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RELATED CORRESPONDENCE

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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' SECOND 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 these

1/ Motion to Reopen Discovery on Issue No. 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 submitted an initial set of voluntary responses on November 16, 1984.4/ Applicants submit the remaining portion of their voluntary responses below.

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

13-63. For the 2 CLASIX-3 runs described in OPS-38A92, provide the following information: a detailed sequential description of the accident scenarios analyzed, including the times of all significant events, break sizes, operator actions and errors, ECCS flow rates, initial power level and decay heat assumptions, etc., and the basis for choosing these scenarios.

4/ Applicants' Voluntary Answers to a Portion of OCRE's Late-Filed Thirteenth Set of Interrogatories to Applicants (Issue No. 8) (November 16, 1984).

Response:

The steam and hydrogen releases used in both CLASIX-3 runs described in OPS-38A92 were from a MARCH code accident analysis and were based on an accident sequence which represents a transient-initiated event followed by a failure of a safety relief valve to close, coupled with a failure of all cooling water supply systems. The scenario is based on NRC-sponsored probability studies of BWR accident scenarios capable of producing significant hydrogen. A description of such a sequence was provided in the NRC Staff's response to OCRE Interrogatory No. 6-2 of NRC Staff Answers to OCRE Sixth Set of Interrogatories and Board Questions to NRC Staff, dated February 3, 1983.

Applicants are not aware of any detailed time sequence of all significant events for the specified sequence. The break size assumed in this sequence is 0.16 ft^2, the flow area of the stuck open relief valve assumed in MARCH. The sequence makes the conservative assumption of no mitigating operator action. The sequence assumes numerous failures, without regard to whether the failures were caused by equipment malfunction or operator error. System design flow rates for ECCS were assumed in the recovery and termination of the accident in OPS-38A92. ECCS flow rates for recovery and termination of the event are described in OPS-38A92, Table 9. Applicants did not perform the MARCH analysis and do not have information to address other "key assumptions" requested by this interrogatory. See response to Interrogatory No. 13-65.

13-64. Explain why a drywell burn was not modeled in the SORV transient (OPS-38A92, p. 2). If it is your position that hydrogen will not accumulate in the drywell in the SORV transient, demonstrate that the CGCS compressors or leaks associated with the high point vents or safety relief valve discharge lines or vacuum breakers will not introduce hydrogen into the drywell.

Response:

The study did model hydrogen combustion in the drywell for the SORV transient. However, a quantity of hydrogen capable of combustion did not exist in the drywell volume at the time the transient was terminated. The model used in the SORV transient analyses was designed to permit a drywell burn if the conditions in the drywell could support combustion. Hydrogen did accumulate in the drywell during the transient. However, as shown in Figure 16 of the referenced document, the hydrogen concentration in the drywell never reached 1% by volume, which is well below the minimum for any combustion.

It is not Applicants' position "that hydrogen will not accumulate in the drywell in the SORV transient."

13-65. OPS-38A92, p. 2 states that MARCH Code values for hydrogen, steam, and energy release were modified for input into CLASIX-3. Concerning the MARCH analysis:

(A). State where, when, and by whom the MARCH Code runs were performed.

(B). State the version of the MARCH Code used.

(C). Describe in detail the accident scenarios considered in the MARCH analysis.

(D). Identify the value used for each and every MARCH input variable for each scenario run.

(E). State when each MARCH run was terminated and why this time was chosen.

(F). Explain in detail how the MARCH Code results were "modified."

(G). Was the hydrogen source term from MARCH considered constant over a time interval, or was interpolation between data points used? If so, explain why and what type of interpolation was used.

Response:

(A) The March analyses were performed by Battelle Columbus Laboratories for the Nuclear Regulatory Commission. Applicants' understanding is that these analyses were performed in 1979.

(B) Applicants do not know which version of the MARCH code was used.

(C) See the response to Interrogatory No. 13-63.

(D) Applicants will make available to OCRE for inspection a copy of a list of input data provided by the NRC to Applicants.

(E) Only one of the MARCH runs was used for the analysis in question. The MARCH analysis was terminated subsequent to postulated reactor vessel melt through the bottom head. The time for termination was chosen based on considerations relevant to severe accident conditions which are beyond the scope of Issue #8.

(F) The MARCH results were modified to produce hydrogen generation rates consistent with the criteria of the proposed rule, published at 46 Fed. Reg. 62281 (December 23, 1981) ("proposed rule"). Accordingly, the results were modified prior to severe core melt to provide a hydrogen release rate representative of a postulated recoverable degraded core accident resulting in hydrogen equivalent to 75% metal water reaction of the cladding in the active core region. The MARCH hydrogen release rate was held at 1 lbm/sec, the peak hydrogen release rate prior to severe core melt, until the requisite amount of hydrogen was released.

(G) The study used linear interpolation between the data points of the MARCH analysis. Linear interpolation was used because the distribution of data points and the large time intervals in the MARCH analysis output did not provide enough definition for a curve-fitting interpolation.

13-66. Explain why drywell vacuum relief valves were neglected in the containment to drywell flow path (OPS-38A92, p. 3).

Response:

There was no need to include the drywell vacuum relief valves in the containment to drywell flow path. The drywell vacuum relief valves use the same drywell penetration as the combustible gas control drywell purge compressors, which are initiated at 20 minutes in the transient analysis. With the drywell purge compressors operating, the containment to drywell path is open and the lines are pressurized. Under these conditions, there is no need to model the vacuum breakers. Figures 9 and 28 of OPS-38A92 show that, prior to initiation of the drywell purge compressors, drywell pressure is always greater than containment pressure. Because of the pressure differential, the vacuum breakers would not operate, and thus, do not need to be included in the analysis.

13-67. Identify the "prior analyses" showing that the containment pressure never drops below atmospheric (justification for neglecting containment vacuum relief valves, OPS-38A92, p. 3).

Response:

The "prior analyses" are contained in the document identified as Reference 1, at page 7 of the OPS-38A92 report. A copy of this document will be made available for inspection at the site upon request.

13-68. Did the OPS analysis consider the apparent contribution to containment pressurization of below atmospheric pressures in the shield building annulus? If so, explain how; if not, why not?

Response:

The below atmospheric pressure in the shield building annulus will not contribute to the containment pressurization analyzed in the OPS report. The containment is a closed system and absolute pressures were used in the analysis. Thus, the CLASIX-3 analysis did not consider the below atmospheric pressure in the shield building annulus because it has no effect on the analysis.

13-69. Explain the basis for the assumption (OPS-38A92, p. 4) that sheet flow has half the cooling capability of droplet flow. Indicate whether the assumption has any basis in experimental data.

Response:

Sheet flow provides less effective cooling capability than spray droplets because there is less total surface area available for cooling the air. At the same time, hydrogen combustion would cause the sheet flow to be dispersed by, and entrained with, the expanding combustion gaseous products, thereby increasing the surface area of the sheet flow which will be available for cooling. Consideration of these factors led to the assumption in OPS-38A92 that sheet flow has half the cooling capability of droplet flow. Applicants know of no experimental data directly applicable to the assumption.

13-70. (A) Does the OPS report assume that both containment spray RHR loops are operating to mitigate hydrogen burning?

(B) Reconcile the assumption that containment sprays are available and operable after the first burn (OPS-38A92, p. 4) with the statement that reinstatement of injection systems (i.e. RHR/LPCI mode) occurs at 6500 seconds into the transient (OPS-38A92, p. 2).

Response:

(A) No.

(B) Applicants' residual heat removal (RHR) system is a redundant, safety-grade system which is designed to mitigate the consequences of loss of coolant accidents and transient events. The RHR system has a number of modes, including a containment spray mode and a low pressure coolant injection (LPCI) mode. Under the design of the RHR system, it is possible to have one or both of these two modes operable.

The OPS-38A92 analysis assumed that the RHR containment spray mode would be available to mitigate a hydrogen burn after the first burn, which occurs at approximately 4500 seconds for both CLASIX-3 runs. The analysis further assumed that emergency core cooling injection systems (LPCI or other injection systems) would initiate at 6500 seconds. The study delayed activation of the injection systems until 6500 seconds into the transient in order to simulate a recoverable degraded core capable of producing significant quantities of hydrogen, as specified in the proposed rule.

13-71. Demonstate that the simulation of a drywell spray (OPS-38A92, p. 4) does not create artificial results.

Response:

The drywell spray (safety injection flow out of the break) was simulated to reduce the steam concentration in the drywell so that combustion will occur earlier in the transient than would occur without the spray. Without the spray, the drywell burn would occur later in the transient and would result in a less severe pressure and temperature response in the containment. Thus, the simulation of the drywell spray produces conservative results.

13-72. Explain the basis for the passive heat sink node criteria on pp. 4-5 of OPS-38A92.

Response:

(First Criterion - Coatings). Two nodes were used to model the potential effects of coatings on the rate of convective and radiant heat transfer from the gases to the passive heat sinks. One node would have been insufficient to properly model the surface heat transfer and to account for the interface temperature between the coating and the steel. More than two nodes were not necessary, because coatings are comparatively thin and have a negligible heat sink capacity.

(Second Criterion - Layers Other than Coatings). All other layers have a minimum of three nodes with the actual number based on the thickness to assure appropriate treatment of their heat transfer characteristics. This distinguishes these layers from coatings and identifies a heat sink capacity for purposes of the analysis.

(Third Criterion - Steel Walls). Because of the high thermal conductivity of steel, closely spaced nodes (approximately .02 inch per node) are required to represent adequately the material heat transfer performance.

(Fourth Criterion - Concrete Walls). Concrete has a low thermal conductivity and can be adequately represented with fewer nodes (i.e., one inch per node for the first six inches, two inches per node for the next twelve inches and six inches per node for the next one and one-half feet.) Also, because of its low thermal conductivity, any concrete beyond the first foot of thickness will have a negligible response during the transient.

13-73. Was the upper pool dump assumed to fill any portion of the holdup volume before reinstatement of vessel injection? If so, specify the amount of water in the drywell pool due to the upper pool dump. If not, why not?

Response:

In the analysis, the water from the upper pool dump did not fill the holdup volume during the dump procedure. Water from the upper pool is piped directly to the suppression pool outside of the drywell and, therefore, cannot directly enter the holdup volume, which is entirely inside the drywell. In addition, at the time of upper pool dump, drywell pressure is greater than containment pressure. Because of this pressure differential, no water would spill from the suppression pool, over the weir wall, into the holdup volume.

13-74. Was the upper pool dump assumed to condense any steam in the drywell? If so, state whether drywell burning occurred as a result. If not, why not?

Response:

No. See the response to Interrogatory No. 13-73.

13-75. Reconcile the assumption that the drywell pool will be formed with emergency procedure guidelines that instruct operators to maintain vessel level below level 8.

Response:

The assumption that the drywell pool will be formed is consistent with the conservative assumption that there will be safety injection flow out of the break and into the drywell. See response to Interrogatory No. 13-71.

13-76. Explain the bases, including support from experimental data, for each and every burn parameter given in Table 4 of OPS-38A92.

Response:

The burn parameters given in Table 4 were selected as a conservative set of values for hydrogen ignition in the BWR Mark III containment and were used in the OPS-38A92 CLASIX-3 analysis of containment pressure and temperature response to hydrogen deflagration.

First Parameter: Testing of the ignitor under conditions which will occur in the containment during the degraded core transient has demonstrated reliable ignition of hydrogen concentrations as low as 5%. See Tamm, H. et al., "A Review of Recent Experiments at WNRE on Hydrogen Combustion," Whiteshell Nuclear Research Establishment, Canada. Presented at the Second International Conference on the Impact of Hydrogen on Water Reactor Safety, October 3-7, 1982 ("Tamm Study") (Figure 1). To be conservative, it was assumed that ignition would not occur until hydrogen had built up to a concentration of 8%.

Second Parameter: Although little testing has been conducted in this area, data exists showing that propagation will occur at 8%. See Tamm Study, cited above. Use of a higher concentration than 8% for propagation would have no effect on the results, since self-ignition would occur at 8%, as discussed above.

Third Parameter: There is extensive data available on the fraction of hydrogen which will be consumed by the combustion resulting from ignition at 8%. Although a few data points indicate that consumption of hydrogen may approach 100%, considering the conservatism in the concentration value specified for ignition, a less extreme value of 85% hydrogen fraction burned was selected as being more representative.

Fourth Parameter: With excess hydrogen present, ignition does not occur until the oxygen concentration is slightly above 5%, at which value there is a sharp transition from non-ignition to ignition. There is a significant quantity of experimental evidence for the 5% value. The most recent data are from experiments conducted at Whiteshell. See Tamm, H. and Ungurian, M., "Ignition Effectiveness of the GM AC 7G Glow Plug in Rich Hydrogen-Air-Steam Mixtures," Whiteshell Nuclear Research Establishment Canada. 1983 ("Tamm and Ungurian Study").

Fifth Parameter: The minimum oxygen required to support continued combustion is applicable only in hydrogen rich burns. There is experimental evidence that all, or almost all, of the oxygen present in a hydrogen rich atmosphere will be consumed if ignition occurs. See Table 1 of the Tamm and Ungurian Study, cited above. Accordingly, the OPS-38A92 study assumed that no oxygen would be necessary to support combustion, once ignition has occurred.

Sixth Parameter: The burn time is an estimate of the time which would be required for the hydrogen to burn in a volume of a given size. The burn time represents the time it takes for a flame to propagate from the point of ignition to the farthest point within the volume. This time is dependent on the size of the volume and the speed at which the flame propagates. The geometry of the plant volumes is well established. The speed at which the flame propagates is dependent on the gaseous mixture. Recent tests at Lawrence Livermore Laboratory show that, for hydrogen concentrations of 10% or less, the flame speed is less than 3 feet/second. See Lowery, W., "Preliminary Results of Thermal Igniter Experiments in H^2-Air-Steam Environments," Lawrence Livermore Laboratory presented at the First International Conference on the Impact of Hydrogen on Water Reactor Safety, January 26-28, 1981. The flame speed was doubled for conservatism, to 6 feet/second. This flame speed and the plant geometry produce the burn times in the table.

13-77. Explain whether the initial conditions in Table 5 of OPS-38A92 are pre-burn conditions or are conditions existing prior to the transient. If the former, explain why no increase in containment pressure, temperature, or steam fraction resulted from the transient; if the latter, explain how pre-burn rises in temperature, pressure, and steam and hydrogen fractions are calculated.

Response:

The initial compartment conditions given in Table 5 of the reference are the conditions existing prior to the transient. The pre-burn conditions are a result of the entire transient history of the containment from initiation of the transient to the point in time of interest as calculated by the CLASIX-3 code.

The CLASIX-3 program uses standard engineering equations for fluid flow, heat transfer, chemical reactions, conservation of mass and energy, perfect gas laws and the ASME steam tables. Beginning with the initial containment conditions specified, CLASIX-3 uses these equations to calculate, as a function of time, the transient response of the containment to the injection of hydrogen, steam, and, eventually, the combustion of the hydrogen.

13-78. Explain the bases for the flow path parameters given in Table 6 of OPS-38A92.

Response:

The flow path parameters are based on plant design parameters for the flow area. The loss coefficient is a conservative estimation based on plant design. The burn propagation delay time is a function of the flow length and the assumed flame speed discussed in response to Interrogatory No. 13-76.

13-79. Explain whether drywell bypass leakage was modeled in the CLASIX-3 analysis. If not, why not?

Response:

Drywell bypass leakage was not modeled in the CLASIX-3 analysis. Expected drywell bypass leakage, as discussed in Applicants' February 21, 1984 supplemental response to OCRE Interrogatory No. 5-61, were considered to be insignificant because of the conservatisms included in the analysis.

13-80. Explain the bases for each and every parameter listed in Tables 7 and 8 of OPS-38A92.

Response:

The parameters listed in Tables 7 and 8 of the referenced report are based on PNPP plant design values, except for the following:

(1) The head/flow parameters for the drywell purge system in Table 7 are based on the equipment performance curve for the PNPP compressor.

(2) The drywell purge system initiation (note *) in Table 7 is based on FSAR Section 6.2.5.5 and a conservative assumption of initiation 20 minutes post-LOCA. This is conservative because it introduces air into the drywell and allows time for sufficient oxygen buildup.

(3) The water density in Table 8 was obtained from ASME tables for the specified temperature and pressure.

13-81. Explain the bases for each and every parameter for the containment/wetwell spray listed in Table 9 of OPS-38A92.

Response:

The first three parameters for the containment/wetwall spray in Table 9 are flow rate, temperature, and drop diameter. These parameters were derived from PNPP plant design values. The next parameter, fall time, was based on the design height of the PNPP compartment and on standard engineering equations. The next parameter, heat transfer coefficient, was based on engineering judgment about heat transfer from water droplets to ambient air. The next parameter, containment to wetwell carry over fraction, was calculated based on the design values for PNPP surface areas, and on the assumption that the sheet flow is half as effective as spray droplets in transferring heat, as discussed in the response to Interrogatory No. 13-69. The last containment/wetwell spray parameter in Table 9, initiation, was based on PNPP operating procedures.

13-83. Explain the bases for the parameters given in tables 10, 11, 12, 13 and 14 of OPS-38A92.

Response:

The heat sink parameters in these tables were based on PNPP plant design values, except for beam length for radiation heat transfer, which was based on the handbook value of 2/3 of the characteristic length of the compartment.

13-84. What is "Chemtree"?

Response:

Chemtree is a surface coating used on steel surfaces in the containment.

13-85. Explain the bases for the parameters given in Tables 15 and 16 of OPS-38A92.

Response:

The parameters given in Tables 15 and 16 are based on PNPP plant design values, except for the initiation times, which are discussed in the Applicants' response to Interrogatory No. 13-70.

13-102. (A) Identify all parameters in the CLASIX-3 code which, at the user's option, may either be calculated internally or supplied in the input.

(B) For each parameter listed in your answer to (A), identify which method was used in the CLASIX-3 analysis for PNPP.

(C) Explain why the chosen method was used.

(D) Explain the effect on the results if the other method had been used instead.

Response:

(A) Any two of the following may be calculated internally at the user's option. The third must be specified as input. They are, for each type of gas in each volume of the containment.

1. Mass

2. Partial pressure

3. Volume fraction and total pressure.

(B) The partial pressures were specified as input and the others were calculated internally.

(C) The partial pressures were used because they were most readily available.

(D) There would have been no effect on the results if either of the other options had been specified as input. In combination with other required input, any one of the three options given in (A) above fully specifies the initial conditions in each volume.

13-104. Did any CLASIX-3 analyses for PNPP assume that hydrogen is burned as it is added or released to the containment or drywell? Identify all such analyses and explain why that model was employed.

Response:

None of the CLASIX-3 analyses for PNPP assumed that the hydrogen was burned at the instant it was introduced into the containment. However, in the analyses, some of the hydrogen was burned during the time period over which hydrogen was being introduced into the containment. When the conditions in any given compartment in the containment reached those required for ignition, combustion was initiated. Hydrogen introduced during each period that a burn was occurring was considered in that burn.

13-106. List the values of all input parameters (including default values) used in CLASIX-3 analyses of PNPP, and explain why each value was used.

Response:

The CLASIX-3 analyses of PNPP used the default values for the following gas constant parameters:

1. The high heat of combustion of hydrogen at 61,000 Btu/lbm.

2. The specific heat at constant pressure for oxygen, nitrogen, hydrogen and steam at 0.2172, 0.2481, 3.399 and 0.588 Btu/lbm-F, respectively. (The value for steam is used only above the upper limit of the steam tables at 1500 F.)

3. The specific heat at constant volume for oxygen, nitrogen, hydrogen and steam at 0.1551, 0.1772, 2.402 and 0.450 Btu/lbm-F, respectively. (The value for steam is used only above the upper limit of the steam tables at 1500 F.)

4. The ideal gas law constant for oxygen, nitrogen, hydrogen and steam at 49.29, 55.16, 766.5 and 85.78 ft-lbf/lbm-R respectively. (The value for steam is used only above the upper limit of the steam tables at 1500 F.)

These default values were based on standard textbook values and are the same values that would have been used if they had been specified as input.

All of the other parameters in the CLASIX-3 analyses required input values. All of the input values used for these other parameters have been discussed in responses to previous interrogatories herein, with the exception of the differential time step used in the program. The time step used throughout the analyses was 0.01 seconds. The time step selected for these analyses was used throughout the verification runs.

13-107. Identify all CLASIX-3 outputs, including but not limited to containment conditions, heat sink data, and burn maxima data, produced for PNPP.

Response:

The initial output from CLASIX-3 consists of a reiteration of all input, followed by a complete listing, with descriptive phrases and dimensions, of all input and default values to be used in the analysis. During the transient, at user-specified intervals (typically 10 seconds), the following information is outputted by the program:

1. Time and current differential time step.

2. The net free volume, temperature, mass and partial pressure of each gaseous constituent, total internal energy in the ambient atmosphere, the rate of hydrogen combustion and the total hydrogen that has combusted for each control volume.

3. Spray mass inflow rate, temperature and rate of heat transfer in each control volume.

4. Hydrogen temperature and mass rate of addition to each volume.

5. Water enthalpy and mass rate of addition to each volume.

6. Volumetric flow rate in each flow path.

7. Angular position and volumetric flow rate of each check valve.

8. Gaseous volumetric flow rate through the suppression pool vent.

9. The total mass of water in the suppression pool and the water levels in both the wetwell and drywell.

10. Suppression pool drawdown rates of flow.

11. Drawdown flow rate entering into the holdup volume.

12. Rate of heat addition from input tables to each compartment.

13. Total volume of water in the holdup volume.

14. For each wall, the surface rate of heat transfer and the surface film coefficient of heat transfer.

15. The temperature of every node of every wall in every control volume.

16. The maximum temperature and maximum pressure, and the times of occurrence, achieved during a burn episode.

17. Restart files which are effectively snapshots of the computer code used in restarting the program as required.

18. Plot files, which are recordings of a subset of the preceding information and are used to generate computer plots of the transient.

13-111. For the 1/20 scale hydrogen combustion test program, describe:

(A) Purpose of the tests;

(B) Detailed design of the test facility;

(C) Location of the test facility;

(D) Scaling of the facility and the basis for using the scaling model;

(E) Tests performed in the facility;

(F) The data obtained from the tests and any analyses of the test data;

(G) Any conclusions drawn from the tests;

(H) Any comments or evaluations of the tests or the facility by the NRC or any NRC contractors.

Response:

(A) The purpose of the tests was to visualize the hydrogen combustion phenomena in a small-scale transparent model of a Mark III containment building.

(B) The test facility was a one-twentieth linear-scale model of a BWR6/Mark III containment vessel (Grand Gulf Nuclear Station). The facility simulates the Mark III wetwell and upper containment including the suppression pool and peripheral blockages in the annular region between the drywell and outer containment walls. Hydrogen was admitted through small scale sparger devices and/or LOCA vents in the suppression pool and ignited by a number of prototypically distributed spark igniters.

The entire outer wall of the facility was fabricated from rolled pyrex glass to allow observation of the hydrogen flames. The drywell was fabricated from a 1/8-inch-thick rolled steel plate and welded to an integral steel support skirt. The drywell wall was covered with a 1-inch-thick layer of insulation selected for proper heat transfer properties. Four vertical members supported the steel roof and the structure was supported and held together with steel channels. Floor gratings were simulated using a steel mesh material supported by U-shaped channels at each floor level. The sides of the vessel were enclosed by eight panels of 7/32-inch-thick rolled pyrex plate. The glass was held in place by circumferential steel bands which compress edges against gasketed seats.

The spherical containment head was replaced by a flat top in the test facility. However, the total containment volume was maintained consistent with modeling rules. Due to structural limitations of the glass walls, two pressure relief systems were incorporated in the design. The first system was designed to allow venting uniformly throughout the facility without disturbing the normal convective flow patterns. The venting also prevented pressure buildup while allowing uniform depletion of the oxygen. This was accomplished by a network of twenty-four 1/2-inch diameter tubes uniformily distributed throughout the vessel. The second pressure relief system was a shear-pin-loaded lid positioned on top of the wetwell to provide pressure protection against deflagration burns.

The foregoing answer is based on a portion of "Hydrogen Combustion Testing in a One-Twentieth-Scale Model of a BWR/Mark III Containment, Volume I: Facility Description and Results Summary," Acurex Corporation (undated), which will be supplied to OCRE for inspection at the site upon request.

(C) The test facility was assembled at Acurex Corporation, Mountain View, California.

(D) The objective of the test was to reproduce the flow and combustion dynamics in the Mark III containment subsequent to hydrogen ignition. Froude scaling was selected as the modeling approach. Froude scaling was used because it is the only accepted engineering approach to scaling applicable to this test facility. See response to Interrogatory No. 13-112(D) below.

(E) The base case test was a hydrogen flow rate of 1 lbm/sec (full scale), 185°F water temperature, a water level corresponding to upper pool dump, and hydrogen release through eight ADS spargers and the sparger at azimuth 312°. This corresponds to an ADS event with a stuck open relief valve. Parameters that were varied included hydrogen flow rate, location of discharge, and water (and consequently initial air) temperature. In addition, a series of tests were conducted with insulation covering the pyrex wall. An additional series of tests looked at effects of removing a floor sector at the refueling level at azimuth 40° which is absent in some Mark III plants (but is present in PNPP).

(F) Because of the large scaling factors, there was no accurate quantitative test data which could be obtained from the 1/20 scale testing. The data obtained to meet the test objective was a visualization and, therefore, a characterization of the burning phenomenon above the suppression pool.

The base case test with a hydrogen flow rate of 1 lbm/sec (full scale equivalent value) through nine spargers resulted in an initial deflagration burn within approximately 10 seconds after initiation of hydrogen injection. The deflagration burn propagated downward to the pool surface where steady diffusion flames were established and anchored above each active sparger. This phase of the phenomenon was termed primary burning and was characterized by strong convective flows which appeared to circulate from the upper containment down to the pool surface very effectively. Certain areas of the containment became "hot chimneys" with hot gases circulating to the upper containment and other areas became "cold chimneys" with cold air circulating down to the flames.

As oxygen was depleted, the flames became more erratic. The flame height increased, but the flames became weaker and were extinguished and reignited for several seconds at a time at individual sparger locations. As oxygen was decreased further due to the facility venting, the flames lifted from the pool surface and became very weak, such that they seemed to be momentarily extinguished by strong horizontal flows. This phenomenon was termed secondary burning.

A variation of test parameters was carried out which showed that as the hydrogen flow was increased, the flames were higher and event duration shorter as expected. Removal of the 40° floor section resulted in essentially no change in the flame characteristics and flow patterns.

A series of tests was conducted to determine the minimum hydrogen flow rate at which steady diffusion flames could be maintained. This was conducted by initiating diffusion flames with a hydrogen release rate of 1.0 lbm/sec (full scale) and gradually decreasing the hydrogen flow rate over several minutes until a continuous diffusion flame could no longer be maintained. As the hydrogen flow rate was decreased the result was a random extinction and reignition of local regions of flames around the circumference of the pool. However, the point at which the flames were no longer continuous, defined as the threshold limit, was repeatable. This lower limit was observed to range from 0.4 to 0.5 lbm/sec (full scale) for cold and hot pool conditions, respectively.

(G) Because of the 1/20 scale, testing had limited objectives and the conclusions which can be drawn from the test results are limited. Because the scaling factors were so large, there was no quantitative data which can be used from the testing program. Therefore, the conclusions reached were as follows:

1) Steady diffusion flames occur above the suppression pool for hydrogen release rates greater than 0.4 to 0.5 lbm/sec.

2) A repeatable threshold for diffusion flames exists at hydrogen release rates of 0.4 to 0.5 lbm/sec.

3) The size of the diffusion flames is directly proportional to the hydrogen release rate.

4) Diffusion flames established strong convective flows with hot and cold "chimneys."

5) Accurate prediction of the thermal environment from diffusion flames would have to be made in a larger scaled facility.

(H) Applicants have not received any comments or evaluations of the tests or the facility from the NRC or any NRC contractors.

13-112. For the 1/4 scale hydrogen combustion test program, describe:

(A) Purpose of the tests;

(B) Detailed design of the test facility;

(C) Location of the test facility;

(D) Scaling of the facility and the basis for using the scaling model;

(E) Tests performed in the facility;

(F) The data obtained from the tests and any analyses of the test data;

(G) Any conclusions drawn from the tests;

(H) Any comments or evaluations of the tests or the facility by the NRC or any NRC contractors.

Response:

(A) The overall objective of the test program is to determine the transient local heat fluxes which can be expected at various locations in the containment (not including the drywell) with particular emphasis on areas where equipment is required to survive a hydrogen generation event may exist.

(B) See Section 4 of draft Final Technical Report on "Design of a 1/4-Scale Test Facility to Model Hydrogen Burning in Mark III Nuclear Containments" (April 1983), and Section 3 of Addendum to draft Final Technical Report (May 1983).

(C) The 1/4 scale test facility is located at the Factory Mutual Research Corporation Test Center in West Glocester, Rhode Island.

(D) Froude modeling is used for simulating the burning of hydrogen or diffusion flames in the 1/4 scale test program. The technical basis for Froude Modeling is discussed in "Final Technical Report - Reduced-scale modeling of Diffusion Flames in Enclosures (The Case of Hydrogen Burning in the Mark III Containment)" (February 1983).

(E) No tests have been performed in the facility.

(F) See response to item (E) above.

(G) See response to item (E) above.

(H) See letter from T. Novak, NRC, to S.H. Hobbs, Chairman, Hydrogen Control Owners Group dated December 8, 1983, and transcript of March 2, 1983 meeting between NRC and Hydrogen Control Owners Group.

13-113 (A) Identify all persons responsible for the answers to the above interrogatories; for each person identified, list the interrogatories answered by that person and give the address, employer, and professional qualifications of the person;

(B) Identify all documents used in answering the above interrogatories.

Response:

(A) The persons responsible for the answers to the above interrogatories are John A. Richardson of Enercon Services, Inc., and G. Martin Fuls of GMF Associates. The interrogatories for which Mr. Richardson and Mr. Fuls are responsible are listed in the attached affidavits. Resumes for Mr. Richardson and Mr. Fuls are also attached.

(B) The documents are identified in response to the individual interrogatories.

Request for Production of Documents

1. Each and every document identified or described in the answers to the interrogatories above.

Response:

Applicants will supply to OCRE for inspection at the site upon request the documents identified or described in Applicants' voluntary answers to OCRE's Thirteenth Set of Interrogatories to Applicants.

2. Design and fabrication of steel containment vessels and related items for reactor buildings 1 and 2, Perry Nuclear Power Plant, Units 1 and 2, SP-660-4549-00 (References 1 in Ultimate Structural Capacity Report).

Response:

A copy of the document will be supplied to OCRE for inspection at the site upon request.

3. Design report for upper and lower personnel air lock, Perry Nuclear Power Plant, Units 1 and 2, Sept. 8, 1982 (Reference 8 in Ultimate Structural Capacity Report).

Response:

A copy of the document will be supplied to OCRE for inspection at the site upon request.

4. Design report for containment equipment hatch assembly, Perry Nuclear Power Plant, Units 1 and 2, Sept. 22, 1982 (Reference 9 in Ultimate Structural Capacity Report).

Response:

A copy of the document will be supplied to OCRE for inspection at the site upon request.

Respectfully submitted,

SHAW, PITTMAN, POTTS & TROWBRIDGE

By: Jay E. Silberg

Jay E. Silberg, P.C.
Harry H. Glasspiegel

Counsel for Applicants
1800 M Street, N.W.
Washington, D.C. 20036
(202) 822-1000

Dated: December 28, 1984

AFFIDAVIT

State of Florida, County of Duval

G. M. Fuls, being duly sworn according to law, deposes and says that he is the President of GMF Associates, a company which supplies consulting services to the nuclear utility industry, and that the facts set forth in the forgoing Applicant's answers to Ohio Citizens for Responsible Energy Interrogatories 13-65, 13-66, 13-67, 13-68, 13-69, 13-71, 13-72, 13-73, 13-74, 13-76, 13-77, 13-78, 13-79, 13-80, 13-83, 13-84, 13-85, 13-102, 13-104, 13-106 and 13-107 dated December 28, 1984, are true and correct to the best of his knowledge, information and belief.

G. Martin Fuls

Sworn to and subscribed
before me this 26 day
of December, 1984.

Jacquelyn Bateman

NOTARY PUBLIC STATE OF FLORIDA
MY COMMISSION EXP. OCT 15,1988
BONDED THRU GENERAL INS. UND.

SUMMARY OF EXPERIENCE

DR. G. MARTIN FULS

EDUCATION:

B.S.	Mechanical Engineering, Carnegie Mellon University
M.S.	Mechanical Engineering, Carnegie Mellon University
Ph.D.	Mechanical Engineering, University of Pittsburgh

SUMMARY OF EXPERIENCE:

Dr. Fuls has over twenty-four years of experience in nuclear power. His most recent major accomplishment has been the development of the CLASIX series of computer programs for the analysis of the nuclear power plant containment system response to a degraded core transient. This series of programs has been used exclusively by the domestic, and several foriegn, electric utilities industry for containment analyses relating to licensing activities on the degraded core, and associated hydrogen, issue. He has been continuously active in this area ever since the accident at Three Mile Island, Unit 2 in 1979.

1984-Present — President
GMF Associates

Provide consultation services to the nuclear power industry on the degraded core and hydrogen licensing issues.

1976-1984 — Advisory Engineer
Offshore Power Systems
Westinghouse Electric Corporation

Provided consultation services to the nuclear power industry in obtaining operating licenses for nuclear generating plants. Also responsible for the development of analytical methods related to obtaining a manufacturing license for floating nuclear power plants. Included the development of the CLASIX series of computer programs for the evaluation of containment response to a degraded core transients.

DR. G. MARTIN FULS

1960-1976 Principal Engineer
Bettis Atomic Power Laboratory
Westinghouse Electric Corporation

Participated in the design, testing and performance evaluation of nuclear propulsion plants for the nuclear navy program. Included plants for submarines, surface ships and the civilian applications at the Shippingport Power Station.

PUBLICATIONS:

More than 16 papers and reports.

MISCELLANEOUS:

Regestered Professional Engineer, Pennslyvania.
Who's Who in the South and Southeast.
Member, American Society of Mechanical Engineers.

State of Georgia, County of Cobb

AFFIDAVIT

John D. Richardson, being duly sworn according to law, deposes and says that he is Director, Atlanta Operations, Enercon Services, Inc., a company providing consulting services to the nuclear industry and that the facts set forth in the foregoing Applicants' Answers to Ohio Citizens for Responsible Energy Interrogatorics 13-63, 13-64, 13-66, 13-70, 13-73, 13-75, 13-79, 13-81, 13-111, and 13-112 dated December 28, 1984, are true and correct to the best of his knowledge, information, and belief.

J. Richardson

Sworn to and subscribed
before me this 28th day
of December 1984.

Hazel J. Lee
Notary Public, Georgia, State at Large
My Commission Expires Dec. 20, 1988

JOHN D. RICHARDSON

EXPERIENCE SUMMARY:

Twelve years professional engineering and engineering management experience including consulting experience in licensing, engineering, and safety analysis; nuclear plant operations management experience (EOOW qualified and SRO certified); and corporate utility management experience in licensing, safety analysis, corporate health physics, emergency planning, and environmental programs.

PROFESSIONAL EXPERIENCE:

1984 to Present — Enercon Services, Inc.

Director Atlanta Operations - General Manager of the Atlanta office. Responsible for providing consulting services to nuclear utilities. Services include licensing, safety analysis, engineering, training, plant reliability, and plant operations support.

1983 to 1984 — Tera Corporation

Project Manager/Division Manager - Responsible for providing consulting services to nuclear utilities. Projects included heavy load handling evaluations, development of criteria for assessing the safety significance of operating plant modifications, power plant reliability studies, independent design reveiws, engineering evaluations and analysis for hydrogen combustion in a Mark III containment.

1976 to 1983 — Mississippi Power & Light Company

1979 to 1983 — Manager of Nuclear Safety & Licensing - Responsible for all licensing and permitting activities with regulatory agencies, including safety and engineering analysis to support resolution of licensing issues. In addition, responsible for Corporate Health Physics, Emergency Planning, and Environmental Programs. Responsible for all interface with the NRC to obtain the

Operating License for the lead domestic BWR 6/Mark III and to develop the Company position and response to NRC enforcement actions. Responsible for safety analysis to support licensing issues including all containment response analysis to hydrogen combustion. Additional responsibilities included representing the Company on industry sponsored owners groups, and function as Chairman of the Mark III Hydrogen Control Owners Group and Chairman of the Containment Issues Owners Group (Humphrey Issues).

1977 to 1979 Operations Supervisor - Overall responsibilty for the initial staffing and training of the Operations Section and the development of the operations program for Grand Gulf Nuclear Station. Overall management of Operations Section concerning all matters of plant operation through initial preoperational testing. Other duties included review and approval of all operating procedures, development of Radwaste Management Program, development of Surveillance Test Program, development of Fire Protection Program, and development of Emergency Plan Implementing Instructions. Member of Plant Safety Review Committee.

1977 Assistant Operations Supervisor - Scheduled and coordinated operations activities including preparation and review of procedures and software required to support startup, licensing and plant operations. Prepared Operations Section Administrative Procedures and functioned as Operations Supervisor in his absence. Participated in the SRO Cold License Certification Training Program conducted by General Electric and was certified SRO at the Morris, Illinois, Dresden Simulator.

1976 to 1977 Shift Supervisor - Supervised and trained operators in preparation for NRC licensing. Design review of plant systems and licensing issues pertaining to plant operations. Preparation of software to support startup, licensing and plant operations. Project Leader for the development of System Descriptions and Lesson Plans used in operator license training.

<u>Westinghouse Electric Corporation, Naval Reactors Facility</u>

1974 to 1976	Nuclear Plant Engineer/Acting Shift Supervisor - Qualified Engineering Officer of the Watch (EOOW), Nuclear Plant Engineer, and Shift Supervisor at a dual reactor naval prototype (AIW). Responsibilities included the supervision of an operating crew of naval and civilian personnel to ensure proper. safe, and efficient plant operation. Coordination and supervision of all maintenance, testing, and training activities during a shift. Qualification Board member for Final Evaluation Watches and Final Oral Boards.

<u>Motorola, Inc., Government Electronics Division</u>

1973 to 1974	Electronic Design Engineer - Design, fabrication and testing of state-of-the-art satellite communications systems.

EDUCATION:

1968 - 1972	B. S. Electrical Engineering, Louisiana State University, Baton Rouge, Louisiana
1973 - 1974	Arizona State University, Tempe, Arizona. Work toward Master of Science - Electrical Engineering.
1979 - 1981	Master of Business Administration, Mississippi College, Clinton, Mississippi

PROFESSIONAL TRAINING:

1974 - 1975	Westinghouse Nuclear Plant Engineer School, EOOW Prototype Training, Shift Supervisor Training School, Quality Control School, Potential Commanding Officer Radiological Controls and Chemistry School.
1977	Basic Reactor Fundamentals, Memphis State University. (22 weeks)
1977	Dresden BWR Technology, BWR Operator Training including SRO certification.
1978	Grand Gulf Technology, Startup Station Nuclear Engineers Course.

ASSOCIATIONS:

Institute of Electrical and Electronic Engineers
National Society of Professional Engineers

REGISTRATIONS:

Registered Professional Engineer in Mississippi

December 28, 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' Second 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 28th day of December, 1984, to all those persons on the attached Service List.

Jay E. Silberg

JAY E. SILBERG

Dated: December 28, 1984

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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

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