November 16, 1984

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

Before the Atomic Safety and Licensing Board

In the Matter of	)	
	)	
THE CLEVELAND ELECTRIC ILLUMINATING COMPANY, ET AL.	)	Docket Nos. 50-440 OL 50-441 OL
	)	
(Perry Nuclear Power Plant, Units 1 and 2)	)	

APPLICANTS' VOLUNTARY ANSWERS TO A PORTION OF OCRE'S LATE-FILED THIRTEENTH SET OF INTERROGATORIES TO APPLICANTS (ISSUE #8)

Discovery on Issue #8 has been closed since September 30, 1982. See Tr. 753. On July 30, 1984, OCRE moved to reopen discovery.1/ OCRE attached to its motion to reopen Ohio Citizens for Responsible Energy Thirteenth Set of Interrogatories to Applicants, dated July 30, 1984. As set forth in Applicants' filings dated August 14, 1984,2/ and September 24, 1984,3/ Applicants voluntarily agreed to answer some of the

1/ Motion to Reopen Discovery on Issue #8 (July 30, 1984).

2/ Applicants' Answer to OCRE Motion to Reopen Discovery on Issue No. 8 (August 14, 1984).

3/ Applicants' Further Answer to Ohio Citizens for Responsible Energy Motion to Reopen Discovery on Issue No. 8 (September 24, 1984) ("Further Answer to Motion").

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late-filed discovery requests submitted with OCRE's motion to reopen. Applicants submit the following partial response.[4/]

All documents supplied to OCRE for inspection will be produced for inspection at Perry Nuclear Power Plant ("PNPP"). Arrangements to examine the documents can be made by contacting Mr. Bradley S. Ferrell of The Cleveland Electric Illuminating Company at (216) 259-3737, extension 5520. Applicants will provide copies of any of the produced documents, or portions thereof, which OCRE requests, at Applicants' cost of duplication. Arrangements for obtaining copies can be made with Mr. Ferrell.

RESPONSES

13-22. Identify all penetrations of the containment pressure boundary; for each penetration identified, give:

* * *

(G). whether the penetration was analyzed in the PNPP ultimate structural capacity of Mark III containments report, and if not, why not.

Response:

The selection and identification of penetrations analyzed in the Ultimate Structural Capacity Report are discussed in Sections 4.4.2, 6.2, 6.3, and 6.4, of the Report.

4/ Applicants are still preparing answers to the remaining interrogatories listed at page 2 of Applicants' Further Answer to Motion. Applicants will file answers to the remaining interrogatories listed therein when they are completed.

13-25. Concerning the document entitled 'Ultimate Structural Capacity of Mark III Containments' identified in Applicants' Supplemental Answer to OCRE Interrogatory 5-49, give the date of the document, and supply the names, addresses, employers, and professional qualifications of all persons responsible for its preparation.

Response:

The document referenced in this interrogatory is an undated appendix to an undated draft report entitled "Interim Report on the Hydrogen Control System." The appendix was prepared in March 1983. It is a revision of an earlier appendix to a draft report entitled "Preliminary Report on the Hydrogen Control System," which Applicants previously identified to OCRE in response to Interrogatory No. 5-51 of OCRE's Fifth Set of Interrogatories to Applicants. Applicants provided a copy of the latter report to Ms. Hiatt by letter dated February 25, 1983. The document referenced in this interrogatory is being reviewed by CEI, and has not been finalized for formal submission to the NRC Staff.

The referenced document was prepared by R. Alley, R. Schmehl, and S. Iyengar of Gilbert Commonwealth Inc. Applicants previously supplied Mr. Alley's resume by letter to Ms. Hiatt dated February 25, 1983. Resumes of Mr. Schmehl and Mr. Iyengar are attached.

13-29. Have Applicants in their analysis of containment capacity considered the variation of material properties with the temperatures associated with hydrogen combustion? If so, identify all such analyses. If not, why not?

Response:

Before deciding whether, or the extent to which, it may be necessary to analyze "variation of material properties with the temperatures associated with hydrogen combustion," CEI must first finalize a temperature time history for hydrogen combustion. Such a temperature time history has not been finalized.

13-32. Identify any study, evaluation, calculation, or analysis performed by of [sic] for Applicants to determine the degree of leakage from electrical penetrations, vacuum breakers, purge/vent valves, hatches, and airlocks due to the pressures and temperatures resulting from hydrogen combustion.

Response:

Potential leakage from the equipment hatch was considered as described at page 22 of the Report. No other study, evaluation, calculation, or analysis as described in this interrogatory has been performed.

13-33. Give the value of each variable in the equations on pp. 10 and 11 of Ultimate Structural Capacity of Mark III Containments report used to solve said equations, and explain how these values were obtained.

Response:

The values are as follows:

F_1 = applied dynamic pressure. F_1 varies as shown in tables 7A and 7B

K = 87.2 psi/inch for knuckle
= 95.2 psi/inch for cylinder
= 14.7 psi/inch for apex

$$T_{el}=[\cos^{-1} - (\frac{Y_{el}}{Y_{st}} - 1.0)]/W$$

W = 83.2 rad/sec. for knuckle, cylinder and apex

t_d = 100.0 sec

Y_{el} = 0.945 for average kunckle
= 1.25 for average cylinder
= 7.293 for average apex
= 0.78 for lower bound kunckle
= 1.01 for lower bound cylinder
= 6.025 for lower bound apex

$$Y_{st} = \frac{F_1}{K}$$

$$M = \frac{F_1}{W^2 Y_{st}}$$

R_m = 82.4 psi for average knuckle
= 119.5 psi for average cylinder
= 107 psi for average apex
= 68.0 psi for lower bound knuckle
= 96.25 psi for lower bound cylinder
= 68.4 psi for lower bound apex

t_m = calculated from equation

$$0 = (F_1 - R_m)\frac{t_m}{M} - \frac{F_1 t_m^2}{2 t_d M} + Y_{el}$$

The values were obtained as follows:

1. F is the dynamic pressure applied to the containment vessel, which produced the deflection and ductility ratios presented in Tables 7A and 7B of the Ultimate Structural

Capacity Report. Tables 7A and 7B provide the results of a parametric study in which the pressure, F, was varied and the deflections and ductility ratios were calculated by the equations documented in the Report. The criteria used to determine the value of the maximum pressure F, are based on the ductility ratios specified in Appendix A to NRC Standard Review Plan, Section 3.5.3.

2. The stiffness, K, is obtained from a unit pressure load case utilizing the KSHEL (linear, elastic) computer program and the model presented in Figure 1 of the Report. The stiffness is calculated by dividing the calculated deflection of a particular point on the containment vessel model by the magnitude of the applied pressure.

3. T_{el} is calculated as shown, using values for W, Y_{el} and Y_{st} (defined below).

4. The value for W was the result of a frequency analysis of the containment vessel performed by Newport News Industrial Corporation.

5. t_d, the duration of the pressure transient, was obtained from an analysis which considered a conservative quantity of hydrogen from the zirconium reaction in the active region of the fuel rods. This hydrogen was postulated to be released to the containment atmosphere.

6. The values of Y_{el} are obtained from the computer analysis described in item 2 above. The unit pressure computer analysis and the resulting containment vessel stresses and

deflections were utilized along with the distortion energy yield criterion (see section 4.1 of the Report) to calculate the vessel deflection, Y_{el}, at a particular point due to the internal pressure, corresponding to a state of membrane yielding in the vessel.

7. Y_{st}, the static deflection of the containment vessel, is calculated as shown, using values for F and K.

8. M, the calculated mass, is calculated as shown, using values for F, W, and Y_{st}.

13-34. Identify all sources of uncertainty in all of the assumptions, judgements, calculations, and models employed in the Ultimate Structural Capacity of Mark III Containments report, and explain what effect they have on the results and conclusions therein.

Response:

The Ultimate Structural Capacity Report does not discuss uncertainty per se. Instead, the authors of the Report used conservative assumptions and judgments in the Reports' calculations and analyses. The use of such conservatisms provided lower bound results for the internal pressure capacity.

For example, the stress-strain relationship used for the plastic analysis of penetration number P 205 is based upon the nominal properties of the material. The actual properties are greater than those used for the analysis. Therefore the actual capacity of the containment vessel based upon penetration number P 205 is greater than that predicted by the analysis.

Other examples of this conservative approach include the analyses of the containment vessel, the analyses of the air lock and equipment hatch, and the analyses of the lower containment vessel penetrations. These analyses are based on linear elasticity. Typically, after yield, the plastic and strain hardening characteristics of the material would permit additional pressure capacity. Thus, analyses based only on linear elasticity produce lower bound results.

13-35. Did the analysis of structural capacity include the effects of deficiencies in construction and fabrication of the containment vessel? If so, explain how these effects were considered. If not, why not?

Response:

The Ultimate Structural Capacity Report is based on current design values for the containment vessel. These values include any impacts of construction or fabrication deficiencies that have been identified. Under Applicants' program all such deficiencies have been analyzed, and corrected where necessary, to assure that the containment design requirements are met. To this extent the design values used in the Ultimate Structural Capacity report "include the effects of" any such deficiencies.

13-40. Did Applicants in their ultimate structural capacity report consider the effects of any changes in material properties or the creation of residual stresses resulting from welding of the containment vessel? If so explain how they were accounted for. If not, why not?

Response:

Applicants "accounted for" potential material impacts from welding by applying the appropriate ASME and AWS standards at the time the welding was performed. These code standards, e.g., standards governing pre-heat and post-weld heat treatment, minimize residual stresses in materials which may result from the welding activity. Thus, it was not necessary separately to consider potential impacts from welding as part of the analyses in the Ultimate Structural Capacity Report.

13-41. Demonstrate that the calculations and methodology employed in the Ultimate Structural Capacity Report are in accordance with provisions of the ASME Code, Section III.

Response:

The ASME Code, Section III, does not specify requirements for the calculations or methodology to be used in an ultimate structural capacity analysis. However, the analysis in the Ultimate Structural Capacity Report did use ASME Code service limits as a conservative basis for calculating the ultimate capacity of the containment. See Ultimate Structural Capacity Report, Section 1.

13-44. Explain and supply the basis for all the following statement appearing on p. 6 of the Ultimate Structural Capacity Report: "Since the yielding in the knuckle occurs only at one point along the meridian, the pressure can be increased above 68.0 psig to 78.0 psig, the level at which hoop buckling occurs in the knuckle."

Response:

The increase from 68.0 psig to 78.0 psig, discussed in the referenced material, is explained by the ductility and the additional strength of the steel in the area in question. The ductility of the steel and the additional strength of the steel allow the forces in the area of the containment vessel which has exceeded the yield stress to be redistributed to the surrounding area which is below the material yield stress.

13-45. Explain and supply the basis for the statements at p. 7 of the Ultimate Structural Capacity Report that local areas at discontinuities having stresses exceeding the yield stress will not affect vessel integrity because the stresses are only on the inside surface of the vessel.

Response:

The interrogatory omits a key statement at page 7 of the report, namely, that "the stresses at the same location on the <u>outside</u> surface of the containment are <u>below</u> the yield stress" (emphasis added). Because the above yield stress in these areas of the containment are limited to the inside surface membrane, this type of stress constitutes "secondary stress" within the meaning of the ASME Code, which states:

> The basic characteristic of a secondary stress is that it is self-limiting. Local yielding and minor distortions can satisfy the conditions which cause the stress to occur, and failure from one application of the stress is not expected.

The secondary stresses at these discontinuities are well within the ASME acceptance criteria. In addition, the average stresses across the thickness of the plates in these areas are much less than the yield stress of the steel. For these reasons, the vessel integrity is not affected by the local secondary stresses referenced at page 7 of the Report.

13-46. Do Applicants consider the pressures in parentheses in tables 6A and 6B (some of which are quite low, e.g. main steam penetration) to be the controlling pressures for the containment? Explain why or why not.

Response:

No. The results of additional detailed analyses, which reflect the strength of the penetrations, and controlling pressures, are summarized in Section 6.4 and Table 12 of the Report.

13-47. Explain the basis for the following assertions appearing on p. 9 of the Ultimate Structural Capacity Report:

(a) Initial yield pressures can be increased if the plastic zone is limited to one radius from the penetration sleeve. Specifically explain how such limitation of the plastic zone can be assured.

(b) It is expected that the vessel strains resulting from one radius yield region around penetrations would not result in objectionable distortions. Define objectionable distortions, with reference to proper authority, and explain the basis for your expectation.

Response:

The stresses calculated for the penetrations in Section 4.4.2 of the Report are based upon a stress concentration approach. Stresses of this type are classified as peak stresses by the ASME Code. The ASME Code states that basic characteristic of a peak stress is that it does not cause any noticeable distortion. Even if local areas around penetrations attain the yield stress of the steel, as long as the containment vessel membrane stresses which occur at points away from discontinuities, such as in the location of penetrations, are at less than yield stress, the capability for the redistribution of forces in the vessel shell is present.

Detailed information about the penetrations is provided in Section 6.0 of the Report.

13-48. Were hydrodynamic loads resulting from hydrogen combustion considered in the analysis of the lower containment penetrations? If not, why not?

Response:

No. The ultimate capacity of the containment is based on a pressure type of loading and is independent of whether these pressures are hydrodynamic-related.

13-49. Identify all deficiencies associated with the inclined fuel transfer tube and penetration. Indicate which of these deficiencies have not been corrected, and for each uncorrected deficiency identified, explain whether it had been considered in the analysis of the fuel transfer penetration in the Ultimate Structural Capacity Report, and if not, why not.

Response:

There are no nonconformances associated with the inclined fuel transfer tube or penetrations which involve the containment pressure boundary. The nonconformances are therefore irrelevant to the analyses contained in the Report.

13-50. * * *

(b) Indicate whether the defect associated with Westinghouse class 1E electrical penetrations has been considered in the analysis of containment capacity. If not, why not?

Response:

No. The identified condition was subsequently brought into compliance with the ASME Code. See response to Interrogatory No. 13-35.

Respectfully submitted,

SHAW, PITTMAN, POTTS & TROWBRIDGE

By: Harry H. Glasspiegel
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Harry H. Glasspiegel

Counsel for Applicants

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(202) 822-1000

Dated: November 16, 1984

RICHARD J. SCHMEHL
Structural Engineer

Experience in structural engineering activities involving steel and concrete design for major power generating facilities.

EXPERIENCE:	GILBERT/COMMONWEALTH since 1976	
1981 to Present	Structural Engineer - Responsible for determining the ultimate internal pressure capacity of the containment vessel for Cleveland Electric Illuminating Company's Perry Nuclear Power Plant, Units 1 and 2. Also responsible for the preparation of answers to Nuclear Regulatory Commission Final Safety Analysis Report questions regarding buckling of the Containment Vessel. Responsible for evaluating the effect of Containment Vessel design changes on the Annulus Concrete. Responsible for the dynamic analysis of Reactor Building steel platforms for the LOCA related loads caused by suppression pool encroachment. Also responsible for the dynamic analysis of the Reactor Building steel platforms and pipes supported from the platform. Responsible for the coordination of the Perry Nuclear Power Plant New Loads Adequacy Evaluation program.	
1981	Structural Engineer - Responsible for the preparation of design criteria for the seismic evaluation of the Auxiliary Building east bracing and the Turbine Building southeast bracing and for the structure seismic upgrading program. Both of the assignments were for the Rochester Gas and Electric Corporation's R. E. Ginna Nuclear Power Station.	
1980-81	Structural Engineer - Responsible for the design of the Annulus Concrete, located between the Containment Vessel and the Shield Building, and the review of the shield building design for loads caused by the addition of the Annulus Concrete for the Cleveland Electric Illuminating Company's Perry Nuclear Power Plant, Units 1 and 2.	
1979-80	Structural Engineer - Responsible for the seismic analysis of the auxiliary structures comprising the Rochester Gas and Electric Corporation's R. E. Ginna Nuclear Power Station, 490 MW.	
1978-79	Structural Engineer - Responsible for the analysis of the Reactor Building for Safety Relief Valve Discharge for the Cleve and Electric Illuminating Company's Perry Nuclear Power Plant, Units 1 and 2, 1200 MW each.	
1978	Structural Engineer - Responsible for the design of a steel spherical containment vessel for a containment study for Mitsubishi International. Also provided reinforcing estimates for various shield building configurations.	

Structural Engineer - Responsible for the design review, according to U.S. criteria, of a steel spherical containment vessel designed to German criteria and a skewed Residual Heat Removal penetration for Kraftwerk Union, AG.

Structural Engineer - Responsible for providing loads and load combinations for a report on containment vessel design of a Boiling Water Reactor for Houston Lighting and Power Company.

1976-78 Structural Engineer - Responsible for the seismic design of cable tray supports for South Carolina Electric & Gas Company's V.C. Summer Station, Unit 1, 900 MW; and the Cleveland Electric Illuminating Company's Perry Nuclear Power Plant, Units 1 and 2.

1972-76 United Engineers and Constructors, Inc., Philadelphia Pennsylvania

1974-76 Design Engineer - Designed concrete and steel structures for the Public Service Company of New Hampshire's Seabrook Nuclear Power Plant.

1972-74 Performed the seismic analysis of buildings for the Seabrook Nuclear Station.

EDUCATION: B.S.C.E., The Pennsylvania State University, 1972
Probability and Statistics for Civil Engineers, University of Pennsylvania

REGISTRATION: Professional Engineer in Pennsylvania (1977)

SOCIETIES: American Society of Civil Engineers
American Concrete Institute

SAMPATH N. S. IYENGAR
Senior Structural Research Engineer

Practical experience in structural analysis and design involving major nuclear power generating facilities; and teaching experience in structural analysis, design and computer applications.

EXPERIENCE:	GILBERT/COMMONWEALTH since 1974	
1974 to Present	Review of the dynamic analysis of the Perry Reactor Building to investigate responses of the attached points of the Hydraulic Control Units on the steel platform. The loading included hydrodynamic and seismic effects.	

Review of seismic qualification of electrical equipment on V. C. Summer project.

Analysis and design of pipe rupture restraints for the V. C. Summer Nuclear Power Plant, Unit 1, 900 MW.

Analysis of masonry walls of Ginna and CR3 nuclear power plants for the as-is and as-fixed conditions, pursuant to NRC Bulletin 80-11 with relevance to applicable seismic criteria.

Transport of programs SAP4 and TPIPE from the CDC machine system to the CRAY machine system and optimization of the program using vectorization features of the CRAY system.

Analysis of the Perry Reactor Building for new loads (NLAE) with the proposed concrete fill in the annulus between the steel containment and concrete shield walls.

Verification of a computer program to solve slab, wall and mat problems with potential for applications in power plant design. Modification of a computer program for dynamic stress analysis of axisymmetric structures; research and development activities of a general nature in structural design as applied to nuclear power plants.

Design of a missile shield on top of the reactor to contain postulated missiles consequent to an accident; ductwork qualification involving pressure or suction resistance and equivalent static seismic loads; and design of ductwork stiffeners for the V. C. Summer Plant.

Investigation, by comparison with test results, of structural adequacy to resist postulated tornado-borne missiles for Perry and V. C. Summer Plants.

Preparation of structural specifications for cooling towers for The Cleveland Electric Illuminating Company's Perry Nuclear Power Plant, Units 1 and 2, 1200 MW each.

(Continued)

Structural investigation of postulated fuel cask drops in nuclear power plants for Metropolitan Company's Three Mile Island Nuclear Station, Unit 1, 871 MW; The Electric Utilities of Croatia and Slovenia's KRSKO Nuclear Power Plant, Unit 1, 600 MW; and South Carolina Electric & Gas Company's Virgil C. Summer Nuclear Station, Unit 1, 900 MW.

1966-74 Lehigh University, Bethlehem, Pennsylvania
1973-74 Assistant Professor - Taught courses in steel and concrete structures and computer programming.

1966-73 Teaching Assistant, Instructor - Assisted in courses on numerical methods and taught a course in computer programming.

1964-66 Washington State University, Pullman, Washington
Teaching Assistant - Assisted in steel design courses.

1953-64 Government of Maharashtra, Bombay, India
Deputy Engineer, Public Works and Lecturer, Department of Technical Education - Held independent charge of public works including roads and buildings, and taught courses mainly in structural analysis and design at undergraduate level.

EDUCATION: B.Sc., University of Mysore, India, 1948
B.E. (Civil), University of Poona, India, 1953
M.S. in C.E., Washington State University, 1966
Ph.D., Lehigh University, 1973
Additional Courses:
71 ACI Code and 73 Handbook, Drexel University, 1974
Nuclear Power Plant Design, Gilbert Associates, Inc., 1975
Speakeasy Computer Program, Gilbert Associates, Inc., 1977

REGISTRATION: Professional Engineer - Pennsylvania (1975)

SOCIETIES: Honor Society of Phi Kappa Phi
Honor Society of Sigma Xi

PUBLICATIONS: Co-author, "Strength and Ductility of A572 (Grade 65) Steel Structures," presented at the Tenth Congress of the International Association of Bridge and Structural Engineering, Tokyo, Japan, September, 1976.

STATE OF PENNSYLVANIA)
:
COUNTY OF BERKS)

AFFIDAVIT

ROGER W. ALLEY, being duly sworn according to law, deposes and says that he is Project Engineer - Structural, Perry Project, of Gilbert Associates, Inc. and that the facts set forth in the foregoing Applicants' Answers to Ohio Citizens for Responsible Energy Interrogatories 13-22, 13-25, 13-29, 13-32, 13-33, 13-34, 13-35, 13-40, 13-41, 13-44, 13-45, 13-46, 13-47, 13-48, 13-49, 13-50,

dated November 16, 1984, are true and correct to the best of his knowledge, information, and belief.

Roger W. Alley

Sworn to and subscribed
before me this 15th day
of November, 1984.

Barbara J. Moreadith

NOTARY PUBLIC
My Commission Expires March 1, 1986
BERKS COUNTY, READING, PA.

November 16, 1984

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

Before the Atomic Safety and Licensing Board

In the Matter of	)	
	)	
THE CLEVELAND ELECTRIC ILLUMINATING COMPANY, ET AL.	)	Docket Nos. 50-440 50-441
	)	
(Perry Nuclear Power Plant, Units 1 and 2)	)	

CERTIFICATE OF SERVICE

This is to certify that copies of the foregoing "Applicants' Voluntary Answers to a Portion of OCRE'S Late-Filed Thirteenth Set of Interrogatories to Applicants (Issue #8)" were served by deposit in the United States Mail, first class, postage prepaid, this 16th day of November, 1984, to all those persons on the attached Service List.

Harry H. Glasspiegel

HARRY H. GLASSPIEGEL

Dated: November 16, 1984

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of THE CLEVELAND ELECTRIC ILLUMINATING COMPANY (Perry Nuclear Power Plant, Units 1 and 2)	)))))))	Docket Nos. 50-440 50-441

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