

3. Memorandum from M. Khanna, NRR, to R. Conte, Region I, "Assessment of the Oyster Creek 3-D Finite Element Analysis of the Drywell Shell," 05/12/2009 (ML091310413)

4. Letter from J. Lipoti, State of New Jersey, to US NRC transmitting, "New Jersey Department of Environmental Protection Oyster Creek Drywell Review," prepared by Becht Nuclear Services, 04/07/2009 (ML091040736)

5. ACRS Consultant Report from Dr. G. M. Wilkowski, to W. J. Shack, ACRS Subcommittee Chairman, "Final Report on Review of Oyster Creek Generation Station 3-D Drywell Confirmatory Analyses," 10/07/2009 (ML92870452)

6. ACRS Consultant Report from J. W. Hutchinson, "Comments on the Buckling Assessment of the Oyster Creek Drywell Shell with Emphasis on the Determination of Capacity Reduction Factors," 10/02/2009 (ML092870423)