AmerGen

An Exelon Company

Michael P. Gallagher, PE
Vice President
License Renewal Projects

AmerGen
200 Exelon Way
KSA/2-E
Kennett Square, PA 19348

Telephone 610.765.5958
www.exeloncorp.com
michaelp.gallagher@exeloncorp.com

10 CFR 50
10 CFR 51
10 CFR 54

2130-06-20300
May 1, 2006

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Oyster Creek Generating Station
Facility Operating License No. DPR-16
NRC Docket No. 50-219

Subject: Response to NRC Request for Additional Information, dated March 30, 2006, Related to Oyster Creek Generating Station License Renewal Application (TAC No. MC7624)

Reference: "Request for Additional Information for the Review of the Oyster Creek Nuclear Generating Station, License Renewal Application (TAC No. MC7624)," dated March 30, 2006

In the referenced letter, the NRC requested additional information related to Sections 4.3, 4.6, and 4.7 of the Oyster Creek Generating Station License Renewal Application (LRA). Enclosed are the responses to this request for additional information.

If you have any questions, please contact Fred Polaski, Manager License Renewal, at 610-765-5935.

I declare under penalty of perjury that the foregoing is true and correct.

Respectfully,

Executed on 05-01-2006

Michael P. Gallagher
Vice President, License Renewal
AmerGen Energy Company, LLC

Enclosure: Response to 03/30/06 Request for Additional Information

A114

cc: Regional Administrator, USNRC Region I, w/o Enclosure
USNRC Project Manager, NRR - License Renewal, Safety, w/Enclosure
USNRC Project Manager, NRR - License Renewal, Environmental, w/o Enclosure
USNRC Project Manager, NRR - OCGS, w/o Enclosure
USNRC Senior Resident Inspector, OCGS, w/o Enclosure
Bureau of Nuclear Engineering, NJDEP, w/Enclosure
File No. 05040

<u>Enclosure</u>

Response to 3/30/06 Request for Additional Information
Oyster Creek Generating Station
License Renewal Application (TAC No. MC7624)

RAI 4.3-1
RAI 4.3-2
RAI 4.3-3
RAI 4.3-4
RAI 4.6-1
RAI 4.7.3-1
RAI 4.7.3-2
RAI 4.7.3-3

RAI 4.3-1

Section 4.3.1 of the license renewal application indicates that the fatigue usage (based on the use of projected cycles for 60-years) for the reactor vessel closure studs, the vessel support skirt and the basin seal skirt to vessel flange junction was predicted to exceed the Oyster Creek acceptance limit of 0.8. The application also indicates that the fatigue usage of these components was shown to be acceptable by using more refined analysis methods. Describe the more refined analyses that were performed for these components.

Response

RPV Closure Studs: The original design analysis for the RPV and closure bolting predicted a cumulative fatigue usage factor for the head closure studs of 0.796 for 40 years of operation. When projected to 60 years of operation the fatigue usage for the head closure bolts would exceed the original acceptance limit of 0.8. In August 2002, a revised analysis for the entire RPV flange/stud region, including the RPV closure studs, was prepared to support a revised head tensioning sequence. The new analysis included a fatigue usage calculation for the studs that used the 1995 Edition of Section III of the ASME Code, including the 1996 Addenda. As a part of the License Renewal process, the closure stud fatigue calculation was revised to use the number of actual cycles projected for 60 years of plant operation, which produces a fatigue usage value of 0.196.

RPV Support Skirt: A revised analysis was performed for the RPV support skirt using Section III of the 1995 Edition of the ASME Code, including the 1996 Addenda. The fatigue calculation used the original RPV Stress Report stresses as input, but was reconciled to use the more modern ASME Code Section III methodology by modifying those stresses, where appropriate, to account for the Young's Modulus ratio, adjust for K_e, and to use the updated fatigue curve. In addition, the analysis used the number of cycles currently projected for 60 years, based on plant operating history. A revised fatigue usage value of 0.710 for 60 years of plant operation was obtained from this evaluation.

RPV Basin Seal Skirt: A revised analysis was performed for the RPV basin seal skirt as a part of license renewal efforts. The stress and fatigue evaluation in the original RPV Stress Report was maintained for this component with two exceptions: (1) a finite element analysis was performed to determine refined geometric stress concentration factors for the critical location, and (2) projected cycles for 60 years of plant operation were used. To accomplish Item (1), a finite element model (FEM) was constructed that included the vessel flange, a portion of the vessel shell, the basin seal skirt, the basin seal skirt plate, and the fillet attachment weld between the basin seal skirt and the RPV flange. Tension and bending loads were applied to the FEM so that revised tension and membrane stress concentration factors could be determined. The original fatigue analysis for the basin seal skirt component was revised to incorporate the refined geometric stress concentration factors, as well as projected cycles for 60 years of plant operation. A fatigue usage value of 0.270 for 60 years of plant operation was obtained using the 1963 Edition of the ASME Code (consistent with the original RPV Stress Report for this location).

RAI 4.3-2

Section 4.3.1 of the license renewal application indicates that the reactor vessel feedwater nozzles were reanalyzed to account for the effects of rapid thermal cycling. The application also indicates that the analysis satisfied the original Oyster Creek reactor vessel design limits. However, Table 4.3.1-2 of the license renewal application indicates that the 40-year fatigue usage of the feedwater nozzle was 0.952. Clarify whether the reanalysis of the feedwater nozzle for the rapid thermal cycling satisfied the original Oyster Creek reactor vessel design fatigue limit of 0.8. Also, indicate when the analysis that calculated the fatigue usage of 0.952 was performed and provide the basis for its acceptance.

Response:

In the original RPV Stress Report, fatigue was analyzed in accordance with GE specification 21A1105. The GE specification provided a fatigue curve and established a conservative fatigue usage factor acceptance limit of 0.8. The original RPV Stress Report fatigue analysis was computed for a 40-year period for the nozzle blend radius region, and had a predicted cumulative fatigue usage value of 0.1.

As a result of the repair of crack indications found in the feedwater nozzles in 1977, the feedwater nozzles were reanalyzed, as documented in MPR Associates, Inc. Report No. MPR-568, "Design Report for Replacement FW Sparger," December 1977. The 1977 MPR analysis did not account for rapid thermal cycling at the feedwater nozzle resulting from bypass leakage behind the thermal sleeve. However, the analysis contains one significant conservatism; the analysis assumed 60 cycles per year of on/off feedwater flow at low power conditions. The feedwater control system was modified following the repair of the feedwater nozzles to prevent such cycling. The MPR report concluded that the feedwater nozzles would reach an allowable usage factor of 1.0 after approximately 36 additional years of plant operation (or 42 years after initial plant startup), including the 60 cycles/year of hot standby (on/off) flow injection cycling. The allowable value of 1.0 used in the re-evaluation was consistent with the ASME Code Section III edition that was used as a basis to perform the fatigue analysis, but was inconsistent with the acceptance criterion cited in the original RPV Stress Report. Oyster Creek has recently changed the cumulative usage factor acceptance limit for the RPV from 0.8 to 1.0, using the 10 CFR 50.59 process, making it consistent with the ASME Section III fatigue usage acceptance limit.

The first step as part of the license renewal process was to establish a 40-year projected fatigue usage for the feedwater nozzles. The 1977 MPR analysis was re-visited, and a 40-year usage factor of 0.952 was developed using the thermal cycles assumed in the MPR analysis. The evaluation was then revised to properly account for modification to prevent low power feedwater thermal cycling, as well as to account for the number of plant transients projected for 60 years of operation. From this revised evaluation, the projected 60-year cumulative fatigue usage for the feedwater nozzles was determined to be 0.368.

As the final step in the license renewal effort, a refined fatigue calculation was performed using Section III of the 1995 Edition of the ASME Code, including the 1996 Addenda to develop detailed and up-to-date input for stress based fatigue (SBF) monitoring of the nozzles in the Fatigue Monitoring Program (FatiguePro). The refined license renewal fatigue calculation used the stresses from the 1977 re-evaluation as input, but modified those stresses where appropriate to use the more modern ASME Code Section III methodology (i.e., Young's Modulus ratio, K_3, updated fatigue curve, etc.). The projected cycles for 60 years of plant operation were again used. A revised fatigue usage value of 0.389 for 60 years of plant operation was obtained from this evaluation.

The 1977 analytical results did not include any usage factor from rapid thermal cycling effects. The impact of rapid cycling was evaluated by MPR in 1983 (MPR-783) and found to have a small impact of cumulative fatigue usage. As a part of the license renewal process, the rapid cycling calculations were re-evaluated for bounding reactor conditions, and it was confirmed that the design of the thermal sleeve/sparger assembly and associated flow baffle installed in 1977 is effective in reducing the fatigue due to rapid cycling to a negligible value.

<u>RAI 4.3-3</u>

Section 4.3.3.2 of the license renewal application discusses the fatigue evaluation of the isolation condenser. Provide the following information regarding the evaluation:

a. **The application indicates that a fatigue analysis was not performed as part of the original component design. The application also indicates that a later evaluation was performed for the tube bundle replacement in 1998. The application further indicates that the design life of the tube bundle replacement is 1500 cycles. Explain how the design life of 1500 cycles was determined. Provide the fatigue usage based on the peak stresses calculated for the Oyster Creek tube bundle replacement.**

b. **The application references the fatigue analysis of the Nine Mile Point Unit 1 isolation condenser. The application indicates that the Nine Mile Point, Unit 1 isolation condenser stress and fatigue results are considered bounding for Oyster Creek. Provide a detailed discussion of how it was determined that the Nine Mile Point, Unit 1 analysis was bounding for Oyster Creek. The discussion should include a comparison of the isolation condenser sizes and the sub-component materials, geometries and thicknesses. The discussion should also address the tube and shell thermal transients and flow rates.**

c. **The application indicates that the isolation condenser piping outside of the containment was evaluated for fatigue as part of a leak-before-break (LBB) analysis completed in 1991. The application also indicates that the piping**

outside the drywell was replaced in 1992. Provide the design criteria that was used to evaluate the replacement piping, including the number and types of thermal transients analyzed. Provide the maximum calculated fatigue usage for the replacement piping.

Response:

(a) At the time of tube bundle replacement in 1998, stress and fatigue analyses were performed for the Oyster Creek Isolation Condensers (Reference: Holtec International Report No. HI-982027, Revision 0, "Oyster Creek Nuclear Plant Emergency Isolation Condenser Tube Bundle Head Stress Analysis," 9/15/98, HOLTEC proprietary). As a part of this assessment, six cross sections were selected for stress analysis:

Cross Section	Description
1	Inlet pipe to nozzle junction
2	Hemispherical head attachment weld
3	Tubesheet
4	Cylinder flange
5	Tubesheet-outer casing attachment weld
6	Outer casing attachment weld

A maximum alternating stress intensity of 189.4 ksi was reported for the limiting cross section based on the thermal and pressure transients evaluated. Specific stresses for each cross-section were not documented. Based on a bounding alternating stress intensity of 195 ksi, 1,500 allowable cycles were established for the Isolation Condensers.

Based on the above allowed cycles and the number of projected cycles presented in Table 4.3.1-1 of the LRA the following fatigue usage for each Isolation Condenser is determined to be:

	40-years	60-years
Isolation Condenser A	0.252	0.321
Isolation Condenser B	0.277	0.347

Note that the above usage factors are very conservative since they utilize the total cycle counts for all years of operation for OCGS even though all of the evaluated cross sections have been replaced well after plant startup.

(b) At the time that the Oyster Creek License Renewal Application was prepared, a plant-specific stress and fatigue analysis had not been located for the Oyster Creek Isolation Condensers. To be prudent and based on field experience, it was decided to add the Oyster Creek Isolation Condensers to the Fatigue Management Program based on use of the Nine Mile Point analyses. The intent was to use the Nine Mile Point evaluation as a gauge to estimate fatigue and to identify when corrective action might be necessary. Since that time, the Holtec Report identified in the response to RAI 4.3-3(a) was located. AmerGen intends to remove all references to the Nine Mile Point analysis, and utilize the Oyster Creek plant-specific analysis as a basis for including the Isolation Condensers into the Fatigue Management Program.

(c) The Isolation Condenser piping outside of the drywell was replaced in 1992. As a part of that replacement effort, a fracture mechanics evaluation was performed, as well as a full ASME Code Section III (1989 Edition) fatigue evaluation on the Isolation Condenser system piping outside the drywell (Reference: MPR Associates, Inc. Document No. MPR-1226, Volumes I and II, April 1991, "Oyster Creek Nuclear Generating Station Leak Before Break Evaluation of Isolation Condenser System Piping Outside Containment"). A conservative number of system actuations was assumed (10 per year for 40 years), during which the condensate return piping was conservatively assumed to undergo a step change in temperature from 70°F to 575°F (the system design temperature). In addition, conservative stress intensity factors were assumed for weld joints and discontinuities between pipe and fittings and within fittings. The calculated usage factors for 40 years of operation were all found to fall below 0.2 (maximum value = 0.174 in the Condensate line), with a Code allowable of 1.0. Therefore, fatigue failure of the piping was not considered to be a concern. Based on these results, the usage factor screening criterion of 0.4, and the fact that the more bounding Isolation Condenser assembly is monitored, the Isolation Condenser piping outside of containment was excluded from the Oyster Creek Fatigue Management Program.

RAI 4.3-4

Section 4.3.4 of the license renewal application discusses the evaluation of the effects of the reactor coolant environment on the fatigue life of components and piping. Table 4.3.4-1 *provides the overall environmental fatigue multipliers for the components* analyzed. Provide the calculation of the environmental factors for the RPV inlet and outlet nozzles and the feedwater nozzle. Explain how each parameter used in the calculation was determined.

Response:

The environmental fatigue calculations for the recirculation inlet and outlet nozzles and the feedwater nozzle are contained in Structural Integrity Associates Calculation No. OC-05Q-314, Revision 0, "Environmental Fatigue Calculations for RPV Locations" (proprietary). The calculations for all three of these locations are performed in accordance with NUREG/CR-6583 (ANL-97/18), "Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low-Alloy Steels," March 1998, as the limiting locations for all three components are low alloy steel material. All three locations were evaluated in a similar fashion, based on the governing fatigue calculation for each component, with the following specifics:

Recirculation Inlet Nozzle: Bounding F_{en} multipliers for hydrogen water chemistry (HWC) and normal water chemistry (NWC) were determined based on maximum transient temperature, minimum (saturated) assumed strain rate, and oxygen values estimated for the recirculation system. An overall usage factor was computed considering the NWC F_{en} value for the time period prior to HWC implementation (41%) and the HWC F_{en} value for the time period after HWC implementation until the end of the 60-year operating period (59%).

Recirculation Outlet Nozzle: The fatigue usage at the outlet nozzle is greater than at the inlet nozzle, primarily because the outlet nozzle experiences the added thermal transients associated with operation of the Isolation Condenser. As a result, the fatigue usage for the outlet nozzle was calculated using a more detailed approach. Load pair specific F_{en} multipliers for hydrogen water chemistry (HWC) and normal water chemistry (NWC) were determined based on the maximum load pair temperature, average computed (tensile) strain rate, and oxygen values estimated for the recirculation system. Because load-pair specific F_{en} multipliers were determined based on load-pair specific strain rates and temperatures, the overall F_{en} multiplier for the recirculation outlet nozzle was determined to be significantly lower than the bounding value described above for the recirculation inlet nozzle. An overall usage factor was computed considering the NWC F_{en} value for the time period prior to HWC implementation (41%) and the HWC F_{en} value for the time period after HWC implementation until the end of the 60-year operating period (59%).

Feedwater Nozzle: Similar to the recirculation outlet nozzle, load pair specific F_{en} multipliers for hydrogen water chemistry (HWC) and normal water chemistry (NWC) were determined based on the maximum load pair temperature, average computed (tensile) strain rate, and oxygen values estimated for the feedwater system. An overall usage factor was computed considering the NWC F_{en} value for the time period prior to HWC implementation (41%) and the HWC F_{en} value for the time period after HWC implementation until the end of the 60-year operating period (59%).

As a result of a recent review of our documents, it was noticed that some of the fatigue usage factors cited in Section 4.3 of the OC LRA do not correspond to the values in our latest fatigue calculations. Some fatigue values changed slightly as a result of incorporation of comments during the finalization of these calculations. As part of the corrective actions for this finding a complete review of all of the values cited in LRA Section 4 was performed to ensure the fidelity of the information provided. The changes are shown in bold face on the new Table 4.3.4-1 (below). The changes are relatively small and do not impact the conclusions discussed in the application.

Table 4.3.4-1
Environmental Fatigue Results for Oyster Creek for NUREG/CR-6260 Components

NUREG/CR-6260 Location	Equivalent OCGS Location	Material	60-Year Fatigue Usage Factor (1)	60-Year Fatigue Usage Factor with Environmental Effects (2)	Overall Environmental Fatigue Multiplier
Reactor Vessel (Lower Head to Shell Transition)	Reactor Vessel (Vessel-Head Junction)	Low Alloy Steel	0.0004	0.0042	10.28
Feedwater Nozzle	Feedwater Nozzle	Low Alloy Steel	0.3889	0.8433	2.17
Recirculation System (RHR Return Line Tee and the RPV inlet and outlet nozzles)	Isolation Condenser Return Line Tee into SDC Line	Stainless Steel	~~0.851~~ **0.1205**	~~0.493~~ **0.43**	~~5.79~~ **3.57**
	RPV inlet nozzle	Low Alloy Steel	0.0151	0.1554	10.28
	RPV outlet nozzle	Low Alloy Steel	~~0.131~~ **.1832**	0.978	5.34
Core Spray System (Nozzle and Safe End)	Core Spray Nozzle Core Spray Nozzle Safe End	Low Alloy Steel Stainless Steel	0.0013 0.0006	0.0129 0.0072	10.28 12.48
Residual Heat Removal Line (Tapered Transition)	Bounded by Isolation Condenser Return Line Tee Location Above	Stainless Steel	N/A	N/A	N/A
Feedwater Line (Feedwater/RCIC Tee Connection)	Limiting Class 1 Location in the Feedwater Line	Carbon Steel	~~0.0789~~ **0.0245**	~~0.178~~ **0.0767**	~~2.26~~ **3.13**

Notes:

1. Revised fatigue usage factors were computed for all of the NUREG/CR-6260 components based on projected cycles for 60 years of plant operation and updated ASME Code fatigue methodology.
2. Environmental fatigue usage was computed using the methodology of NUREG/CR-6583 (for carbon/low alloy steels) and NUREG/CR-5704 (for stainless steels), as appropriate for the material for each location.

RAI 4.6-1

Section 4.6 of the license renewal application discusses the fatigue of the primary containment. The application indicates that a structural evaluation of drywell thinning at various locations was performed in 1986 and 1987. Describe the structural evaluation that was performed and indicate whether the evaluation involved any TLAAs.

Response:

As stated in the Section 4.6 of the LRA, drywell shell plates were not evaluated for fatigue. The structural evaluation of drywell wall thinning cited in Section 4.6 of LRA refers to the statement made in 3.8 of the UFSAR. These evaluations are discussed in Section 3.8.2.8 of the UFSAR. LRA section 4.7.2 discusses drywell corrosion as a TLAA. Updated information regarding the evaluation of drywell wall thinning is provided in the response to RAI 4.7.2-1, which was transmitted to the NRC on April 7, 2006 in AmerGen letter 2130-06-20289.

RAI 4.7.3-1

The staff needs the following additional information to complete its review of this TLAA:

a. **An explanatory figure of the equipment pool and the reactor cavity wall areas affected by the rebar corrosion and leakages.**

b. **The extent of areas of walls affected by the corrosion and leakages.**

c. **Calculated maximum stresses in the affected rebars during (1) normal operating condition, (2) the postulated accident condition, and (3) during the postulated seismic event for which the walls are designed.**

d. **The effect of the 60-year corrosion on the stresses calculated in Item c. above.**

Response:

a. The attached explanatory figure (Figure 1) provides a plan of the area affected by rebar corrosion and water leakage. Locations affected by corrosion and leakage are localized and noted on the figure. Water and rust stains were observed around hairline cracks on the exterior surfaces of these walls in 1986. As a result, these areas were considered suspect for rebar corrosion. The walls are also affected by the elevated temperature in the upper region of the drywell, evaluated under Integrated Plant Assessment Systematic Evaluation Report, SEP Topic III-7B.

b. Areas of reactor cavity and equipment pool walls affected by rebar corrosion are limited to localized portions of the walls between elevation 95' and 119'. These local areas were documented in a Material Nonconformace Report MNCR #86-870 in 1986 when a reddish brown deposit (rust like) was observed in and around hairline cracks in the walls. Later, it was determined that these deposits were from iron oxide corrosion products from the embedded reinforcement steel and the corrosion was most likely a product of water leakage during refueling outages. It was considered probable that treated water entered the pre-existing cracks in the concrete wall, which wetted the surface of the rebar (Ref. 1). Based on these determinations GPU concluded that a corrosion damage assessment was necessary to establish the degree of rebar corrosion.

   To accomplish the corrosion assessment, concrete core samples were taken and tested as described in response to RAI 4.7.3-2 (1) below to determine if water intrusion into the cracks created an environment that is aggressive to the rebar. The test results show that the environment is not aggressive and only minimal rebar corrosion, if any, should be expected. However since rust was observed in and around the hairline cracks on the walls, GPU conservatively used 0.001 inch/year corrosion rate, as discussed in response to RAI 4.7.3-2 (1) below. Using this corrosion rate, Oyster Creek estimated that the rebar diameter will be reduced by 0.002 inch/year and that the diameter of the affected #8 and #11 rebar will be reduced by approximately 8% and 6% respectively over a 40-year period. This information was submitted to the NRC Staff in a letter dated December 5, 1990 (Ref. 2). Engineering evaluation and subsequent analysis determined that the reduction in rebar diameter has no impact on structural integrity of the affected reinforced concrete walls as discussed in item c below.

   In 1993 GPU conducted additional evaluations to assess the condition of the rebar using a corrosion rate that is based on plant operating experience. For this evaluation GPU reviewed the loss of metal in the upper region of the drywell shell thickness that are based on actual UT measurements. The review indicated that drywell shell thickness in this area was reduced by approximately 0.020 inches. Conservatively assuming the affected #8 and #11 rebar experienced the 0.020 inches corrosion all around; the rebar diameter would be reduced by 0.040 inches. This represents approximately 8% reduction in cross section area of #8 rebar and 6% reduction in the cross section area of the #11 rebar.

   The GPU evaluation also noted that given the minimal amount of time the rebar is exposed to moisture, the fact that concrete provides an alkaloid environment which limits corrosion of reinforcing, and the fact that no additional indications of corrosion have been observed, GPU believes significant corrosion has not occurred and will not occur in the future. This information was transmitted to the NRC Staff in a letter dated November 19, 1993.

   The NRC review of information submitted in the November 19, 1993 letter was documented in the drywell shield wall Safety Evaluation Report dated May 11, 1994. The Staff found GPU's evaluation acceptable but was concerned about concrete cracks on the outside surfaces of the drywell shield wall and the potential of additional corrosion that could occur because of water leakage or spills. The Staff recommended the upper portion of the drywell shield wall be monitored and repair, as necessary, cracks greater than 0.02 inches in width.

As recommended by the Staff, the upper portion of the drywell shield wall is monitored for cracks during refueling outages. To date, cracks greater than 0.02" have not been identified and no repairs have been made. In addition, exterior surfaces of the wall are observed for water and rust stains. Inspections conducted in 1994, 1996, 1998, 2002, and 2005 identified no indications of water stains or rust stains and no concrete spalling or additional significant cracking was observed. This provides reasonable assurance that significant rebar corrosion is not occurring.

Based on the information above and as discussed in more detail in response to RAI 4.7.3-2 item (2) below, AmerGen concurs with GPU's conclusion that significant corrosion has not occurred in the current term. AmerGen evaluated and concluded that significant corrosion will not occur during the period of extended operation. However, because the rebar is inaccessible for direct visual examination, AmerGen is conservatively assuming an additional 0.010 inches loss of metal all around the #8 and the # 11 rebar. This represents a total rebar corrosion of 0.030 inches (0.020 inches during the current term plus 0.010 inches during the extended period of operation). This results in approximately 13% reduction of cross section area of the #8 rebar and 8% reduction in cross section area of # 11 rebar. The impact of this reduction in rebar stress is addressed in item c below.

c. The calculated maximum stress used to evaluate the affected rebar by corrosion is 32.8 kips per square inch (ksi). This maximum stress is based on the comprehensive analysis conducted by GPU to assess the impact of observed cracking and elevated temperature on the spent fuel pool structure and the drywell shield wall. The analysis was conducted using a finite element ANSYS model of the north side of the reactor building, which includes half of the drywell shield wall (DSW) and the spent fuel storage pool (SFP). The results of the analysis were summarized in ABB Impell Report No. 03-0370-1341 and transmitted to the NRC Staff in a letter dated September 1992. The results of the analysis show that for load combinations that involve operating and seismic loads (combinations 3.3.2c and 3.3.2d in ABB Impell Report No. 03-0370-1341, page 39), the maximum calculated stress is 32.8 kip per square inch (ksi). This maximum stress is only in a few elements in the area of the fuel transfer canal on the south wall of the spent fuel pool. The areas affected by rebar corrosion are in the south side of the reactor building, away from the transfer canal and from the heavily loaded spent fuel pool area. Thus, using 32.8 ksi stress for areas affected by rebar corrosion is very conservative because loads in the north side half of the reactor building are significantly higher than the south half of the building due to the fuel pool structure weight and the weight of the high density spent fuel racks.

In a letter from Alexander W. Dromerick (NRC) to John J. Barton (GPU), "Request for Additional Information -SEP Topic III-7B, Shield Wall Temperature", dated July 26, 1993, NRC requested GPU provide numerical values of stresses under load combinations 3.3.2c and 3.3.2d in the concrete and reinforcing bars in the drywell shield wall, above elevation 95 ft. The NRC also requested GPU to discuss the measures taken (if any) to prevent the migration of moisture through the cracks to alleviate rebar corrosion. In response to the request, GPU transmitted to the NRC in a letter dated November 19, 1993, ABB Impell Report No. 0037-00196-01 (see Attachment 1). The report summarizes rebar and concrete stresses at locations used to evaluate the capacity of the drywell shield wall above elevation 95 ft. for load combinations 3.3.2c and 3.3.2d. These load combinations include normal operating loads, and design basis seismic loads. The maximum calculated reinforcement stresses in critical locations (worst area) are 32.8 ksi for load combination 3.3.2c and 31.4 ksi for load combination 3.3.2d.

As discussed in response to item b of this RAI, the estimated reduction in rebar cross section area, in locations affected by this rebar corrosion, through the period of extended operation is 13% for #8 rebar and 8% for #11 rebar. This results in a stress increase of 14.5% for the #8 rebar and 9.1% for the #11 rebar.

Based on stress increases discussed above and using the maximum calculated stress of 32.8 ksi, the maximum stress in the #8 rebar affected by corrosion is 37.6 ksi and 35.8 ksi for the #11 rebar. These stresses remain below ACI yield stress of 40 ksi. The calculated 32.8 ksi stress is overly conservative for the DSW affected by the rebar corrosion because it is based on the highly loaded spent fuel pool (high density racks) and for the highly stressed area around the slot in the south wall of the fuel pool.

d. As discussed in TLAA 4.7.3 analysis, periodic inspections of the reactor cavity and equipment pool walls conducted since the mid 1990s show no signs of water intrusion or indications of further deterioration of the rebar. The TLAA was based on the corrosion rate of 0.001 inch/year reported to the NRC in the December 5, 1990 letter. However after submittal of the LRA, the more recent corrosion information identified to the NRC in the November 19, 1993 was discovered. This 1993 letter informed NRC that GPU believes that corrosion is not ongoing.

   Although there is no evidence of continuing rebar corrosion, AmerGen is conservatively assuming a corrosion of 0.010 inches all around the rebar during the period of extended operation, in addition to the assumed corrosion of 0.020 inches all around for the current term. This results in a total assumed corrosion of 0.030 inches, yielding a reduction of cross section area of 13% for #8 rebar and 8% for #11 rebar. The maximum tensile stress in rebar affected by corrosion is found to be 37.6 ksi, for the reinforcing steel having the minimum yield strength of 40 ksi. Since the corrosion continues to be localized there is no significant impact on structural integrity of the reinforced concrete walls. See response to RAI 4.7.3-2, (2) for additional rationale why the corrosion rate is conservative.

RAI 4.7.3-2

The staff requests the applicant to provide (1) the bases for the corrosion rate established in the analysis, (2) assertions that these rates will not be exceeded during the period of extended operation, and (3) a summary of the program for monitoring the actual corrosion of the rebar during the period of extended operation.

Response:

(1). The corrosion rate was derived based on chemical analysis of concrete core samples taken on a location that is representative of the drywell shield wall and the equipment pool walls concrete. The samples were analyzed via standard gravimetric, titrimetric, EDAX (Energy Dispersive X-ray), and leachate ion chromatography techniques. In addition, a pH determination was derived from the leachate sample. The samples were analyzed for total composition, chlorides, and sulfates. The test results indicated that rebar is exposed to a non-aggressive environment that contains 10 ppm chlorides, 890 ppm sulfates, and a pH of 11.6.

Based on the results of these analyses it was concluded that only a mild corrosion environment would exist due to an absence of aggressive levels of contaminants within an alkaline environment. Under this type of environment and considering that this rebar is not continuously wetted, it is estimated the rate of corrosion would be approximately 0.001 inch/year. Published corrosion data in Reference 4, for carbon steel (not rebar) in alkaline environment, was used as input to establish the corrosion rate of 0.001 inch/year. This was considered appropriate since the environmental conditions within the crack annulus are pH controlled rather than oxygen controlled. For evaluation of the walls, Oyster Creek conservatively estimated that the rebar diameter will be reduced by 0.002 inch/year as reported to the NRC in a letter from GPU to NRC dated December 5, 1990.

In 1993 GPU conducted additional evaluations to assess the condition of the rebar using a corrosion rate that is based on plant operating experience. For this evaluation GPU reviewed the loss of metal in the upper region of the drywell shell thickness that are based on actual UT measurements. The review indicated that drywell shell thickness in this area was reduced by approximately 0.020 inches. Conservatively assuming the affected #8 and #11 rebar experienced the 0.020 inches corrosion all around; the rebar diameter would be reduced by 0.040 inches. This represents approximately 8% reduction in cross section area of #8 rebar and 6% reduction in the cross section area of the #11 rebar.

The GPU evaluation also noted that given the minimal amount of time the rebar is exposed to moisture, the fact that concrete provides an alkaloid environment which limits corrosion of reinforcing, and the fact that no indication of corrosion has been observed, GPU believes significant corrosion has not occurred and will not occur in the future.

Based on this information and as discussed in more detail in response to RAI 4.7.3-2 item (2) below, AmerGen concurs with GPU's conclusion that significant corrosion has not occurred in the current term. AmerGen evaluated and concluded that significant corrosion will not occur during the period of extended operation. However, because the rebar is inaccessible for direct visual examination, AmerGen is conservatively assuming this rebar would be subject to additional corrosion of 0.010 inches all around the rebar during the period of extended operation.

(2). Oyster Creek asserts that the corrosion rate used to evaluate rebar corrosion is conservative and the rebar yield stress of 40 ksi will not be exceeded during the period of extended operation. First, the estimated corrosion of 0.020 inches for the current term is based on carbon steel in a slightly corrosive environment. The rebar is not subject to a corrosive environment as shown by concrete test samples. The assumed 0.010 inches for the period of extended operation is also conservative because there is no evidence of ongoing corrosion based on the existing monitoring activities in accordance with the Structures Monitoring Program (B.1.31).

Secondly, rebar embedded in concrete is passivated by the alkalinity of the concrete mix by forming a protective hydrous ferrous oxide on their exposed surfaces. Even when portions of the reinforcements are exposed via cracks in the concrete, which acts as a passageway for environmental contact, the rate of corrosion is generally low due to the barrier effect of the pre-existing oxide film. The limited corrosivity under these conditions within a crack annulus is a product of the alkaline leachant from the concrete and the slow diffusion of oxygen within the annulus and through the protective oxide layer. This type of condition would promote a weak electro-chemical corrosion cell, precluding dissolution of the protective film.

Thirdly, the cause of corrosion was attributed to water leakage from the reactor cavity and equipment pool during refueling outages. The source of leakage has been investigated extensively and determined to be due to cracks in the stainless liner of the wall. The cracks are now sealed with a strippable coating prior to filling the reactor cavity and the equipment pool with water. The strippable coating has been found effective in minimizing water leakage.. AmerGen has made a commitment (see AmerGen letter to NRC dated April 4, 2006) to continue applying the strippable coating during the period of extended operation.

Fourth, the water used to fill the reactor cavity and the equipment pool is treated in accordance with BWRVIP-130 guidance as described in Oyster Creek Water Chemistry aging management program (B.1.02). The treated water maintains an environment that is non-aggressive consistent with concrete sample test results described in item (1) above.

Also as discussed in NUREG-1801 Rev. 1, and EPRI Report #1002950, corrosion of embedded steel in concrete is not significant if the steel is not exposed to an aggressive environment defined as concrete pH<11.5 or chlorides >500 ppm. Oyster Creek concrete samples test, described in response to RAI 4.7.3-2 (1) above indicate that concrete pH=11.6, and chlorides=10 ppm. Thus the reinforcement is exposed to a non-aggressive environment and the corrosion is expected to be insignificant.

On the technical basis described above, AmerGen asserts that the estimated total corrosion of 0.020 inches all around the rebar diameter and the assumed of corrosion of 0.010 inches during the period of extended operation is bounding and will not be exceeded during the period of extended operation. Visual inspections conducted in 1994, 1996, 1998, 2002, and 2005, in accordance with the Structures Monitoring Program (B.1.31), identified no indications of water stains, or rust stains. This provides objective evidence and reasonable assurance that significant rebar corrosion is not occurring and that the walls will continue to perform their intended function during the period of extended operation.

(3). The walls affected by rebar corrosion are in the scope of the Oyster Creek Structures Monitoring Program (B.1.31). The walls will be inspected every refueling outage while the reactor cavity and equipment pool are full of water to ensure that water leakage during refueling is detected. The walls will be visually inspected for new cracks, crack growth, water stains, and rust stains. Monitoring these parameters provides reasonable assurance that significant rebar corrosion will be detected before a loss of an intended function.

<u>RAI 4.7.3-3</u>

The staff requests the applicant to provide the quantitative aspect (i.e., corrosion rate and amount of corrosion predicted) in Section A.5.3 of the UFSAR Supplement.

<u>Response</u>:

The license renewal application (LRA) Section A.4.5.3 will be revised to reflect the following paragraph:

Corrosion of reinforcing bar in localized areas of the reactor cavity and equipment pool walls was suspected as a result of observed rust in and around cracks in the walls between elevation 95' and 119'. To assess the condition of the reinforcing bars, concrete core samples were taken in 1988 and chemically analyzed to determine if water intrusion into concrete cracks created an environment that is aggressive to rebar. These analyses showed that the environment is not aggressive and thus corrosion should not be significant.

However because of the observed rust like substance in and around the cracks, the affected rebar were conservatively assumed to be subject to corrosion of 0.020 inches all around the rebar during the current term. Engineering analysis concluded the corrosion amount of reinforcing bars would not impact structural integrity of the affected walls during the current period of operation.

For the period of extended operation, corrosion of the reinforcing bars and the rate of corrosion is a TLAA. Although there is no evidence of continuing rebar corrosion, AmerGen is conservatively assuming additional corrosion of 0.010 inches all around the rebar during the period of extended operation. Corrosion of the reinforcing bar has been projected to the end of the extended period in accordance with 10 CFR 54.21(c)(1)(ii), and determined that the intended function of the drywell shield wall and the equipment pool wall will be maintained through the period of extended operation.

References:

1. Letter from R. W. Keaten (GPU) to U.S. Nuclear Regulatory Commission, SEP Topic III-7B, Drywell Shield Wall Integrity, dated April 19, 1994

2. Letter from J. C. DeVine, Jr. (GPU) to NRC, Oyster Creek Drywell Containment, dated December 5, 1990

3. Letter from Alexander W. Dromerick (NRC) to John J. Barton (GPU), Evaluation of Effects of High Temperature on Drywell Shield Wall and Biological Shield Wall, SEP Topic III-7.B "Design Codes, Design Criteria, Load Combinations, and Reator Cavity Criteria", dated May 11, 1994

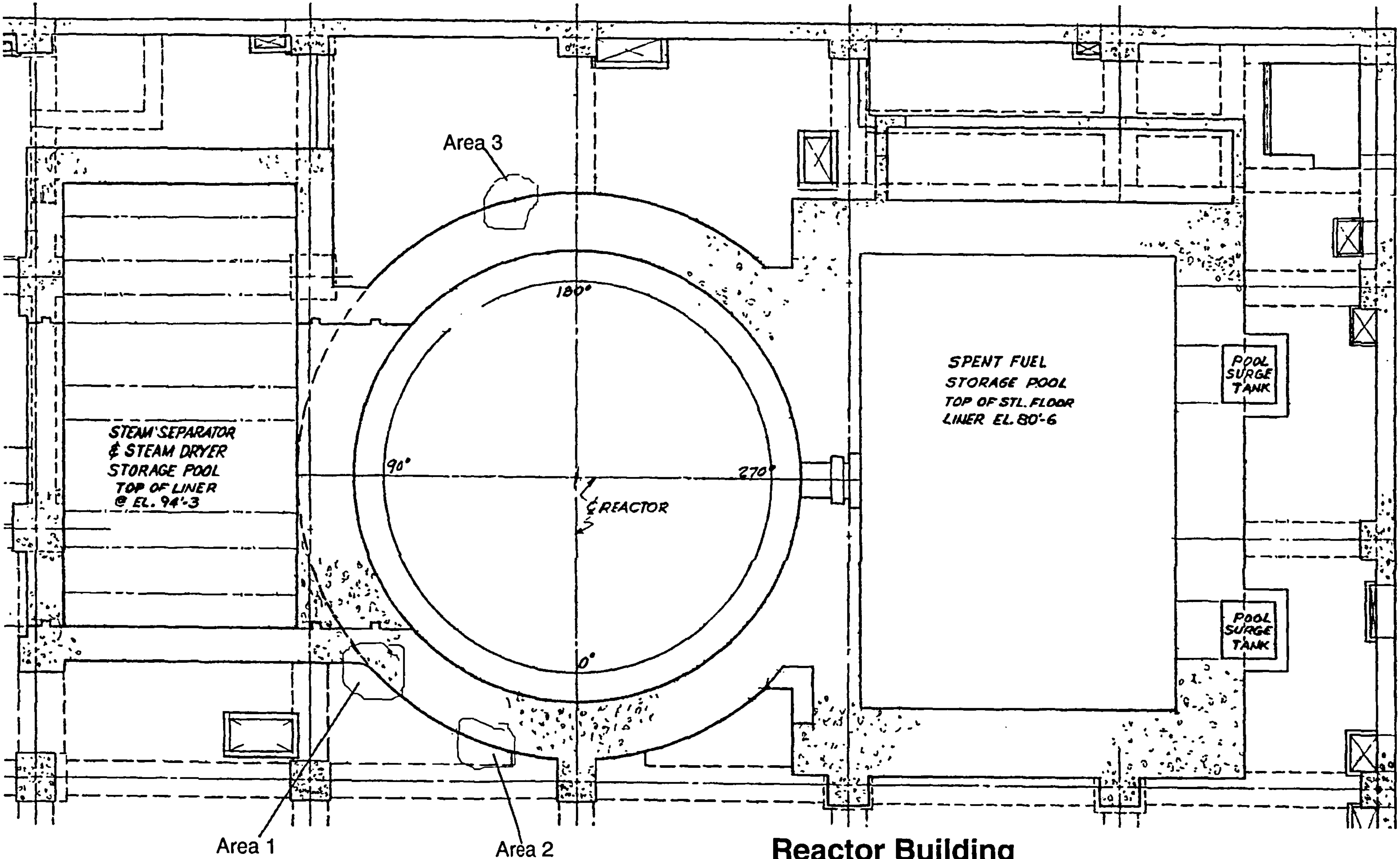
4. Uhlig's Corrosion Handbook, 2nd Edition

5. Letter from R. Keaten (GPU) to NRC, Request for Additional Information- SEP Topic III-7B, Shield Wall Temperature, dated November 19, 1993

RAI 4.7.3-1

Figure 1

Local Areas Affected by Rebar Corrosion



Reactor Building
Plan @ 95'3"

Note: Local areas between el. 95' & el. 119' where water and rust stains were observed.

Fig.-1. Local Areas Affected by Rebar Corrosion (Area 1, 2, & 3)

RAI 4.7.3-1

Attachment 1

ABB Impell Report No. 0037-00196-01

Nuclear

GPU Nuclear Corporation
One Upper Pond Road
Parsippany, New Jersey 07054
201-316-7000
TELEX 136-482
Writer's Direct Dial Number:

November 19, 1993
5000-93-0069
C321-93-2300

U. S. Nuclear Regulatory Commission
Att: Document Control Desk
Washington, DC 20555

Gentlemen:

Subject: Oyster Creek Nuclear Generating Station (OCNGS)
Docket No. 50-219
Request for Additional Information - SEP Topic III-7B,
Shield Wall Temperature (TAC No. M76879)

Your letter dated July 26, 1993 requested additional information relating to SEP Topic III-7B.

As requested, we have calculated stresses in the concrete and reinforcing bars in the drywell shield wall above elevation 95 ft. We have also determined that the stresses are below allowables considering the existing (cracked) condition of the shield wall. Attached is a report (ABB Impell Corporation Report No. 0037-00196-01, Rev. 0, dated September 20, 1993) which provides stresses in the concrete and steel and percent margin compared to allowables at each critical section. As you requested, all values are provided for load combinations 3.3.2c and 3.3.2d from the original Impell report (p. 39).

The analysis predicts cracking of the outside surface of the drywell shield wall above elevation 95'0". However, the analysis does not predict cracking of the inside surface except locally around the notch in the south wall of the spent fuel pool. A visual inspection of the outside surface of the wall was conducted and no indication of reinforcing corrosion was observed.

During normal plant operation, very little moisture is present in the vicinity of the drywell shield wall due to the relatively high temperatures. During refueling activities, the reactor cavity is flooded and the inside surface of the wall is exposed to the water. However, a steel plate covers this surface and prevents the water from directly contacting the concrete. While water leaks past this plate, it is not expected to cause substantial corrosion of the reinforcement due to the small percentage of time the cavity is flooded.

To obtain an estimate of the amount of corrosion that might have occurred, we reviewed loss of metal in the drywell shell in this area of the plant. In the upper regions of the drywell, approximately .020 inches in thickness of the drywell shell has been lost. If the No. 8 and No. 11 reinforcing bars present in the vicinity of the notch lost this amount of metal all around, the reduction in steel area would be only 10% and 4% respectively.

Given the minimal amounts of time the reinforcing is exposed to moisture, the fact that concrete provides an alkaloid environment which limits corrosion of reinforcing, and the fact that no indication of corrosion has been observed, GPU Nuclear is confident that significant corrosion has not occurred and will not occur in the future. Since significant margin exists between calculated reinforcing stresses and allowable stresses, and since the analysis indicates the spent fuel pool does not require any structural support from the drywell shield wall, GPU Nuclear believes that the adequate structural integrity of the shield wall is maintained.

Sincerely,

RWKeaten

R. Keaten
Director, Technical Functions

Attachment
RK/YN/plp

cc: Administrator, Region I
NRC Resident Inspector
Oyster Creek NRC Project Manager

OYSTER CREEK NUCLEAR GENERATING STATION
ADDITIONAL STRESS INFORMATION FOR THE DRYWELL SHIELD WALL
FROM THE STRUCTURAL EVALUATION OF THE SPENT FUEL POOL

Prepared for:

GPU Nuclear Corporation
One Upper Pond Road
Parsippany, NJ 07054

Prepared by:

ABB Impell Corporation
225 Scientific Drive
Technology Park / Atlanta
Norcross, GA 30092

Report No. 0037-00196-01
Revision 0
September 20, 1993

ASEA BROWN BOVERI
ABB Impell Corporation

TABLE OF CONTENTS

1.0 INTRODUCTION

1.1 Objectives

The objective of the structural evaluation of the Spent Fuel Pool (SFP) at the Oyster Creek Nuclear Generating System (OCNGS), initiated under GPUN Contract No. PC-082008, was to evaluate the SFP concrete structure for consolidated and unconsolidated fuel loads with other design basis loads. The general technical requirements for this evaluation are defined in GPUN Specification SP-1302-53-047, Revision 1, [Ref. 8].

The analysis considers the effects of dead load, live load, thermal gradients, seismic load and cask drop accident using prescribed loads and load combinations. The specific evaluation of section capacities and stresses are performed in accordance with ACI-349 [Ref. 6]. A detailed finite element model of the SFP concrete structure including all connecting and supporting members was generated to consider the effects of internal force redistribution and to obtain forces on the Reactor Building structural elements. The results and conclusions of the analyses are reported in ABB Impell report no. 03-0370-1341, Revision 0, [Ref. 1].

GPUN has requested, under contract no. PC-0443867, [Ref. 7], numerical values of stresses in the concrete and reinforcing steel in the OCNGS drywell shield wall above elevation 95 ft. These stresses are to be provided for load combinations 3.3.2c and 3.3.2d as contained on page 39 of ABB Impell report no. 03-0370-1341, revision 0, [Ref. 1]. Additionally, GPUN requested that the stresses be evaluated per ACI 349, [Ref. 6].

This report provides a summary of the stresses in the drywell shield wall above elevation 95 ft., and demonstrates that the stresses in the concrete and the reinforcing steel are within allowables.

1.2 Scope

The scope of work for this project is as follows:

1.2.1 Retrieve backup files for SFP analysis

The backup tapes for the Spent Fuel Pool Analysis are obtained from storage. The ANSYS postprocessing files for iteration 4S, Analysis Case C, load combinations 3.3.2c and 3.3.2d from the previous analysis are reloaded on to the computer. This iteration is chosen because it represents the most realistic estimation of the present stiffness of the Reactor Building.

1.2.2 Verification of retrieved data

The results for load combinations c and d are examined closely to ensure that the appropriate load combinations are loaded. This is accomplished by comparing stress results to those reported in the SFP Report and Calculations.

1.2.3 Generate color contour stress plots and tables

These postprocessing files are then used to produce plots and lists of the vertical and hoop stress levels in the drywell shield wall above elevation 95 feet for load combinations 3.3.2c and 3.3.2d. Both the outer and inner surfaces of the drywell shield wall are examined. The load capacities of the concrete and reinforcing steel at the highest stressed area in this region are determined and reported.

1.3 Brief Summary of Results

The results of the review of stresses in the drywell shield wall above elevation 95 ft., demonstrate that all sections meet the requirements of ACI 349, [Ref. 6], for load combinations 3.3.2c and 3.3.2d.

2.0 METHODS OF ANALYSIS

The following two load combinations are loaded to obtain stresses in the concrete shield wall:

c. $0.75\,(1.4D + 1.7L + 1.4T_o \pm 1.9E)$
d. $D + L + T_o \pm E'$

Where:

D	=	dead load as specified in Section 3.1.1 of [Ref. 1], Rack Conditions 2 and 3
L	=	design live load as specified in Section 3.1.2 of [Ref. 1]
T_o	=	thermal load due to temperature differential across the slab or wall. Two critical cases were considered as specified in Section 3.1.4 of [Ref. 1]
E	=	OBE seismic load as specified in Section 3.1.3 of [Ref. 1]
E'	=	SSE seismic load as specified in Section 3.1.3 of [Ref. 1]
C	=	Cask drop load as specified in Section 3.1.5 of [Ref. 1]

Verification that appropriate load combinations are loaded is contained in ABB Impell Calculation no. 0037-00196-C002, revision 0, [Ref. 3]. Contour stress plots and numerical values of stresses in the drywell shield wall above elevation 95 ft., are provided for the above load combinations. An ACI evaluation is also performed to demonstrate that the drywell shield wall and reinforcing steel are within allowables. These are documented in ABB Impell Calculation no. 0037-00196-C003, revision 0, [Ref. 4].

3.0 SUMMARY OF RESULTS

Figure 3.0-1 shows a view of the drywell shield wall for which stresses are extracted, (i.e. above elevation 95 ft.). Figures 3.0-2, 3.0-3 and 3.0-4 show the vertical stress (S_z) levels in the drywell shield wall above elevation 95 feet for load combination c, viewed in the Southwest, Southeast and North directions. Figures 3.0-5, 3.0-6 and 3.0-7 show the circumferential (hoop) stress (S_y) levels in the drywell shield wall above elevation 95 feet for load combination c, viewed in the Southwest, Southeast and North directions.

Figures 3.0-8, 3.0-9 and 3.0-10 show the vertical stress (S_z) levels in the drywell shield wall above elevation 95 feet for load combination d; viewed in the Southwest, Southeast and North directions. Figures 3.0-11, 3.0-12 and 3.0-13 show the circumferential (hoop) stress (S_y) levels in the drywell shield wall above elevation 95 feet for load combination d, viewed in the Southwest, Southeast and North directions.

These figures show that the highest stress levels in the shield wall and SFP occur in the SFP south wall at the fuel transfer opening. This location is a natural stress riser due to the stress concentrations around the slot. Figures 3.0-5 and 3.0-6, indicate a stress riser in the east and west shield wall a few feet below elevation 119 ft. However, the stress levels here are lower than those around the fuel transfer opening and this area of the wall contains similar reinforcing steel patterns as in the area around the fuel transfer opening, [Ref.9]. Therefore, the critical section is the area around the fuel transfer opening for all stresses and hence an ACI evaluation of this area is performed to demonstrate that the drywell shield satisfies the requirements of ACI 349, [Ref. 6].

Table 3.0-1 shows linearized hoop (S_y) stresses around the fuel transfer opening, for locations see Figure 3.0-14. See Section 5.2.4 of [Ref. 1], for further discussion on linearized stresses. The results shown in Table 3.0-1 indicate that circumferential tensile stresses rapidly decay away from the bottom of the fuel transfer opening. At a vertical distance of 90 inches away from the bottom of the fuel transfer opening, the tensile stress is shown to have decreased to 41% of the maximum tensile stress at the corner of the fuel transfer opening.

Table 3.0-2 shows linearized vertical (S_z) stresses around the fuel transfer opening, for locations see Figure 3.0-14. The results shown in Table 3.0-2 indicate that vertical tensile stresses rapidly decay away from the bottom of the fuel transfer opening. At a horizontal distance of 79 inches away from the bottom of the fuel transfer opening, the vertical tensile stress is shown to have decreased to 22% - 42% of the maximum tensile stress at the side of the fuel transfer opening.

In evaluating the capacity of the shield wall, two types of sections are examined. Figure 3.0-15 shows a typical section representing the bending stress acting in the circumferential direction , which corresponds to the section A-A shown in Figure 3.0-14. The depth of beam is taken to be the thickness of the shield wall below elevation 95 ft., and above the bottom of the spent fuel pool (i.e., 90 inches). The stresses from Table 3.0-1 are averaged and the corresponding moment and axial forces are evaluated considering moment-axial force interaction for ACI 349, [Ref. 6]. See Section 5.2.4 of [Ref. 1] for further discussion on computing moments and axial forces from averaged linearized stresses using RCBEAM.

For the vertical stresses (parallel to the side of the fuel transfer opening), the representative beam cross section is shown in Figure 3.0-16. For this cross section, the depth of the beam is taken to be 60 inches which corresponds to the thickness of the shield wall above elevation 95 ft. Two sections (Section A-B and section C-C as shown in Figure 3.0-14) are used to determine two sets of averaged stresses and associated moments and axial forces acting on the typical beam cross section shown in Figure 3.0-16. The resulting moments and axial forces are evaluated in the same fashion as for Section A-A.

The results of the ACI evaluation are presented in Table 3.0-3 which demonstrates that the concrete around the fuel transfer opening is within allowables and in compliance with ACI 349, [Ref. 6]. Table 3.0-4 demonstrates the maximum tensile stresses (S_t) in the reinforcing steel around the fuel transfer opening, as shown in Figure 3.0-14, are within allowables.

Figure 3.0-1: <u>**Model of drywell shield wall above elevation 95 ft.**</u>

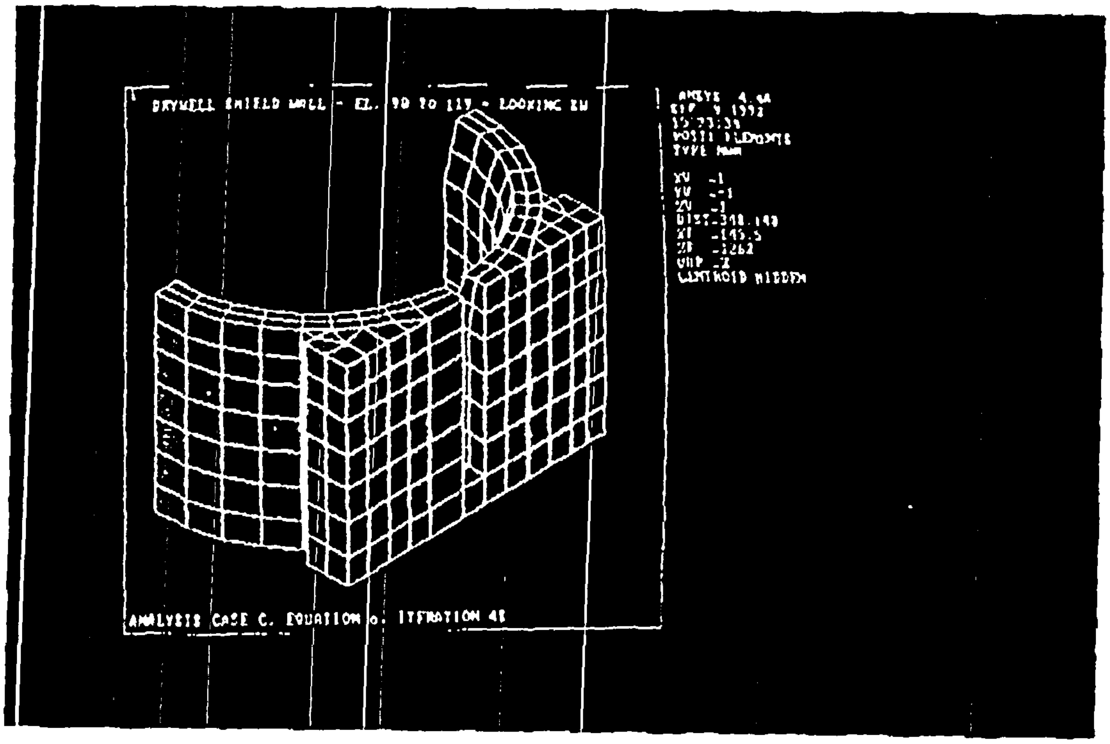


Figure 3.0-2: Vertical Stress (Sz) in the drywell shield wall above elevation 95 ft, Looking Southwest - Load combination c

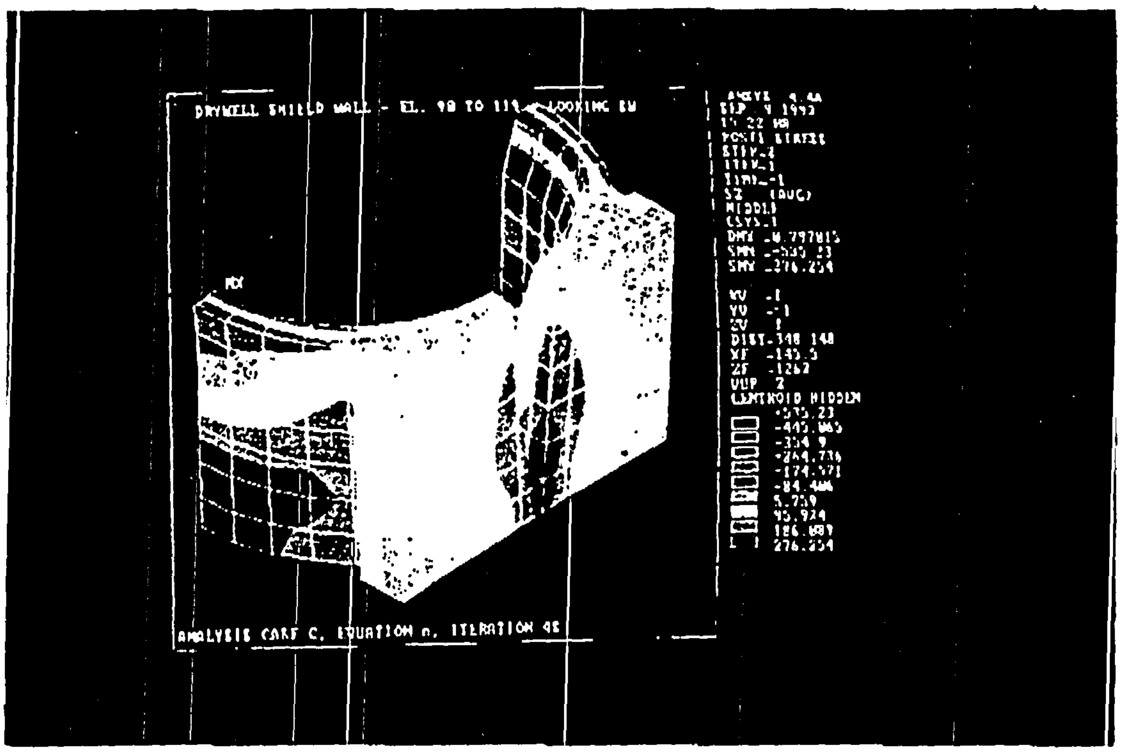


Figure 3.0-3: Vertical Stress (Sz) in the drywell shield wall above elevation 95 ft, Looking Southeast - Load combination c

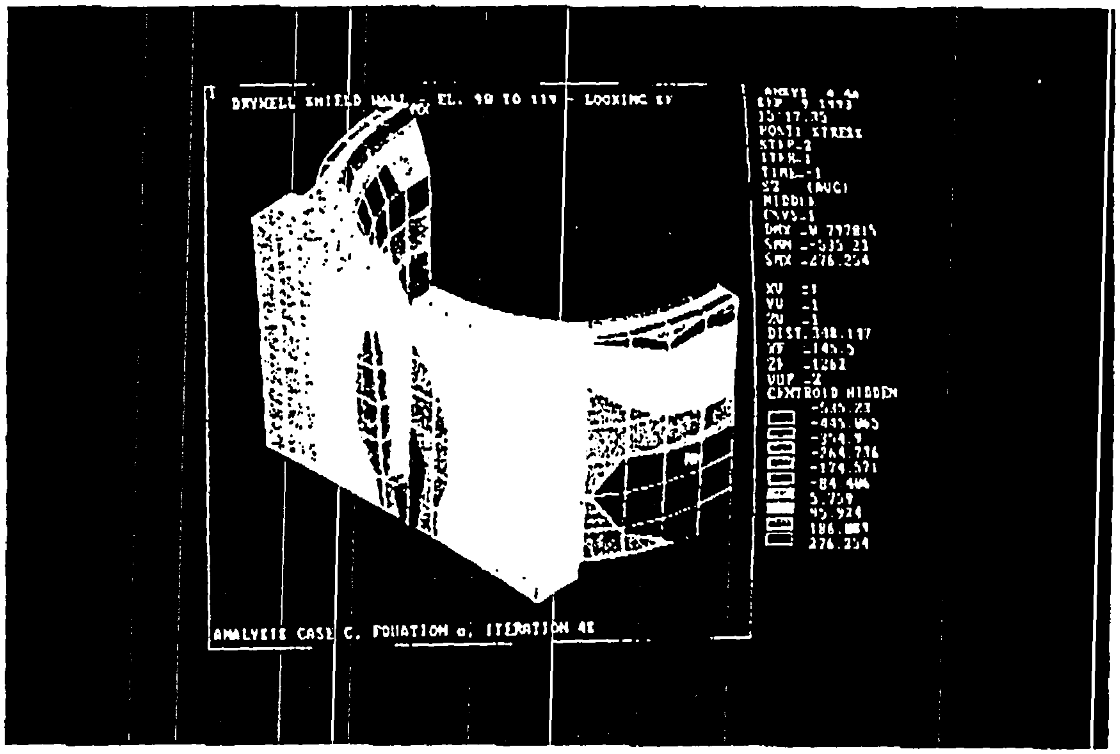


Figure 3.0-4: Vertical Stress (Sz) in the drywell shield wall above elevation 95 ft, Looking North - Load combination c

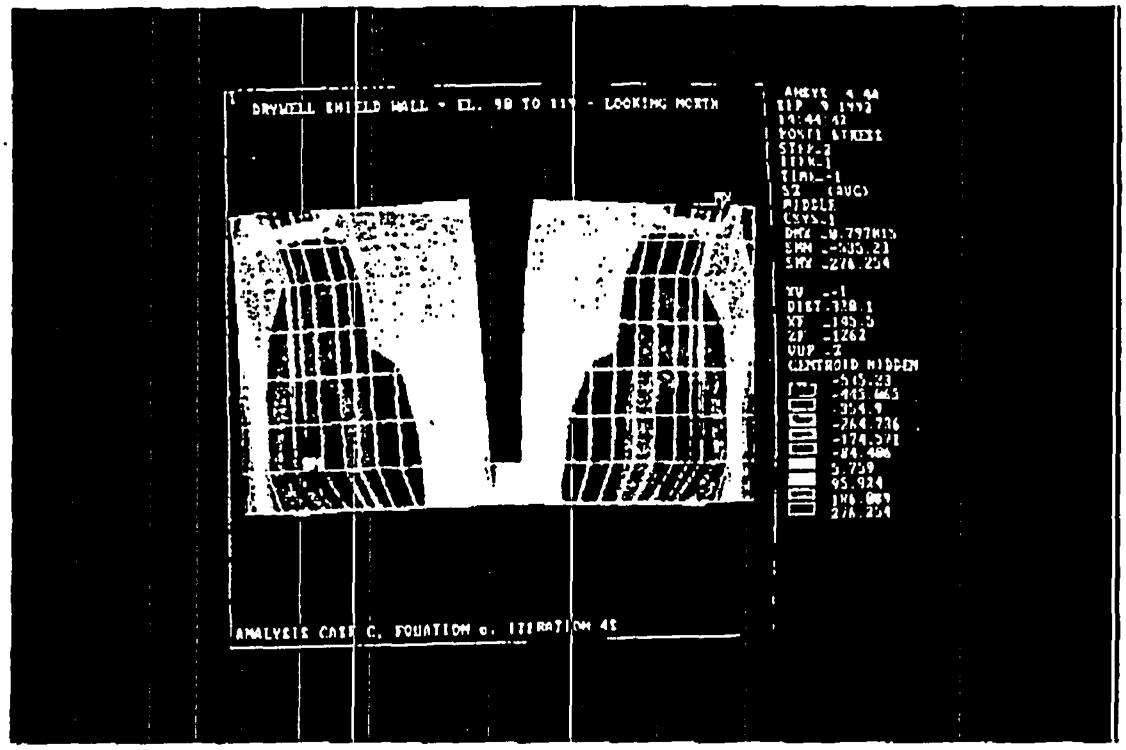


Figure 3.0-5: Circumferential (hoop) Stress (Sy) in the drywell shield wall above elevation 95 ft, Looking Southwest - Load combination c

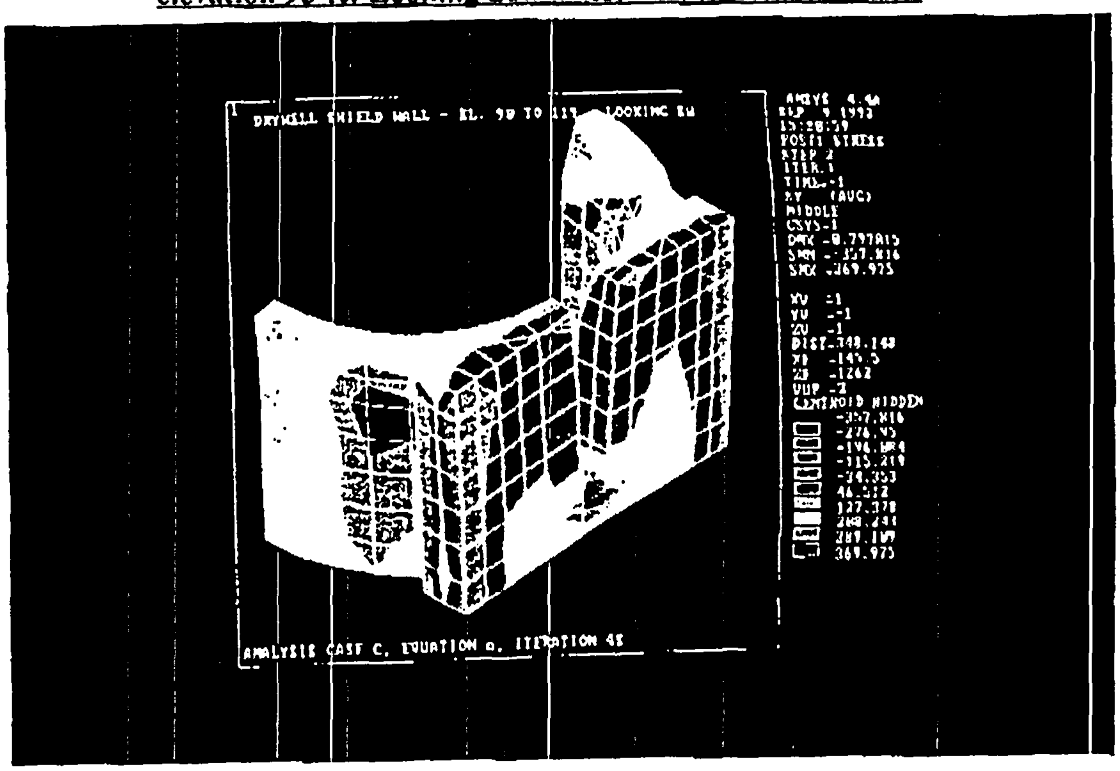


Figure 3.0-6: Circumferential (hoop) Stress (Sy) in the drywell shield wall above elevation 95 ft. Looking Southeast - Load combination c

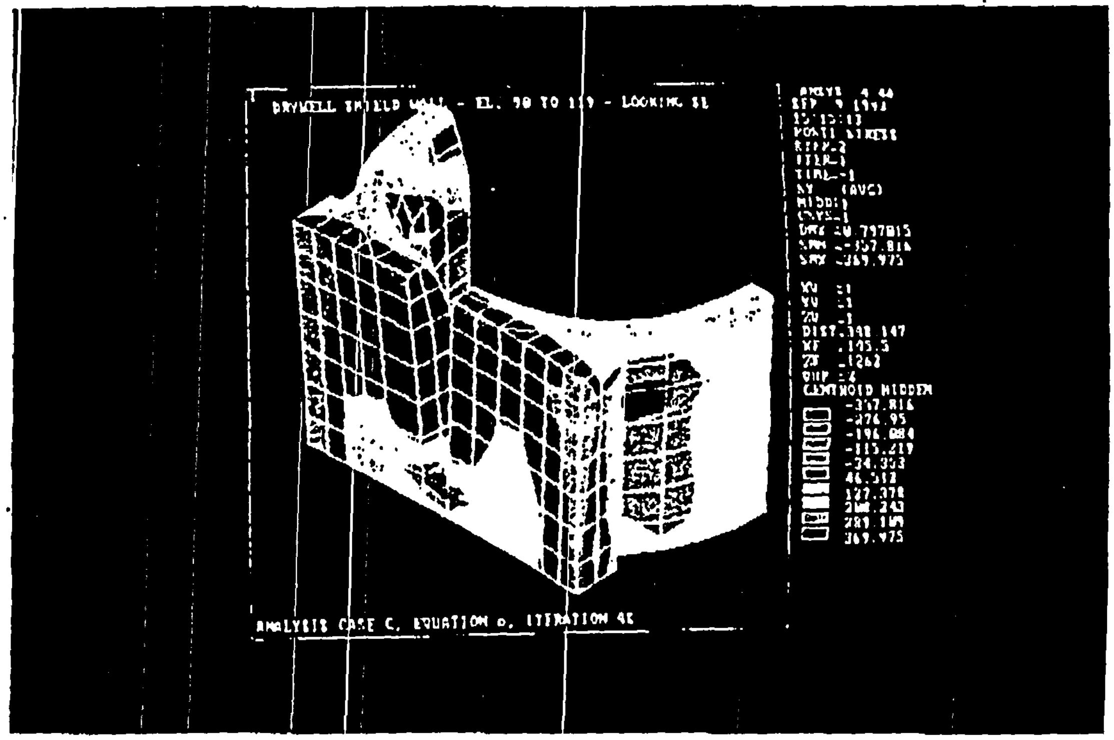


Figure 3.0-7: Circumferential (hoop) Stress (Sy) in the drywell shield wall above elevation 95 ft. Looking North - Load combination c

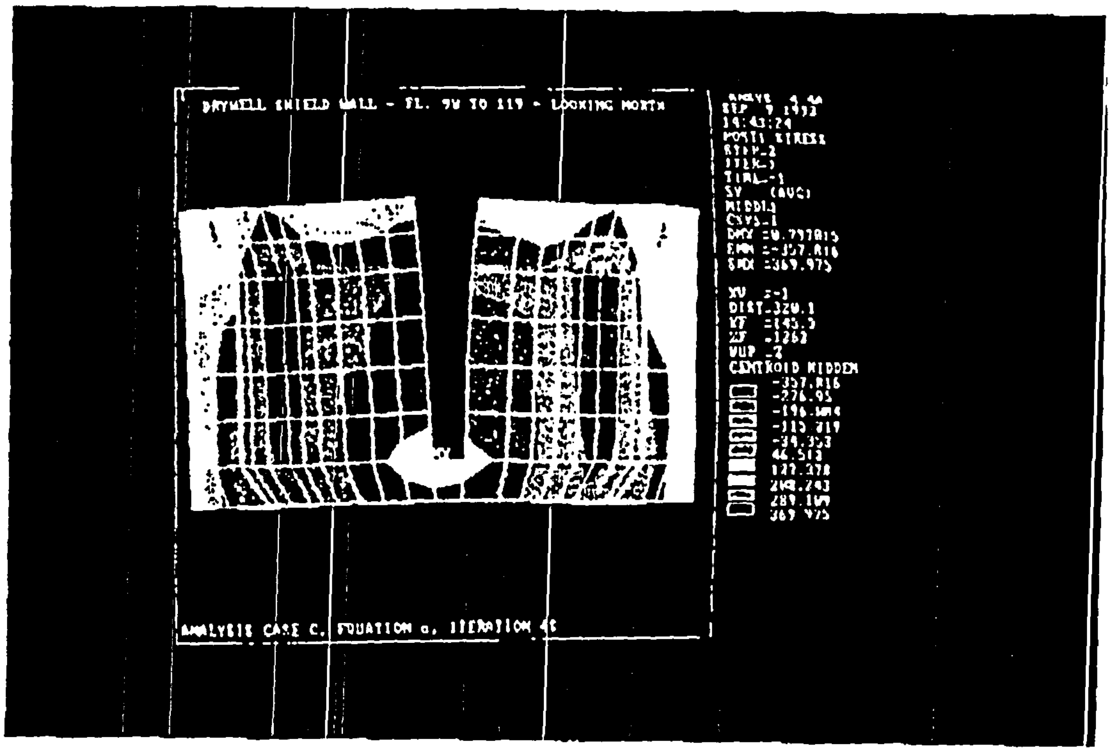


Figure 3.0-8: Vertical Stress (Sz) in the drywell shield wall above elevation 95 ft, Looking Southwest - Load combination d

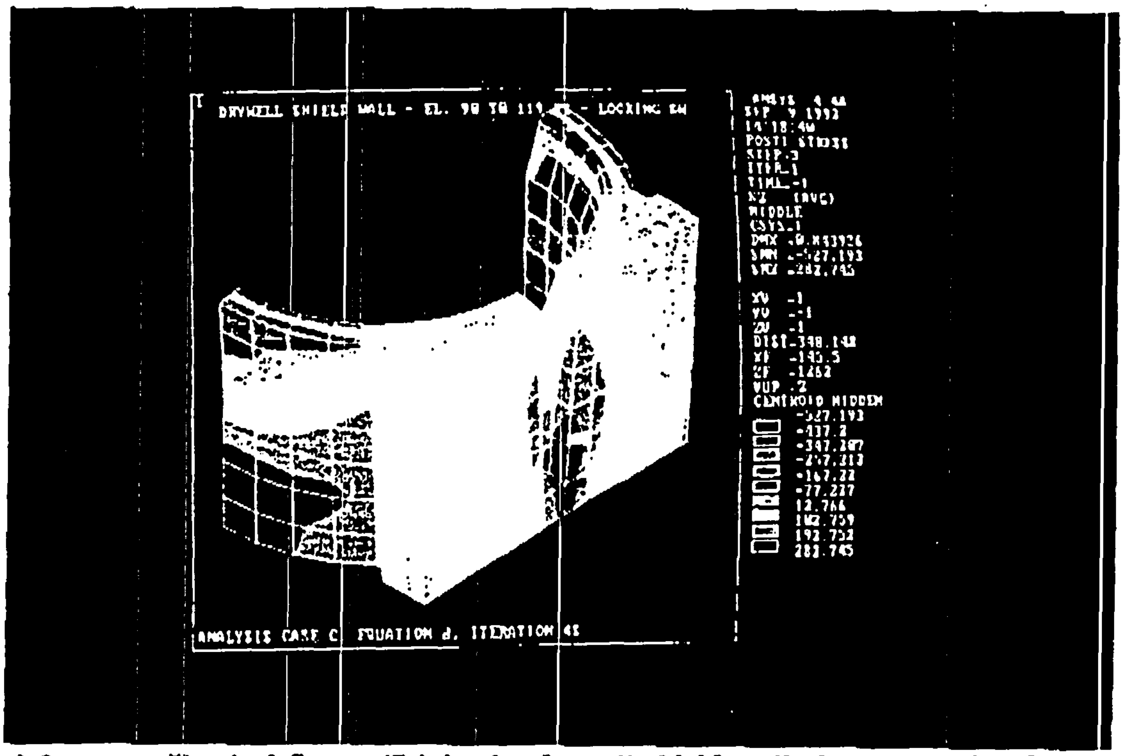


Figure 3.0-9: Vertical Stress (Sz) in the drywell shield wall above elevation 95 ft, Looking Southeast - Load combination d

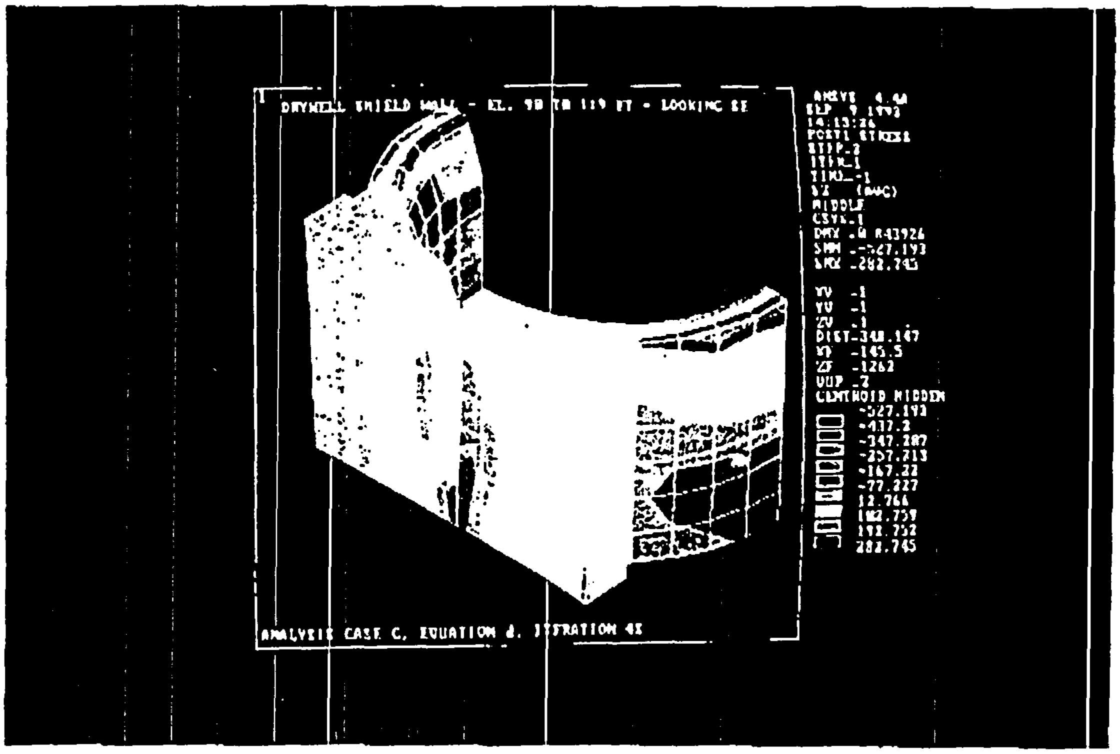


Figure 3.0-10: Vertical Stress (Sz) in the drywell shield wall above elevation 95 ft, Looking North - Load combination d

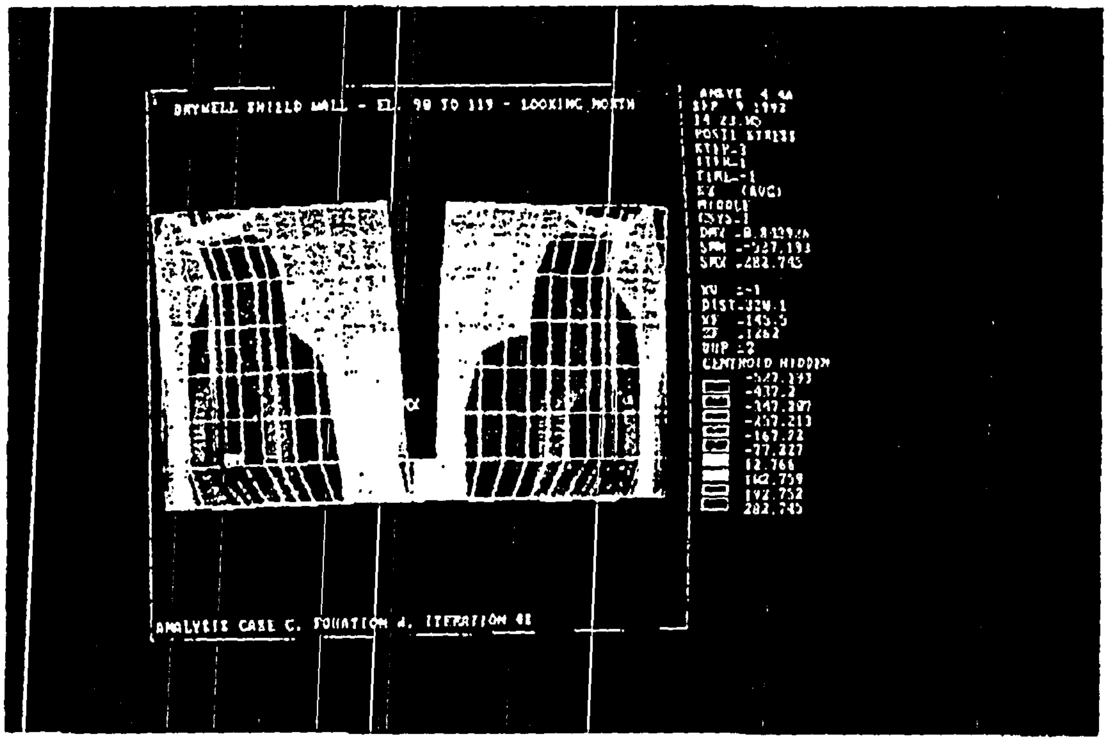


Figure 3.0-11: Circumferential (hoop) Stress (Sy) in the drywell shield wall above elevation 95 ft, Looking Southwest - Load combination d

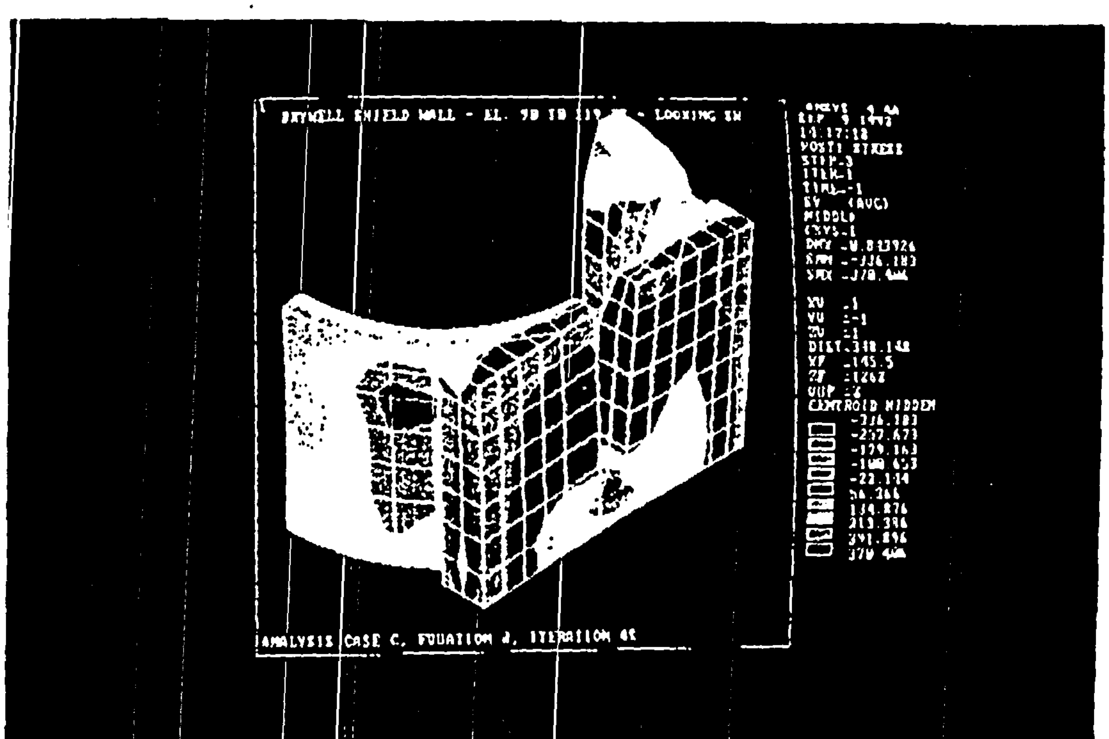


Figure 3.0-12: Circumferential (hoop) Stress (Sy) in the drywell shield wall above elevation 95 ft. Looking Southeast - Load combination d

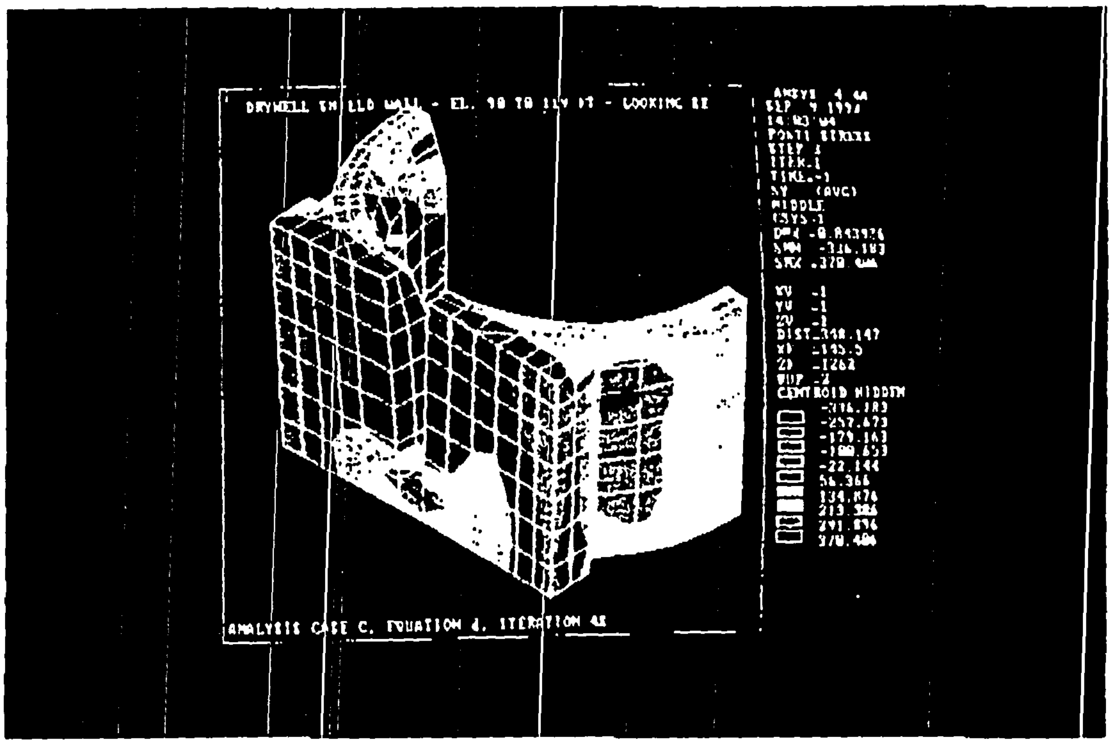


Figure 3.0-13: Circumferential (hoop) Stress (Sy) in the drywell shield wall above elevation 95 ft. Looking North - Load combination d

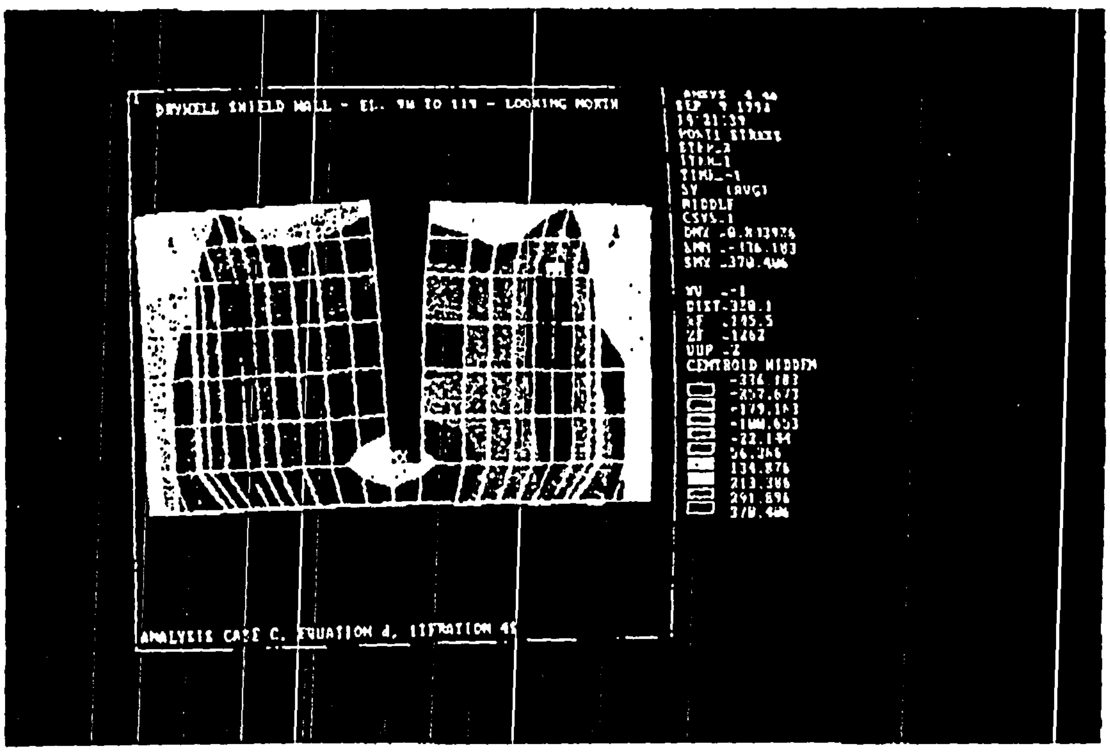


Figure 3.0-14: Location of sections used to evaluate the capacity of the drywell shield wall above elevation 95 ft.

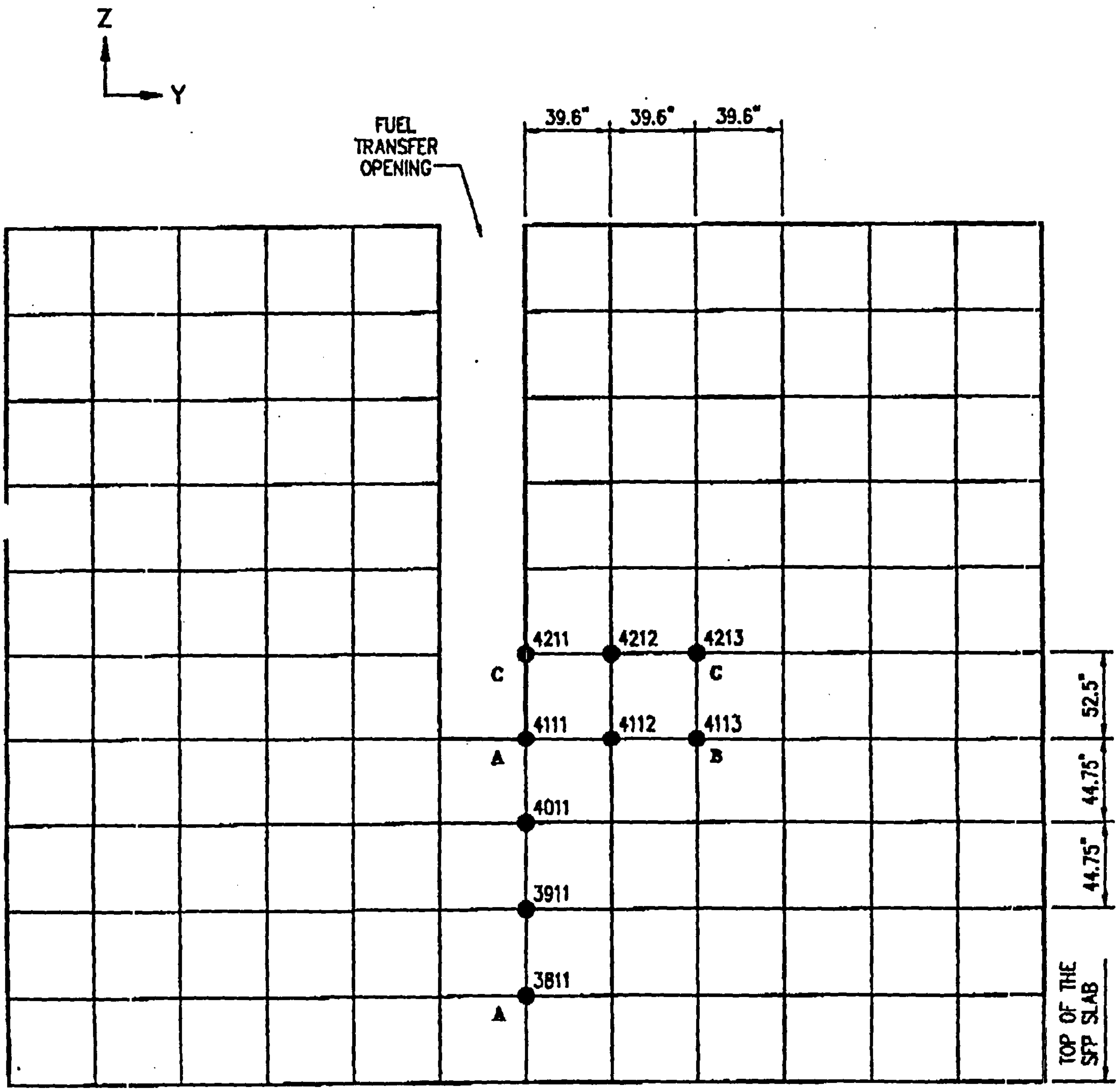
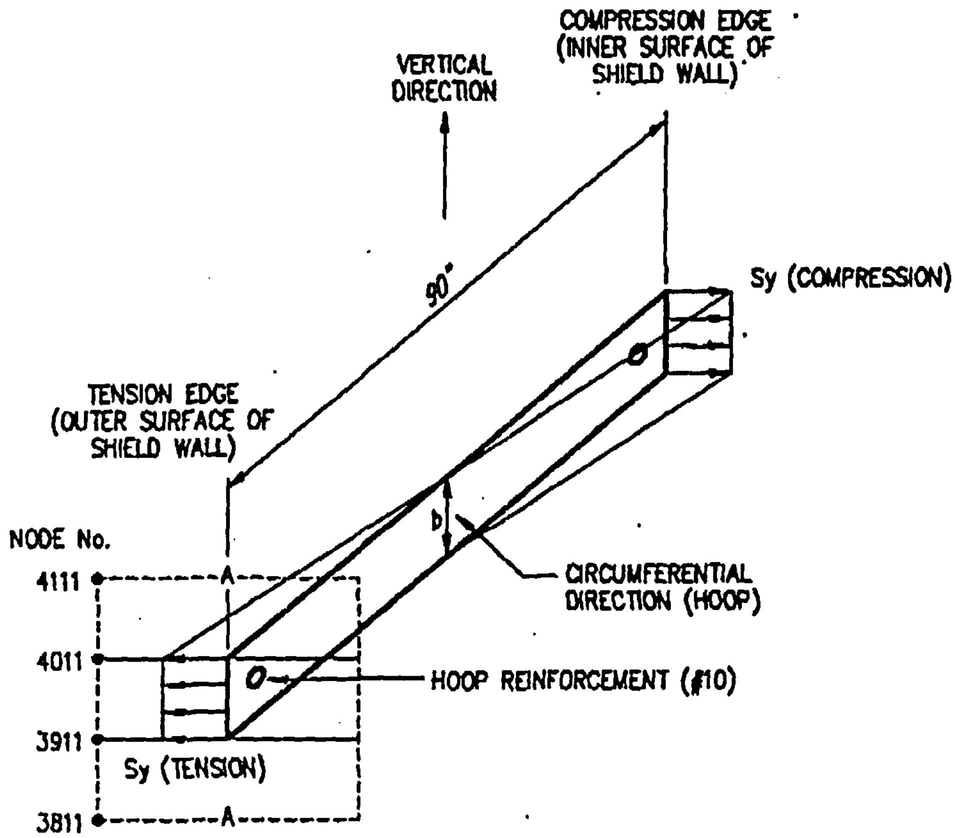
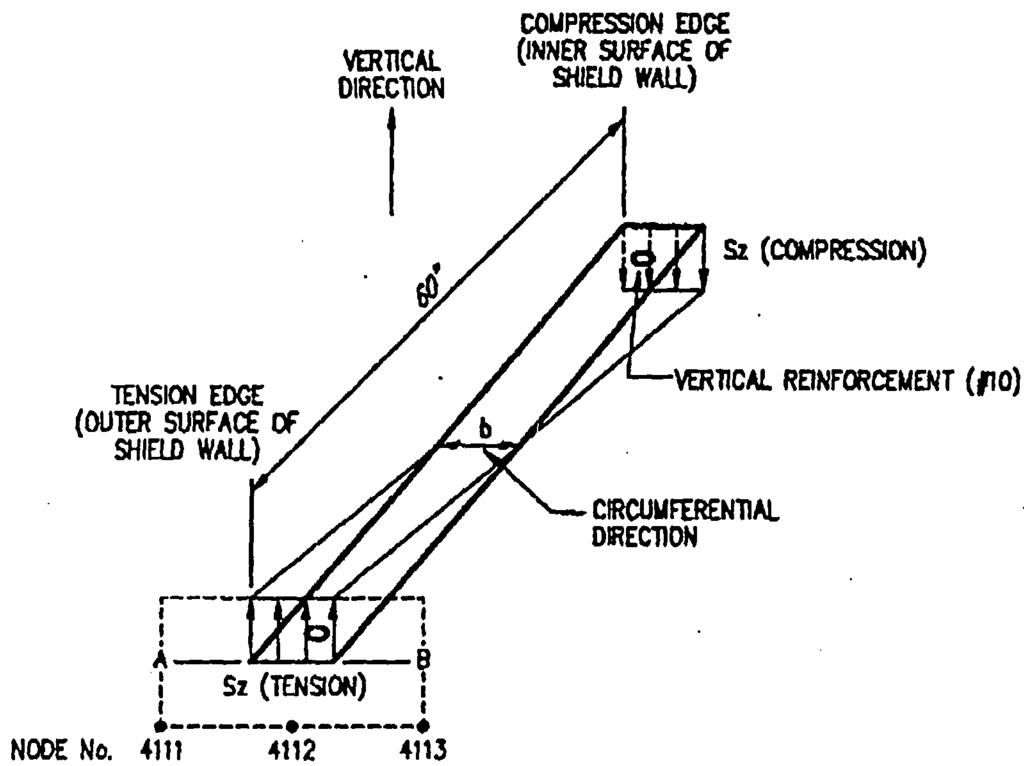


Figure 3.0-15: **Representation of a typical beam cross section corresponding to the average stress on RCBeam Section A-A**



NOTES:
1. DEPTH OF BEAM SECTION: 90".
2. THE HOOP STRESS (Sy) SHOWN ABOVE WAS OBTAINED BY AVERAGING THE HOOP STRESSES FROM THE LINEARIZED STRESSES AT 4111, 4011, 3911, 3811 SHOWN ON FIGURE 3.0-14.
3. RCBEAM SECTION A-A IS IDENTIFIED ON FIGURE 3.0-14.

Figure 3.0-16: **Representation of a typical beam cross section corresponding to the average stress on RCBeam Section A-B and C-C**



NOTES:

1. DEPTH OF BEAM SECTION: 60"
2. THE VERTICAL STRESS (Sz) SHOWN ABOVE WAS OBTAINED BY AVERAGING THE VERTICAL STRESSES FROM THE LINEARIZED STRESSES SHOWN ON FIGURE 3.0-14.

   FOR RCBEAM SECTION A-B, RESULTS FOR NODES 4111, 4112, 4113 WERE USED TO COMPUTE THE AVERAGE Sz FOR RCBEAM.

   FOR RCBEAM SECTION C-C, RESULTS FOR NODES 4211, 4212, 4213 WERE USED TO COMPUTE THE AVERAGE Sz FOR RCBEAM.

Table 3.0-1: Linearized Hoop Stresses S_y (psi) in the South Wall of the Spent Fuel Pool for Load Combination c and Load Combination d. (1) [Ref. 2]

RCBEAM SECTION: A-A (2)

Nodal ID(3)	Load Combination c		Load Combination d	
	Outer Surface	Inner Surface	Outer Surface	Inner Surface
4111	331	22	332	21
4011	157	-112	154	-122
3911	137	-110	136	-115
3811	116	-113	114	-117

Notes:

(1) The stress S_y corresponds to the stress in the circumferential direction in the shield wall.

(2) The sections employed in RCBEAM to determine the capacity of the concrete and the tension in the reinforcing steel are identified in Figures 3.0-14, 3.0-15 and 3.0-16.

(3) A series of linearized stresses were computed to determine the average stress acting on the section for which RCBEAM was used to perform the ACI evaluation. The node number indicated in this column corresponds to the node on the outer surface of the south wall of the spent fuel pool model of the pair of nodes used to determine the linearized stresses at this location. Figures 3.0-14, 3.0-15 and 3.0-16 show the location of the nodes shown in this table.

Table 3.0-2: Linearized Vertical Stresses S_z (psi) in the South Wall of the Spent Fuel Pool for Load Combination c and Load Combination d.[1] [Ref. 2]

RCBEAM SECTION: A-B [2]

Nodal ID[3]	Load Combination c		Load Combination d	
	Outer Surface	Inner Surface	Outer Surface	Inner Surface
4111	261	123	216	96
4112	111	-12	108	-14
4113	57	-70	74	-61

RCBEAM SECTION: C-C

Nodal ID[3]	Load Combination c		Load Combination d	
	Outer Surface	Inner Surface	Outer Surface	Inner Surface
4211	232	9	282	108
4212	147	-45	154	8
4213	98	-79	74	-58

Notes:

(1) The stress Sz corresponds to the stress in the vertical or axial direction in the shield wall.

(2) The sections employed in RCBEAM to determine the capacity of the concrete and the tension in the reinforcing steel are identified in Figures 3.0-14, 3.0-15 and 3.0-16.

(3) A series of linearized stresses were computed to determine the average stress acting on the section for which RCBEAM was used to perform the ACI evaluation. The node number indicated in this column corresponds to the node on the outer surface of the south wall of the spent fuel pool model of the pair of nodes used to determine the linearized stresses at this location. Figures 3.0-14, 3.0-15 and 3.0-16 show the location of the nodes shown in this table.

TABLE 3.0-3: Moment Capacity Margin in the SFP/Drywell Shield wall Analysis Case C. [Ref. 2]

ITER.	EQ.	SECTION OF WALL	AXIAL FORCE[1] (kips)	MOMENT M_u (ft-kips)	MOMENT CAPACITY[2] M_n (ft-kips)	MARGIN %
		South-Wall[3]				
4S	c	A-A	-38	173	192	10
4S	d	A-A	-35	176	206	15
4S	c	C-C	-48	47	105	55
4S	d	C-C	-54	44	91	52

Notes:

(1) A negative axial force produces a tensile axial stress on the slab cross section. The moment capacity is based on the same axial force.

(2) Axial Force, Moment, and Moment Capacity are for a 12" wide section of wall.

(3) The sections employed in RCBEAM to determine the capacity of the concrete section are identified in Figures 3.0-14, 3.0-15 and 3.0-16.

Table 3.0-4: Maximum Tensile Stresses (S_t) in the Reinforcement in the South Wall of the Spent Fuel Pool for Load Combination c and Load Combination d. [Ref. 4]

RCBEAM Section[1]	Load Combination c		Load Combination d	
	S_t (ksi)	S_{yield}(ksi)	S_t (ksi)	S_{yield}(ksi)
A-A	32.8	40	31.4	40
A-B	22.7	40	24.1	40
C-C	27.5	40	30.0	40

Notes:

(1) The sections used to determine the average stresses on a typical beam cross section are identified in Figure 3.0-14, 3.0-15 and 3.0-16.

4.0 CONCLUSIONS

The review of the stresses in the drywell shield wall above elevation 95 ft. indicates that both the concrete and reinforcing steel are within allowables and in compliance with ACI 349, [Ref.6].

5.0 REFERENCES

[1] ABB Impell, "OCNGS Structural Evaluation of the Spent Fuel Pool", Report No. 03-0370-1341, revision 0.

[2] ABB Impell Calculation No. 0370-187-007, "GPUN SFP Analysis Case C Using Cracked Transformed Properties.", Revision 0.

[3] ABB Impell Calculation No. 0037-00196-C002, "GPUN SFP Analysis - Verification of Retrieved Data", Revision 0.

[4] ABB Impell Calculation No. 00037-00196-C003, "GPUN SFP Analysis Case C Additional Stress Information for the Drywell Shield Wall", Revision 0.

[5] ANSYS, Version 4.4, Swanson Analysis Systems, Inc.

[6] ACI 349 - "Code Requirements for Nuclear Safety Related Concrete Structures," 1980.

[7] GPUN Contract No PC-0443867, "Engineering Services for the Oyster Creek Drywell Shield Wall", August 30, 1993.

[8] GPUN, "Specification for Oyster Creek Nuclear Generating Station Spent Fuel Pool Structure Qualification for Consolidated Spent Fuel Storage," SP-1302-53-047, Revision 1.

[9] Burns & Roe Drawing, 4066-3 (As-Built), Revision 3, Reactor Building Cross Section Details.