UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

Before the Atomic Safety and Licensing Board

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

OCRE RESPONSE TO "APPLICANTS' INTERROGATORIES AND REQUEST FOR PRODUCTION OF DOCUMENTS TO INTERVENOR OHIO CITIZENS FOR RESPONSIBLE ENERGY (SECOND SET)"

Intervenor Ohio Citizens for Responsible Energy ("OCRE") hereby files its response to Applicants' Second Set of Interrogatories, dated September 22, 1982. In order to conserve its scarce resources, OCRE will not reproduce herein the interrogatories propounded it; the interrogatories are answered in the same sequence and numeration encountered.

OCRE will not produce herewith the documents identified in these responses, as requested by Applicants, since most of these documents are publicly available and the production of same by OCRE would be too great a burden on its limited resources. If Applicants are unable to obtain any document otherwise, OCRE will provide a copy at a cost of $0.10 per page plus postage.

ISSUE #8

1. No such persons have yet been identified by OCRE.

DS03

2. No witnesses have been identified as yet.

3. Final Safety Analysis Report for the Perry Nuclear Power Plant, Sections 6.2.5 and 7.3.1.

   Letter, dated September 16, 1982, from A. Schwencer, NRC, to D. Davidson, CEI, re Request for Additional Information Regarding Degraded Core Hydrogen Control for PNPP.

   Proposed Rule to 10 CFR Part 50, "Interim Requirements Related to Hydrogen Control" 46 Fed Reg 62281, December 23, 1981.

   Final Rule to 10 CFR Part 50, "Interim Requirements Related to Hydrogen Control" 46 Fed Reg 58484, December 2, 1981.

   SECY-80-107, February 22, 1980, "Proposed Interim Hydrogen Control Requirements for Small Containments"

   SECY-80-107A, April 22, 1980, "Additional Information Re Proposed Interim Hydrogen Control Requirements"

   SECY-80-107B, June 20, 1980, "Additional Information Re Proposed Interim Hydrogen Control Requirements"

   Regulatory Guide 1.7, Revision 2 (November 1978), "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident"

   Branch Technical Position CSB 6-2, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident"

   SECY-81-245, April 17, 1981, "Interim Amedments to 10 CFR Part 50 Related to Hydrogen Control and Certain Degraded Core Considerations"

"Pressure and Temperature Transients Resulting from Postulated Hydrogen Fires in Mark III Containments" by Mark P. Paulsen and John O. Bradfute, Energy Incorporated, EI 75-4, February 1975.

NUREG/CR-1659 Vol. 4, "Reactor Safety Study Methodology Applications Program: Grand Gulf #1 BWR Power Plant" October 1981.

NUREG-0626, "Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in GE-Designed Operating Plants and Near-Term Operating Licensing Applications" February 1980.

NUREG/CR-2540, "A Method for the Analysis of Hydrogen and Steam Releases to Containment During Degraded Core Cooling Accidents" February 1982.

NEDO-10812, "Hydrogen Flammability and Burning Characteristics in BWR Containments" General Electric, April 1973.

NUREG/CR-0913, "Generation of Hydrogen During the First Three Hours of the Three Mile Island Accident" July 1979.

NUREG/CR-1575, "Hydrogen Mixing in a Closed Containment Compartment Based on a One-Dimensional Model with Convective Effects" September 1980.

NUREG/CR-1561, "The Behavior of Hydrogen During Accidents in Light Water Reactors" August 1980.

NUREG/CR-2017, "Proceedings of the Workshop on the Impact of Hydrogen on Water Reactor Safety" Volumes 1-4, September 1981.

NUREG/CR-1250, "Three Mile Island: A Report to the Com-

missioners and to the Public" Volume II, by the NRC Special Inquiry Group, M. Rogovin, Director.

"Containment of a Reactor Meltdown" by Jan Beyea and Frank Von Hippel, The Bulletin of the Atomic Scientists, Vol. 38, No. 7, August/September 1982, pp. 52-59.

Transcript of May 27, 1982 meeting to discuss concerns of John Humphrey re Mark III containment, pp. 174-177, 181-185, 202-205, 257-260.

IE Bulletin 79-08, April 14, 1979, "Events Relevant to Boiling Water Power Reactors Identified During Three Mile Island Accident"

IE Bulletin 79-05, April 2, 1979, and 79-05A, April 5, 1979, "Nuclear Incident at Three Mile Island"

4. Potentially any or all of the documents identified above may be offered as exhibits or used during cross-examination in support of Issue #8.

5. To the extent that this interrogatory requires OCRE to define the accident scenario which will govern the litigation of Issue #8, OCRE objects to this interrogatory, as this is not OCRE's responsibility. The Appeal Board stated that "(i)t is the Licensing Board's function to determine what a TMI-2 type accident is, insofar as the Perry facility is concerned." ALAB-675, slip op. at 19, footnote 13.

To the extent that this interrogatory requires OCRE to perform a detailed time-domain analysis of a TMI-2 type accident scenario specific to Perry, OCRE objects to this interrogatory, as OCRE does not have the resources for running the computer simulation models necessary to determine the rate and quantity of hydrogen production during the accident.

Furthermore, it should be noted that the TMI-2 accident does not "represent a unique scenario by which large amounts of hydrogen with eventual core cooling could be acheived. It is not possible to define a unique scenario since there are numerous ways in which the same end results could be obtained." NUREG/CR-2540 at 3. An examination of Table 5-4 of NUREG/CR-1659, Vol. 4 reveals that for the BWR/6-Mark III, containment failure from hydrogen burn is a likely result of 18 of the 36 accident sequences analyzed, which included large breaks, small breaks, and transients.

The TMI-2 accident involved a feedwater transient, stuck-open PORV, and operator error, with the consequences being a damaged core, an estimated 35-50% metal-water reaction, and the detonation of some of the hydrogen thus produced in the containment. An equivalent scenario for the BWR/6 would involve generally similar events. For the purpose of responding to the subsequent interrogatories, OCRE will assume that the $T_{23}PQE\text{-}\gamma$ sequence analyzed at pp. 6-12 to 6-14 and C-6 to C-10 of NUREG/CR-1659 Vol. 4 is an equivalent to the TMI-2 type accident. This accident involves:

T_{23} - loss of feedwater transient

P - failure of a safety/relief valve to reseat

Q - failure of the power conversion system to to provide makeup water

E - failure of the ECCS

γ - containment failure due to hydrogen burn

According to the NUREG/CR-1659 analysis (specific to Grand Gulf), core melting begins at 71 minutes. At pp. C-21 to C-28 of NUREG/CR-1659 some of the parameters calculated with the MARCH code for the T_{23}PQE sequence are presented graphically as a function of time. These graphs do not include any information on the rate and quantity of hydrogen production. The dominant contributor sequences to T23PQE are listed on p. D-4 of NUREG/CR-1659. The only operator error assumed for some of the sequences is the failure of the operator to manually initiate the ADS.

6. OCRE believes that any accident which is physically possible is credible. Credibility does not depend on predictability or probability. The history of the nuclear industry has shown that accidents can begin and proceed in the most unlikely and unpredictable ways, e.g., Browns Ferry fire, TMI-2, Browns Ferry 3 partial scram failure, Fermi breeder accident, etc.

   In addition, Appendix B of NUREG-0626 indicates that feedwater transients and safety/relief valve failures occur with moderate frequency in BWRs.

7. OCRE does not have the resources for determining, through sophisticated computer models, the off-site radiation doses associated with the T_{23}PQE-γ sequence. Even if OCRE were able to run such computer models, it should be noted that

such analyses are subject to large uncertainties and are dependent upon various assumptions, such as meteorological conditions and the effectiveness of protective actions. NUREG/CR-1659 Vol. 4 considers the $T_{23}PQE-\gamma$ sequence to fall into release category BWR-3 and to have a probability of 2.7×10^{-7} per reactor-year. Using Figure 5.4 of the Perry FES, NUREG-0884, and assuming that the $T_{23}PQE-\gamma$ sequence does occur at PNPP, it can be expected that at least 100 persons will receive whole body radiation doses in excess of 200 rems; that at least 20,000 persons will receive thyroid doses in excess of 300 rems; and that at least 100,000 persons will receive whole body doses in excess of 25 rems.

8. 10 CFR 100.11(a)(2) states that a person located at the low-population-zone boundary should not receive a dose in excess of 25 rems whole body or 300 rems to the thyroid. PNPP's LPZ has a radius of 2.5 miles. About 4225 residents live within the LPZ, which also has a peak transient population of 1575. Perry SER, NUREG-0887, Section 2.1.3. Thus the worst-case total LPZ population would be 5800 persons. The estimates given in the response to Interrogatory 7, above, indicate that the number of persons receiving at least the 10 CFR 100 doses vastly exceeds the LPZ population. Thus, it is reasonable to assume that a person located at the LPZ boundary would receive radiation doses greater than the 10 CFR 100 values.

9. (a) All documents relied upon were identified in the responses.

(b) No such persons have yet been identified.

10. (a) OCRE has no documents specifically concerning igniters; however, some of the documents identified in the response to Interrogatory 3, above, contain some applicable information.

(b) No persons have provided OCRE with information, expert advice or knowledge.

(c) OCRE believes that igniters will not safely control hydrogen produced by the $T_{23}PQE$ scenario. Figure C-11, p. C-23 of NUREG/CR-1659 Vol. 4 indicates that approximately 55% of the fuel rod cladding has reacted to liberate hydrogen by 105 minutes into the accident. Assuming that all of this hydrogen is released into the wetwell (and ignoring any other sources of hydrogen, e.g., radiolysis of water), Figure 2 of NUREG/CR-1561 indicates that in a Mark III, a 55% metal-water reaction will result in a hydrogen concentration in the containment of about 22 vol-%. Since the Mark III containment is not inerted, oxygen is present to support combustion. Igniters are designed to induce hydrogen combustion at concentrations of 4-8 vol-% (Applicants' Answer to OCRE Interrogatory 5-14). These igniters, as with all parts of the PNPP hydrogen control system, are initiated manually. There may be a delay of up to 60 minutes before the hydrogen analyzers, the first component to be activated, are in use. PNPP FSAR Section 6.2.5. Of course, there is also the possibility that, as with any manual action, the hydrogen control system will not be actuated at all, especially during the busy and stressful time of an accident. Figure C-11 of NUREG/CR-1659 indicates that the metal-water reaction begins at about 68 minutes into the accident; the reaction rate gradually increases until 105 minutes, at which time there is a rapid increase in reaction rate. Therefore, even if one assumes that the igniters can control hydrogen at the lower generation rates, at 105 minutes the rate increases such that there is a dangerous rise in hydrogen concentration. Specifically: prior to t = 105 minutes, about 18% of the fuel rod cladding has reacted. The resultant hydrogen is assumed to be removed by the hydrogen control system (recombiners-igniters). After t = 105 minutes, an additional 37% of the cladding reacts rapidly; according to the NUREG/CR-1561 data, 37% metal-water reaction corresponds to a 15% concentration by volume of hydrogen in the containment. According to Figure 3 of NUREG/CR-1561, adiabatic combustion of hydrogen, with initial conditions of atmospheric pressure, temperature of 25°C, and air saturated with water vapor, will result in a pressure of 5.7 atmospheres (84 psia) and a temperature of 1700°K (2600°F), for an initial hydrogen concentration of 15 vol-%. Compare these values with the Mark III design maxima of 15 psig and 185°F. Containment failure is certainly probable. The igniters would probably be the source of ignition. Assuming the most limiting condition, i.e., that the hydrogen control system is not effective at all at the lower rates and concentrations, combustion of the full 22vol-% of hydrogen would yield a pressure of 7 atmospheres (103 psia) and a temperature of 2400°K (3860°F), again from Figure 3 of NUREG/CR-1561. Figure C-14 of NUREG/CR-1659 Vol. 4 indicates that a hydrogen burn is expected at t≈ 100 minutes, with the pressure rising to 175 psia.

It should be noted that for other accident sequences, e.g., AE and SE (large and small pipe breaks with failure of the ECCS), the rate of hydrogen generation may be even greater. Figure C-25 of NUREG/CR-1659 Vol. 4 indicates that for these sequences, a 50% metal-water reaction occurs within 50 minutes.

The safety of using igniters to control hydrogen has been questioned by many; e.g.:

* General Electric: in view-graphs presented to the NRC and contained in SECY-80-107A, GE considered burning to be "impractical for significant rates and all sources" and identified several other problems with ignition, e.g., how to remove the heat of combustion and the difficulty of ensuring prompt ignition. Interestingly, GE also considers other hydrogen control methods (inerting, recombiners, venting) to be impractical as well.

* NRC: "the Commission believes that control methods that do not involve burning provide protection for a wider spectrum of accidents than do those that involve burning" 46 FR 62282, December 23, 1981.
* Hermann L. Jahn, Battelle, Frankfurt, Germany: "Enforced burning e.g. by spark plugs may be expected to be of questionable value and perhaps dangerous." NUREG/CR-2017, Vol. 3, p. 273.
* A. L. Berlad, et al., Brookhaven National Laboratory: "In assessing the advantages and disadvantages of various hydrogen control approaches, we have strongly favored methods which eliminate the combustion of accident-released hydrogen." NUREG/CR-2017, Vol. 4, p. 168.

11. Pre-accident inerting of the containment will prevent hydrogen combustion. The effectiveness of post-accident inerting depends on the rapidity with which it is initiated. Containment inerting, however, will not prevent failure of the containment by overpressurization by hydrogen and other noncondensible gases, with no combustion. NUREG/CR-1659 Vol. 4 at p. 6-14 indicates that there is an equal chance of containment failure in the $T_{23}PQE$ sequence by either hydrogen burn or overpressurization; the latter mode of containment failure, however, results in a less severe release category. General Electric, in SECY-80-107A, states that for a 100% metal-water reaction, the hydrogen pressure alone could exceed the containment design pressure, and thus inerting may not prevent con-

tainment failure. In NUREG/CR-2017, Vol. 4, p. 162, it is stated that "of all the hydrogen control approaches considered, a strategy of continuous inerting of the containment building is the only one which clearly eliminates the combustion hazard, does not involve adverse environmental effects, and succeeds in a way that is independent of the accident scenario." Post-accident inerting (i.e., inerting during the accident period) has the disadvantage of increasing containment pressure through the addition of inert gas during a time when the containment atmosphere cannot be vented. NUREG/CR-2017, Vol. 4, p. 179.

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12. No such persons have yet been identified.
13. No witnesses have been identified as yet.
14. NUREG/CR-2156, "Radiation-Thermal Degradation of PE and PVC: Mechanism of synergism and Dose Rate Effects," Roger L. Clough and Kenneth T. Gillen, Sandia National Laboratories, June 1981.

NUREG/CR-2157, " Occurrence and Implications of Radiation Dose-Rate Effects for Material Aging Studies," Kenneth T. Gillen and Roger L. Clough, Sandia National Laboratories, June 1981.

Final Safety Analysis Report for PNPP, Section 3.11.

Proposed rule to 10 CFR 50, "Environmental Qualification of Electric Equipment for Nuclear Power Plants" 47 FR 2876, January 20, 1982.

Memorandum for Raymond F. Fraley, Executive Director, ACRS, from Robert B. Minogue, Director, NRC-RES, dated April

16, 1982, re final rule 10 CFR 50.49 and analysis of public comments on proposed rule.

Final Rule, 10 CFR 50, "Environmental Qualification of Electric Equipment" 47 FR 28363, June 30, 1982.

IE Bulletin 79-01, "Environmental Qualification of Class 1E Equipment" February 8, 1979.

IE Bulletin 79-01A, June 6, 1979.

IE Bulletin 79-01B, Supplements 2 and 3, Sept. 30, 1980 and Oct. 24, 1980.

CLI-80-21, 11 NRC 707 (May 27, 1980) Memorandum and Order, in the matter of Petition for Emergency and Remedial Action.

IE Information Notice 81-20, "Test Failures of Electrical Penetration Assemblies" July 13, 1981.

IE Information Notice 81-29, "Equipment Testing Experience" September 24, 1981.

IE Information Notice 82-03, "Environmental Tests of Electrical Terminal Blocks" March 4, 1982.

Documents pertaining to specifications for electrical cables used at PNPP, listed in the attached letter of November 2, 1982 from Ronald Wiley, CEI to Susan Hiatt, OCRE.

15. Potentially any or all of the documents identified above may be offered as exhibits or used during cross-examination in support of Issue #9.

16. Based on research performed at Sandia National Laboratories, documented in NUREG/CR-2156 and NUREG/CR-2157, OCRE believes that the following polymers degrade more rapidly when exposed to lower levels of radiation for longer periods of time (i.e., normal service conditions) than when exposed to high levels for shorter periods

(i.e., integrated-dose accelerated aging qualification tests):

polyvinyl chloride (PVC)

polyethylene (PE)

crosslinked polyolefin (CLPO)

ethylene propylene rubber (EPR)

chloroprene rubber (CP)

chlorosulfonated polyethylene (CSPE)

17. OCRE believes that any components or equipment using these polymers in a radiation environment may suffer from degradation. Applicants have stated that safety-related electrical cable and wire used in a radiation environment use crosslinked polyolefin, crosslinked polyethylene, and ethylene propylene rubber as insulation. (Applicants' answer to OCRE Interrogatory 3-4) Therefore, OCRE believes that the cable types identified below, when used in the locations identified in the answers to Interrogatories 18, 20, and 21, below, may be subject to dangerous degradation. Particular circuits will be identified later.

Cable type	Use	Polymer
EKB- (xx)	control	CLPO, hypalon jacket
EKC-71 to EKC-107	instrumentation	CLPO
EKF- (xx)	thermocouple	CLPE, hypalon jacket
EKC-1 to EKC-50	instrumentation	CLPE, CSPE jacket
EKA-61 to EKA-110; EKA-140 to EKA-150; EKA-161 to EKA-206	power	CLPO, hypalon jacket

Cable type	Use	Polymer
EKA-2 to EKA-58; EKA-111 to EKA-138; EKA-151 to EKA-159	power	EPR, CSPE jacket

(This information was obtained from documents provided by Applicants for inspection at PNPP; copies of same provided to OCRE are listed in the November 2, 1982 letter from R. Wiley, CEI, to S. Hiatt, OCRE. OCRE is unsure whether hypalon is the same as CSPE.)

18. This interrogatory is somewhat unclear as to whether the locations to be identified are those made dangerous by a possible accident caused or aggravated by the polymer degradation or those in which the radiation levels or other environmental conditions are such that this degradation is likely. Assuming the latter, these locations are generally identified in the answers to Interrogatories 20 and 21, below.

Of particular concern to OCRE are those locations where cables serving redundant and/or diverse safety systems or components may be subject to similar degradation effects due to similar radiation environments. E.g., FSAR Figures 8.3-14 to 8.3-17 indicate that some principal cable routes for both Division 1 and Division 2 Class 1E circuits are located within the same radiation zones.

NUREG/CR-2156 indicates that such "dangerous locations" might be very localized and specific, more specific than the zones identified below. A cable (the discovery of which prompted the Sandia studies) which had been in service in a nuclear facility and which exhibited severe embrittlement in certain locations exhibited no deterioration in adjacent areas along the cable. This suggests that detailed dosimetry mapping would be necessary in order to pinpoint the exact locations of concern. OCRE suspects that such dosimetry mapping would require sophisticated computer analyses and possible could not be performed except by experiment. Since such techniques are beyond OCRE's resources, this Intervenor cannot identify the "dangerous locations" with any greater specificity than the radiation zones described below.

19. It was estimated in NUREG/CR-2156 (p. 8) that the maximum dose rate experienced by the severely degraded portions of the electrical cable (the discovery of which prompted the Sandia research) was about 25 rad/hr. This cable was in an unsafe condition, since the insulation fell off the wire when bent. Based on this information, OCRE would therefore conservatively estimate that an average dose rate greater than or equal to 10 rad/hr might cause dangerous deterioration. (Should future research implicate even lower levels, OCRE will revise these responses accordingly.)

20. The radiation zones identified herein are the same as the environmental zones described in FSAR Table 3.11-1. The maximum normal gamma radiation doses given for these zones are integrated over 40 years. Since 40 years is equivalent to 3.5×10^5 hours, the average dose rate was obtained by dividing the FSAR values by 3.5×10^5. Using data from FSAR Tables 3.11-2 through 3.11-8, the following zones

have dose rates during normal operation of at least 10 rad/hr:

Zone	Dose Rate, rad/hr
DW-1	80
DW-2	129
DW-3	5.7 (This zone was included because a large neutron fluence, 2.9 x 10^8 Ntn/cm^2/hr, is also present.)
DW-4	80
DW-5	80
CT-5	514
AB-8	22
FB-6	27

21. The radiation zones identified herein are the same as the environmental zones identified in FSAR Table 3.11-1. The gamma radiation doses given for these zones for accident conditions are integrated over 6 months. Since 6 months is equivalent to 4380 hours, the average gamma dose rate was obtained by dividing the FSAR values by 4380. In addition, the beta dose rates are calculated to be 67 krad/hr for all drywell zones and 25 krad/hr for all containment zones (see Notes 2 and 3 to Tables 3.11-2 to 3.11-8). The total average dose rate is thus the sum of the gamma and beta values. Accident conditions are those identified in the FSAR tables. Using data from FSAR Tables 3.11-2 through 3.11-8, the following zones have dose rates during accident conditions of at least 10 rad/hr:

Zone	Dose Rate, rad/hr
DW-1	128×10^3
DW-2	128×10^3
DW-3	128×10^3
DW-4	128×10^3
DW-5	128×10^3
CT-1	35×10^3
CT-2	29×10^3
CT-3	29×10^3
CT-4	29×10^3
CT-5	29×10^3
CT-6	29×10^3
CT-7	29×10^3
CT-8	29×10^3
AB-2	6.8×10^3
AB-3	4.3×10^3
AB-4	9.4×10^3
AB-5	2.5×10^3
AB-7	3.9×10^3
AB-8	2.2×10^3
FB-6	1.2×10^3
FB-7	169
FB-8	347

22. (a) All documents were identified in the response. The calculations made in answering Interrogatories 20 and 21 used data in Amendment 9 of the FSAR. Should a later amendment provide different data,

the responses will be amended accordingly.

(b) No such persons have been identified.

23-26. The information requested by these Interrogatories is provided by the attached affidavit. OCRE objects to the portion of Interrogatory 26 dealing with the location where the document search was conducted as this information is not relevant to either Issues 8 or 9 and infringes too closely upon the work-product doctrine. F.R.C.P. 26 (b) (3).

27. The following statements pertain to Issue #8:

(a) Separate Views of Commissioners Gilinsky and Bradford and Separate Opinion of Commissioner Bradford, to Duke Power Company (Wm. B. McGuire Nuclear Station, Units 1 and 2) CLI-81-15, 14 NRC 1 (1981); these statements concern the litigation of hydrogen control under 10 CFR Part 100.

(b) Separate Views of Commissioner Gilinsky to Final Rule to 10 CFR Parts 2 and 50: "Licensing Requirements for Pending Construction Permit and Manufacturing License Applications" 47 Fed Reg 2286 at 2300 (January 15, 1982), in which the Mark III is characterized as a weak containment which should be required to be stronger.

Respectfully submitted,

Susan L. Hiatt

Susan L. Hiatt
OCRE Representative
8275 Munson Rd.
Mentor, OH 44060

THE CLEVELAND ELECTRIC ILLUMINATING COMPANY
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OFFICE OF SECRETARY
DOCKETING & SERVICE
BRANCH

November 2, 1982

Ohio Citizens for Responsible Energy
c/o Ms. Susan Hiatt
8275 Munson Road
Mentor, Ohio 44060

Dear Ms. Hiatt:

On behalf of Ohio Citizens for Responsible Energy, you requested copies of documents relative to issue #9 which you inspected at PNPP on October 24, 1982.

Copies of each of the following documents are enclosed:

From the file entitled, "Cable - Misc. Safety Related Instrumentation Specification SP-793-01":

Bill of Material Sheets: EKB 51, 61, 71 & 81
EKC-71 through EKC-107

Rockbestos Co. Spec Sheets: RSS-6-104, RSS-6-105, RSS-6-110, RSS-6-112, RSS-6-116, RSS-6-200, RSS-6-207

From the file entitled, "Cable Thermocouple Extension - Class 1E Specification SP-567", Conformed Spec. pp. 7-11

Bill of Material Sheets: EKF-1 through EKF-35 (except those which are deleted)

From the file entitled, "Cable Instrumentation - Class 1E Specification SP-561 Auditable Material", Conformed Spec. pp. 7-11

Bill of Material Sheets: EKC-1 through EKC-50

From the file entitled, "Cable Class 1E Small Power & Control Cable SP-560 Auditable Material," Conformed Spec. pp. 7-11

Bill of Material Sheets: EKA-61 through EKA-206
EKB-1 through EKB-100

From the file entitled, "Cable Class 1E Medium Voltage Power Cable Specification SP-559 Auditable Material", Conformed Spec. pp. 7-13; 16-18

Bill of Material Sheets: EKA-2 through EKA-159

From the binder entitled, "Project Design Criteria":

pp. 2-48, 2-48a and pp. 2-68 through 2-79

From the file entitled, "Cable Class 1E Medium Voltage Power Cable - AEIC Standards"

p. 8 from document entitled: "Specifications for Polyethylene and Cross-linked Polyethylene"

p. 9 from document entitled: "Specifications for Ethylene Propylene Rubber"

Please sign below and return one copy of this letter to indicate that you have received these documents.

Applicants' cost of duplication for these documents @ $0.10 per page is $30.30.

Please remit a check in the above amount payable to THE CLEVELAND ELECTRIC ILLUMINATING COMPANY, and address to:

Ronald G. Wiley
c/o Perry Nuclear Power Plant
P. O. Box 97
Perry, Ohio 44081

Sincerely,

THE CLEVELAND ELECTRIC ILLUMINATING CO.

Ronald G. Wiley

Ronald G. Wiley

Requested documents received by:

Susan L. Hiatt Nov. 9, 1982
Susan L. Hiatt Date
Ohio Citizens for Responsible Energy

RGW:dlp

cc: Jay E. Silberg

AFFIDAVIT

I, Susan L. Hiatt, being duly sworn depose and say that I am responsible for the answers given in OCRE's Response to Applicants' Interrogatories and Request for Production of Documents to Intervenor Ohio Citizens for Responsible Energy (Second Set) and that these answers are true to the best of my knowledge and belief.

Susan L. Hiatt

Susan L. Hiatt

Sworn to and subscribed before me this 15th day of November, 1982.

Randi B. Jenkins

Notary Public

RANDI B. JENKINS, Attorney - at - Law
Notary Public, State of Ohio
My Commission Has No Expiration Date
Section 147.03 R. C.

CERTIFICATE OF SERVICE

This is to certify that copies of the foregoing OCRE RESPONSE TO "APPLICANTS' INTERROGATORIES AND REQUEST FOR PRODUCTION OF DOCUMENTS TO INTERVENOR OHIO CITIZENS FOR RESPONSIBLE ENERGY (SECOND SET)" were served by deposit in the U.S. Mail, first class, postage prepaid this 15th day of November 1982 to those on the service list below.

Susan L. Hiatt

Susan L. Hiatt

SERVICE LIST

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