UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
CLEVELAND ELECTRIC ILLUMINATING COMPANY, ET AL.	)	Docket No. 50-440 OL 50-441 OL
	)	
(Perry Nuclear Power Plant, Units 1 and 2)	)	

NRC STAFF ANSWERS TO OCRE SIXTH
SET OF INTERROGATORIES
AND BOARD QUESTIONS TO NRC STAFF

On September 13, 1982, Ohio Citizens for Responsible Energy (OCRE) filed its "Sixth Set of Interrogatories to the NRC Staff." By letter dated October 29, 1982, NRC Staff Counsel advised OCRE's Representative that the Staff would not voluntarily respond to OCRE's interrogatories. By its Memorandum and Order (Concerning Discovery from Staff on Hydrogen Issues) dated December 23, 1982, the Licensing Board directed the Staff over its objections to provide answers not only to certain of OCRE's "Sixth Set of Interrogatories," but also to five Board questions.

The Staff's answers (with the affidavits of their preparers) to OCRE's interrogatories (except Nos. 6-4, 6-8 and 6-36 which were either withdrawn or not pursued) and the Board's questions are attached.

Respectfully submitted,

James M. Cutchin IV
Counsel for NRC Staff

Dated at Bethesda, Maryland
this 3rd day of February, 1983

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ANSWERS TO OCRE'S SIXTH SET
OF INTERROGATORIES TO NRC STAFF

INTERROGATORY NO. 6-1

What does the Staff consider to be the equivalent of a TMI-2 accident at Perry? Provide the probability of its occurrence and a thorough description of its consequences, including fuel failure modes, effect on containment integrity, and off-site doses to the public at 2, 5, 10, and 50 miles from PNPP.

ANSWER

The Staff has concluded that there is no TMI-2 type accident scenario (loss of coolant accident compounded by one or more human errors) for Perry or any BWR that both can result in core damage and is credible. (Also see the answers to Board Questions No. 1,2 and 3.)

INTERROGATORY NO. 6-2

What does the Staff consider to be the worst-case accident in terms of H_2 generation at Perry? Provide the probability of its occurrence and a thorough description of its consequences, including fuel failure modes, effect on containment integrity, and off-site doses to the public at 2, 5, 10, and 50 miles from PNPP.

ANSWER

The worst-case TMI-type accident which could occur for a BWR/6 such as Perry and which is considered by the Staff to be credible would not result in fuel failure or hydrogen generation. Several scenarios would

be approximately equal in severity. One such scenario would involve a loss of feedwater, a stuck open relief valve (small break LOCA), failure of the control rod drive pumps, and the operator inhibiting injection from HPCS or RCIC. If power were available to the containment coolers for this scenario, the ADS would not automatically actuate because the drywell pressure would not reach the permissive setpoint for automatic ADS actuation. If the operator were to fail to manually actuate the ADS, the reactor vessel water level would decrease to near the top of the core before the remaining high pressure system could make up the inventory lost through the stuck open relief valve. No fuel failure or hydrogen generation would result from this event. Any other small break LOCA inside the containment would trigger the drywell pressure permissive for ADS and would be less severe than the stuck open relief valve scenario.

A design basis LOCA analyzed with conservative licensing models shows some core heat up and hydrogen generation but the peak cladding temperature and hydrogen generation are within 10 CFR 50.46 limits. A design basis LOCA analyzed with "best estimate" models shows very limited core heat up (peak cladding temperature of approximately 1000° F) and would result in essentially no hydrogen generation. The design basis LOCA is not a TMI-2 type of event.

An accident which could result in fuel damage and hydrogen generation for a BWR/6 would involve many systems failures and/or operator errors and is judged by the Staff to be incredible. One such accident could occur as follows:

A loss of feedwater flow could occur from pump failures, feedwater controller failures, or reactor system variables such as a high vessel water level trip signal. Following loss of feedwater flow, the reactor power would start to drop, the turbine control valves would start to close to regulate pressure, the water inventory in the reactor vessel would continuously decrease, and reactor trip would occur on a low vessel water level (Level 3) trip. Decay and stored heat would continue to generate steam after reactor trip, and the reactor vessel water inventory would continue to decrease. When the low-low vessel water level (Level 2) trip was reached, the recirculation pumps would trip, and all of the primary system isolation valves, including the MSIVs, would receive a signal to close. HPCS and RCIC could fail to inject into the vessel due to hardware failure or operator error. The closure of the MSIVs would cause the vessel pressure to rise to the lowest relief valve setpoint. The relief valves would open automatically to relieve the overpressure. When the pressure dropped below the reclosure setpoint one of the relief valves could fail to close. The control rod drive pumps could fail, the drywell pressure could remain below the ADS permissive and the operators could fail to manually actuate the ADS. The operators could also fail to open any other safety/relief valve to reduce the vessel pressure. For this scenario, the core would be uncovered and hydrogen generation would occur. The probability of such an event has not been quantified, but is judged by the Staff to be extremely low because <u>all</u> of the following failures would have to occur:

1) Loss of feedwater Because Perry has two 60 percent capacity turbine driven feed pumps and one 20 percent capacity motor driven feed pump, a total loss of feedwater for more than a few minutes is not very likely. If some feedwater were restored, there would be no problem.

2) HPCS and RCIC failure This would require a failure of two safety-related high pressure systems which have diverse power sources. HPCS may be powered by either offsite power or its own dedicated diesel. RCIC is powered by reactor steam.

3) A stuck-open safety/relief valve

4) Failure of the control rod drive (CRD) pumps There are two CRD pumps which may be powered by either offsite power or by the station diesels. One of these pumps would be operating prior to initiation of the accident.

5) Failure of automatic actuation of ADS and failure of the operators to manually actuate the ADS If the drywell pressure increased to the ADS pressure permissive, multiple failures would be required to prevent automatic ADS. If the drywell pressure did not increase, power would be available to the containment coolers and hence to other high pressure injection systems. The operators would have to ignore multiple indications of low water level and inoperable injection systems even though symptomatic procedures and extensive training would instruct the operators to manually initiate ADS.

The Staff judges such a combination of failures to be so unlikely as to be incredible.

INTERROGATORY NO. 6-3

Has the Staff (or anyone on its behalf of to its knowledge) performed MARCH code calculations specific to Perry for any accident sequences? If so, produce these analyses. If Perry-specific calculations have not been performed, produce all MARCH code analyses performed for Grand Gulf (most useful are graphical presentations of the calculated parameters versus time, e.g., pp. C-13 to C-44 of NUREG/CR-1659, Volume 4).

ANSWER

Neither the Staff (nor anyone on its behalf or to its knowledge) has performed MARCH Code calculations specific to Perry for any accident sequences. MARCH Code calculations were performed for Grand Gulf by a Staff contractor (Brookhaven National Laboratory). The reports of the MARCH Code calculations for Grand Gulf are available for inspection and copying in the NRC Public Document Room. (See the answer to Interrogatory No. 6-11 for the titles of the documents.)

INTERROGATORY NO. 6-5

Commissioner Gilinsky has stated that the Mark III is a weak containment that should be required to be stronger (47 FR 2300, January 15, 1982). How could the Perry containment be strengthened? Include a cost estimate of all measures that could strengthen the Perry containment.

ANSWER

The Mark III containment does have a lower ultimate capacity when compared against PWR dry containments. However, it is designed with ample safety margin to perform the required containment functions consistent with the Mark III pressure suppression system characteristics during and after various postulated design basis accidents. From a safety standpoint the Staff does not believe it is necessary to strengthen the Perry containment. Were some means of strengthening to be considered, a complete design study would have to be performed before the feasibility of such an effort could be established. The cost estimate would depend totally on the type of strengthening measure chosen. No such analyses or cost estimates have been performed.

INTERROGATORY NO. 6-6

SECY-80-107A contains view-graphs presented by General Electric to the NRC which state that containment inerting, hydrogen ignition, recombiners, and purging are all impractical for significant rates of H_2 production. Does the Staff agree? If not, why not?

ANSWER

The Staff agrees that recombiners and simple purging are impractical for controlling hydrogen produced at significant rates. However, the hydrogen ignition system is practical to accommodate hydrogen production at significant rates.

Also, in reviewing SECY-80-107A, the Staff finds that the view-graphs provided by GE do not state explicitly that containment inerting is impractical for significant rates of hydrogen production.

INTERROGATORY NO. 6-7

The Commission has stated that hydrogen 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). What are the bases for this statement?

ANSWER

The statement is based on the realization that sequences are conceivable that would not be mitigated by a burning system but which might be mitigated by an inerting system. Such sequences would include:

1. Loss of all AC power, so that no power would be available to operate igniters.
2. Generation and release of hydrogen at such a high rate that very high pressure could not be avoided when the hydrogen is ignited.

However, the probabilistic safety study of the Grand Gulf plant has not identified these as sequences that dominate risk.

INTERROGATORY NO. 6-9

What is the status of the proposed rule to 10 CFR Part 50, "Interim Requirements Related to Hydrogen Control," 47 FR 62281, December 23, 1981?

ANSWER

The NRC Staff has completed its review of public comments received on the proposed rule and has drafted a Commission paper which is currently undergoing review and concurrence by appropriate NRC offices. Upon completion of this review, the Commission paper will be sent to the

Committee on Review of Generic Requirements (CRGR) for concurrence, and then to the Commission. A final rule could be published by May 1983.

(This response was prepared based on consultations with Morton R. Fleishman, NRC Regulatory Analysis Branch, Division of Risk Analysis, Office of Nuclear Regulatory Research, who is the designated Staff contact for further information on the proposed rule.)

INTERROGATORY NO. 6-10

What types of hydrogen control systems are available for preventing H_2 buildup and/or explosion in Mark III containments? Briefly discuss each system, listing the advantages and disadvantages of each. Which system is favored by the Staff? Why?

ANSWER

Depending on the manner in which hydrogen is being generated, there are basically two major hydrogen control systems available; hydrogen recombiners and distributed ignition systems. Hydrogen recombiners are used for Design Basis Accidents in which the bulk of the hydrogen is generated by radiolysis and metal corrosion. The zircaloy metal-water reaction is small for Design Basis Accidents. Thus, the hydrogen recombiners are acceptable for use in preventing the build-up of hydrogen from the slow hydrogen producing mechanisms which would exist for Design Basis Accidents.

For degraded core accidents (beyond Design Basis Accidents) significant amounts of hydrogen would be produced from the zircaloy metal-water reaction. The distributed ignition system is the hydrogen control approach employed by the Mark III BWR Plant Owners for that situation. The hydrogen ignition system glow plugs have been the most widely researched. Some of the advantages and disadvantages of the hydrogen ignition system are as follows:

Advantages:

This system would burn hydrogen as it is released to maintain concentrations below the theoretical detonable limit. The system would consist of thermal glow plugs and associated electrical equipment distributed throughout the containment and drywell

Inadvertent operation of the system during normal plant operation would not subject the containment to a pressure transient. Inadvertent or planned operation of the system following a LOCA-type event would have no effect on containment pressure unless hydrogen were present.

The system utilizes a simple design which is conducive to high reliability. The system is capable of being tested during plant operation.

Disadvantages

The mode of burning hydrogen adds heat to the containment representing a challenge to containment integrity by both increasing pressure and temperature. The resulting high temperatures and pressure could represent a challenge to the functionability of essential equipment in containment.

The hydrogen control systems identified above are the only systems proposed by CEI. Since the recombiner system is designed to be used for Design Basis Accident events and the ignition system is designed for use during degraded core events, any relative ranking of the designs by the Staff would be inappropriate.

INTERROGATORY NO. 6-11

It is stated in the discussion of the proposed rule (46 FR 62282) that there are ongoing programs of research pertaining to hydrogen generation, release, burning, and control. Please list all such research programs. Briefly describe the status of each, along with any interim findings and the expected date of completion and publication of results.

ANSWER

The following are research programs on hydrogen generation, release, burning and control. The status and the findings of each are provided.

A. Title/FIN No.: Review of Grand Gulf Hydrogen Igniter System (A1308)

Office: NRR/DSI/CSB

Start Date: 7/15/81

Completion Date: 1/15/82

Status/Results:

This program is complete. A final report is now in preparation. The results of the research have been used in the Grand Gulf review as discussed in Grand Gulf SER Supplement No. 3 (NUREG-0831, July 1982).

B. Title/FIN No.: Design Basis for Hydrogen Control Systems (A3389)

Office: NRR/DSI/RSB

Start Date: 9/1/81

Completion Date: 1/30/83

Status/Results:

This is a contract with Brookhaven National Laboratories (BNL). Grand Gulf, Perry and Clinton were the facilities studied. Reports have been prepared by BNL on all of these plants and are available in the NRC Public Document Room. BNL has estimated hydrogen release schedules for various degraded core scenarios including periods of recovery from the accident. The results are given in the following reports:

An Assessment of Postulated Degraded Core Accidents in the Grand Gulf Reactor Plant (DRAFT), R.D. Gasser, (June 1982)

Analysis of Full Core Meltdown Accidents in the Grand Gulf Reactor Plant, R.D. Gasser, (August 1982)

March Analysis of Hydrogen Burning During Degraded Core Accidents for the Clinton Power Station, BNL Memorandum, (September 27, 1982)

Degraded Core Accidents in the Perry Power Plant, BNL Memorandum, (December 13, 1982)

C. Title/FIN. No.: Hydrogen Burn Survival (HBS) (A1270)

Office: RES/DAE/SAAB

Start Date: 8/81

Status, Interim Finding and Completion Dates:

At the present time, the results of this study, including an analysis of the equipment survivability of components in the Sequoyah environment and the data from experiments are being evaluated. The analysis consisted of two separate activities: (a) an estimate of component temperatures in response to a given gas temperature profile for Sequoyah; and (b) several HECTR code calculations of hydrogen burns in the Sequoyah containment. The first activity led to the conclusion that for the gas temperature profile given, our calculations did not yield component temperatures any higher than those projected by TVA.

The preliminary results of HECTR analysis suggests that small changes in the assumptions regarding combustion may result in relatively large differences in equipment temperatures. More work is underway to investigate this matter.

At the present time Sandia test data are being used to verify the calculational models for ice condenser plants. In general, the phenomena are well represented, but we are still examining the degree of detailed quantitative agreement.

With respect to proposed completion dates, the HBS program plan is scheduled for updating in February 1983. The current plan indicates that information from the analysis and experimental activities should be frozen as of approximately April 1, 1983, and the development of an algorithm based on that data must be initiated. The draft of the algorithm (i.e., an analytical procedure which can be used to estimate the temperature response of components to hydrogen burns) is to be provided by October, 1983. Recent developments in the licensing of ice condenser plans and the need to produce a report covering the deliverables might need to be reconsidered. A detailed Sequoyah report in July 1983 and a more general algorithm capable of treating BWRs are possibilities to be considered in the program planning update meeting.

D. Title/FIN No.: Hydrogen Combustion Mitigative and Preventive Schemes (A1336)

Office: RES/DET/CEBR

Start Date: 2/5/82

Completion Date: Decision on Practicality of Form Studies - 5/83
Decision on Feasibility of Water/Fog/Charged Droplet Studies - 6/83

Interim/Results -

(1) NUREG/CR-2865 September 1982, "Hydrogen Combustion in Aqueous Foams." This study concludes that with hydrogen concentrations in the range of 15% - 28%, aqueous foam, in combination with deliberate ignition, causes a reduction in the pressure rise of up to a factor of two and one-half. Despite this overall pressure reduction, the flame speed is accelerated by more than an order of magnitude for combustion in the foam. However, no evidence of detonations was observed in small-scale tests.

E. Title/FIN No.: Containment Safety Under Internal Explosions (B7085)

Office: RES/DET/MSEB

Start Date: 7/25/80

Completion Date: 5/6/82

Results:

(1) NUREG/CR-2897 (September 1982), "Calculations of Hydrogen Detonations in Nuclear Containments by Random Choice Method." Computer codes (CRTDET, SPHDET and TWODIM) for solving the hydrogen detonation problem in the containment of a nuclear reactor were developed and used.

(2) NUREG/CR-2898 (September 1982), "Reinforced Concrete Containment Safety Under Hydrogen Explosion Loading." The Indian Point Unit 3, Nuclear Power Plant, was used for a study of the behavior and safety of a typical steel-lined reinforced concrete containment structure under postulated hydrogen deflagrations and/or detonations. Static analyses of deflagration yielded structural capacities from 2.98 to 3.62 (170 psig) times design pressure. Using very conservative assumptions, dynamic analyses indicated failure close to cylinder midheight in the hoop direction for detonation initiation at midheight, and failure around the dome apex for initiation at the base.

F. Title/FIN No.: Hydrogen Behavior Program (A1246)

Office: RES/DAE/SAAB

Start Date: 8/80

Status of Research

Over the past two years the hydrogen behavior program has been focusing on providing an improved technical understanding of hydrogen generation, transport and combustion phenomena. The program is both analytical and experimental, with a main portion of the analytical effort directed at developing a combustion code, HECTR, for reactor accident analysis. The code has been used in Sequoyah accident analysis and is currently being used to analyze a broad spectrum of accident sequences for Grand Gulf. Additionally, analytical work has been

initiated to model flame acceleration phenomena and to modify existing codes for hydrogen transport. The experimental program is being carried out principally in three facilities; an assortment of 16 foot insulated tanks called the Variable Geometry Experimental System (VGES) for both deflagration and detonation studies, an insulated 5 cubic meter tank called the Fully Instrumented Test System (FITS) for deflagration and detonation studies in steam and a hydrogen steam jet facility to study auto-ignition of H_2 steam: air mixtures. Additionally, small scale flow acceleration work is underway at McGill University in Canada and a large scale flame acceleration facility is being constructed at Sandia National Laboratory.

Interim Finding

- Interim results have confirmed other recent studies that combustion completeness and the subsequent pressures and temperatures are a function of the physical and chemical properties of the gases, the type of igniter, the nature and geometry of obstacles, and the initial velocity of the gases.

- An important result from recent code calculations is that the occurrence of a detonation is not necessarily tantamount to containment failure, containments could survive a variety of postulated local (i.e., limited in size) detonations.

- The deliberate ignition scheme has been employed in several plants and calculations indicate that a reduction in risk can result for certain postulated accident scenarios.

- The hydrogen steam jet work has shown that once ignited a stable flame will be established for releases covering a range of conditions (i.e., flow rates, concentrations and gas temperatures). Auto-ignition results are similar to those previously reported.

Schedule

- Within a year most experimental research on ordinary deflagrations in intermediate scale vessels will be completed.

- Small scale experiments on accelerated flames and detonations will be completed within two years. Large scale flame acceleration studies will begin this year.

- Modelling efforts for all combustion phenomena will continue for the next few years.

- Dependir on what is learned most of the research is expected to be completed in the next few years.

Reports:

A Review of H_2 Detection in Light Water Reactor Containment. E. C. Neidel, J. G. Castle, Jr., J. E. Gover, NUREG/CR-2080, SAND81-0326 (published 12/81).

CSQ Calculations of H_2 Detonations in the Zion and Sequoyah Nuclear Plants, R. K. Byers, NUREG/CR-2385, SAND81-2216, (published 9/82).

An Evaluation of the RALOC Computer Code, L. D. Buxton, D. Tomasko, G. Padilla, and J. Orman, NUREG/CR-2764, SAND82-1054, (published 8/82).

X-ray Measurements of Water Fog Density, A. L. Camp, SAND82-1292, (published 11/82).

Reports Anticipated for Fiscal Year 1983:

Light Water Reactor Safety Research Program Semiannual Report, April-September 1982, M. Berman, editor. second quarter.

Light Water Reactor Safety Research Program Semiannual Report, October, 1982-March 1983, M. Berman, editor, fourth quarter.

Accident Analysis Calculations for Zion, Sequoyah, and Grand Gulf, B. W. Burnham, to be published during the 3rd quarter of 1983.

Light Water Reactor Hydrogen Manual, A.L. Camp, C. Kupiec, et. al, to be published during the 3rd quarter of 1983.

Proceedings of the 2nd International Workshop on the Impact of hydrogen on Water Reactor Safety, October 3-7, 1982, M. Berman, editor third quarter.

Experiments in the VGES 16-ft Tank, J. C. Cummings, W. B. Bendick, P. G. Prassinos, topical report to be published during the third quarter of 1983.

The FITS Experimental Facility and Hydrogen:Air:Steam Deflagrations, S.F. Roller, et al., topical report to be published during the 4th quarter of 1983.

Users Manual for the HECTR Code - 3rd Quarter 1983.

The Steam:Hydrogen Jet Facility, Autoignition and Flame Stability Experiments, J. Shepherd, et al., topical report to be published during the 4th quarter of 1983.

Codes for Predicting Pressures and Temperatures During and After Hydrogen Combustion in Containments, M. P. herman A. L. Camp, et al., topical report to be published during the 4th quarter.

Flame Propagation and Flame Acceleration Computer Codes, B. R. Sanuers, et al., topical report to be published during the 4th quarter of 1983.

Flame Acceleration, DDT, and Detonation Experiments at McGill University, J. H. S. Lee, et al., topical report to be published during the 4th quarter of 1983.

INTERROGATORY NO. 6-12

SECY-80-107 at p. 30 states that the Staff believes that the Mark III containment has a failure pressure of at least twice the design pressure.

(a) Is this estimate based on static or dynamic pressures?

(b) Provide all factual bases and experimental evidence supporting this belief.

ANSWER

(a) The estimate is based on a static pressure analysis.

(b) The above-noted Staff belief is based on the results of an analysis provided by the Applicant in an enclosure to a letter from Dalwyn R. Davidson to Robert L. Tedesco, dated January 25, 1982 and titled, "Perry Nuclear Power Plant Units 1 and 2 Ultimate Structural Capacity of Mark III Containments." This letter and its enclosure are available at the NRC Public Document Room. The design pressure for Perry containment is 15 psig. and the corresponding failure pressure as estimated by the analysis is 68 psig, which is approximately four and one-half times the design pressure.

INTERROGATORY NO. 6-13

Has the Staff performed any analyses on the ultimate strength of the Perry containment? If so, produce them. Discuss all assumptions, judgments, and approximations made in the analyses and the bases for them.

ANSWER

The Staff did not perform analyses on the ultimate strength of Perry containment. The ultimate capacity analysis of Perry containment was performed by the Applicant and reviewed by the Staff. (See the answer to Interrogatory No. 6-12(b) above.)

INTERROGATORY NO. 6-14

At what range of concentrations (volume-%) of H_2 are recombiners of the type to be used at PNPP effective in reducing the H_2 concentration below flammable limits?

ANSWER

The hydrogen recombiners that are to be used at PNPP are manufactured by Westinghouse Electric Corporation. In tests of the recombiners conducted by Westinghouse over a range of hydrogen concentrations from zero to 4.6%, recombiner efficiency was essentially 100%.

INTERROGATORY NO. 6-15

If the recombiners were ineffective in reducing H_2 concentrations, would the recombiners become an ignition hazard? At what H_2 concentration?

ANSWER

The heater section of the hydrogen recombiner at PNPP consists of a thermally insulated vertical metal duct with a metal sheathed electric resistance heater provided to heat a continuous flow of containment gas mixture up to a temperature which is sufficiently high to react the hydrogen and oxygen. The gas mixture enters the recombiner and flows up through the heater section and out the top by natural convection. No circulation fans are required and the air flow rate is established by an orifice plate at the bottom of the recombiner. The most plausible reason the recombiner would be ineffective in reducing the hydrogen concentrations would be if the heater failed. In this case, the recombiner, obviously would not become an ignition hazard. Hydrogen concentrations substantially higher than 4.6% would signify that a degraded core accident rather than a Design Basis Accident had occurred. In that event, ignition of the mixture, whether by the recombiner or by the glow plugs, would be acceptable.

INTERROGATORY NO. 6-16

At what range of H_2 concentrations (volume-%) are glow plug igniters effective in reducing H_2 concentrations below flammable limits?

ANSWER

Glow plug igniters are effective in initiating combustion at hydrogen concentrations of approximately four volume percent (the flammable limit) and above.

INTERROGATORY NO. 6-17

Does the Staff believe that the igniters could pose a hazard to the integrity of the containment and the equipment therein by causing severe detonations?

ANSWER

The Staff does not believe that igniters such as those proposed for the Perry plant could cause a severe detonation. In the event of large hydrogen releases, the distributed ignition system allows for a controlled burning of lean hydrogen mixtures in the containment, thus precluding the formation of large volumes of hydrogen at detonable concentrations.

INTERROGATORY NO. 6-18

Does the Staff believe that the normal, expected operation of the igniters (controlled ignition) could pose a threat to the integrity of the containment or the equipment therein by causing high temperatures and cyclic pressure pulses?

ANSWER

The Staff does not believe that the normal, expected operation of the igniters could pose a threat to the integrity of the containment or the equipment therein by causing high temperatures and cyclic pressure pulses.

INTERROGATORY NO. 6-19

In the Staff's opinion, has the Perry hydrogen control system met the requirements of GDC 41, 42, and 43 of 10 CFR Part 50? List all criteria not met.

ANSWER

In the Staff's opinion, the only Perry hydrogen control system that is required by NRC regulations (the recombiner system) meets the requirements of GDC, 41, 42, and 43 of 10 CFR Part 50. See Perry SER (NUREG-0887), pages 6-14 and 6-15. The distributed ignition system design is not required to meet these criteria, and thus was not reviewed for conformance with these criteria, because the ignition system is required neither for mitigating Design Basis Accidents nor by existing NRC regulations.

INTERROGATORY NO. 6-20

Has the Staff analyzed the Perry containment for sources of ignition? If so, produce the results of the analysis.

ANSWER

The Staff has not analyzed the Perry containment for sources of ignition.

INTERROGATORY NO. 6-21

Has the Staff analyzed the Perry hydrogen control system against all applicable regulations, regulatory guides, branch technical positions, and other standards? If so, produce the results of this analysis, especially describing any instances in which criteria and guidelines

have not been met. If this analysis has not been performed, when does the Staff intend to do so?

ANSWER

The Staff has concluded that the only Perry hydrogen control system that is required by NRC regulations (the recombiner system) conforms with all applicable regulations, regulatory guides, branch technical positions and industry standards. The results of the Staff's evaluation are documented in the Perry SER (NUREG-0887), pages 6-14 and 6-15. (Also see the answer to Interrogatory No. 6-19).

INTERROGATORY NO. 6-22

FSAR Section 6.2.5.2.1 states that delaying the start of the analyzers until 15-60 minutes following the LOCA will avoid exposing the analyzer to severe sample conditions. In the Staff's opinion, can severe conditions persist beyond 15-60 minutes after the LOCA? After transient sequences?

ANSWER

In the Staff's opinion, severe environmental conditions will not persist beyond 15 minutes following the onset of a design basis LOCA, and the environmental conditions associated with small break LOCA's are less severe than those for the design basis LOCA. Because the hydrogen analyzers are designed to withstand, and function at, peak design basis LOCA conditions, the expected environmental conditions for lesser transients such as small break LOCA's are bounded. See Perry FSAR, Section 6.2.5.2.1. Thus, the conditions for which the analyzers are to

be qualified envelope the peak conditions for the accidents and transients of interest.

INTERROGATORY NO. 6-23

In the Staff's opinion, for containment H_2 concentrations above 4 vol-%, would the mixers accelerate combustion by providing a uniformly combustible atmosphere in the containment? Why or why not?

ANSWER

It is the Staff's opinion that for containment hydrogen concentrations above four volume percent the mixers would not accelerate combustion by providing a uniformly combustible atmosphere in the containment, because the mixers are designed to produce relatively low flow rates.

INTERROGATORY NO. 6-24

In the Staff's opinion, could the ignition of hydrogen by the glow plugs produce missiles that could damage the containment or equipment therein?

ANSWER

In the Staff's opinion, the ignition of hydrogen by the glow plugs would not produce missiles which could damage the containment or equipment. This view is based on the relatively slow rates of combustion expected for the lean mixtures that would be ignited by the glow plugs.

INTERROGATORY NO. 6-25

Provide off-site radiation doses (whole body and thyroid) to the public at 2, 5, 10, and 50 miles from PNPP resulting from containment purge.

(a) what the Staff considers to be the equivalent of a TMI-2 accident at Perry;

(b) what the Staff considers to be the worst-case accident in terms of H_2 generation for Perry;

(c) the following accident sequences as defined in NUREG/CR-1659, Volume 4 (RSS Methodology applied to Grand Gulf):

(1) AI	(7) T_1PQI	(13) T_1C
(2) AE	(8) T_1PQE	(14) T_1QUW
(3) AC	(9) $T_{23}PQI$	(15) $T_{23}C$
(4) SI	(10) $T_{23}PQE$	(16) $T_{23}QW$
(5) SC	(11) T_1QW	(17) $T_{23}QUW$
(6) SE	(12) T_1QUV	(18) $T_{23}QUV$

ANSWER

The staff knows of no circumstances under which containment purge would be a recommended response within several months following any of the twenty accident sequences listed in the question. As was done at TMI-2, containment purge would eventually occur in order to remove long-lived ^{85}Kr from the containment atmosphere for purposes of re-entry. After several months ^{85}Kr would be the sole radioisotope present in the containment atmosphere in amounts detectable off-site during purge, and the off-site consequences would be identical for all accidents involving massive core damage.

The maximum integrated whole body doses at the four requested distances, assuming continuous purge of the entire ^{85}Kr inventory over several days, are given in the table below.

Distance (miles)	Integrated Whole Body Dose (millirem)
2	0.01
5	.003
10	0.001
50	0.0001

Most of the accident sequences listed involve multiple failures of redundant safety systems, are beyond the design bases considered in the safety review, and could lead to containment failure by mechanisms not involving hydrogen. It would be pointless to purge a failed containment. The consequences and risks associated with accidents beyond the design bases of the Perry units have been considered, and are described in Section 5 of NUREG-0884, the Perry Final Environmental Statement.

INTERROGATORY 6-26

In the Staff's opinion, would overpressure from H_2 production alone (no explosion) be sufficient to rupture the containment? From what % metal-water reaction?

ANSWER

In the Staff's opinion, hydrogen production alone, even for 100% metal-water reaction, would not create pressures high enough to rupture the containment.

INTERROGATORY NO. 6-27

Describe the pressure and temperature transients which would be experienced by the containment from the complete combustion of the following concentrations of hydrogen (vol-%, assume abundant oxygen):

(a) 4%
(b) 6%
(c) 9%
(d) 12%
(e) 18%
(f) 24%
(g) 33%

ANSWER

The Hydrogen Control Owners Group (HCOG) performed a sensitivity study that includes temperature and pressure profiles in the Grand Gulf containment for combustion initiated at approximately 8% hydrogen and for burning to 85% completeness. Representative profiles (labelled Figures 43 and 46) which were extracted from the report, "CLASIX-3 Containment Response Sensitivity Analysis" submitted by the HCOG with a letter from J. Richardson to H. Denton dated January 15, 1982 are attached. Containment transients outside the range of this study have not been analyzed. The entire report is available in the NRC PDR.

INTERROGATORY NO. 6-28

Are the results given above based on any experimental data or studies specific to either the Perry or the generic Mark III containment? Produce all such studies.

ANSWER

The results are based on sensitivity studies for the Grand Gulf Nuclear Station using the CLASIX-3 computer code to evaluate the Mark III containment response to hydrogen burn transients. As noted in the

answer to Interrogatory No. 6-27, the entire report is available for inspection and copying in the NRC PDR.

INTERROGATORY NO. 6-29

List any assumptions made in the preparation of such studies, e.g., regarding the quenching effects of steam/humidity or the effect of containment structures and equipment on flame fronts.

ANSWER

Studies of several cases were involved. The report consists of 31 pages of text, 24 tables and 417 figures and the assumptions cannot readily be summarized. As noted above, the entire report is available for inspection and copying in the NRC PDR.

INTERROGATORY NO. 6-30

If the Staff has performed any analyses of the Perry containment, did this analysis consider containment penetrations as possible points of rupture? If not, why not?

ANSWER

The Staff didn't perform analyses of the Perry containment. The design and final analyses of Perry containment were performed by the Applicant's vessel contractor, Newport News Industrial Corporation. The design and analyses were based on the ASME Code Section III, Subsection NE and Regulatory Guide 1.57. The analyses did consider containment penetrations as possible points of failure.

INTERROGATORY NO. 6-31

In the Staff's opinion, could blowdown through the suppression pool or direct drywell-to-containment leakage exceed the capacity of the mixers?

ANSWER

In the Staff's opinion, blowdown through the suppression pool or direct drywell-to-containment leakage would not exceed the capacity of the mixers.

INTERROGATORY NO. 6-32

In the Staff's opinion, could direct drywell-to-containment leakage dissipate hydrogen outside the area from which the recombiners take suction or outside the regions where the igniters are located?

ANSWER

In the Staff's opinion, direct drywell-to-containment leakage would not dissipate hydrogen outside the area from which the recombiners take suction or outside the regions where the igniters are located.

ANSWER NO. 6-33

In the Staff's opinion, would the drywell-to-containment differential pressure ever be great enough (e.g., after upper pool dump) that the mixer compressor head is insufficient to clear the upper suppression pool vents?

ANSWER

In the Staff's opinion, the drywell-to-containment differential pressure would never be great enough (e.g., after upper pool dump) that the mixer compressor head is insufficient to clear the upper suppression pool vents.

INTERROGATORY NO. 6-34

In the Staff's opinion, could the recombiners produce "hot spots" which could adversely affect the containment or equipment therein?

ANSWER

In the Staff's opinion, the recombiners could not produce "hot spots" which could adversely affect the containment or equipment therein.

INTERROGATORY NO. 6-35

Does the Staff consider the manual actuation of all components of the Perry H_2 control system acceptable? If so, how can this be justified, since large amounts of H_2 can be produced within minutes of core overheating (NUREG/CR-1651, pp. 36-37; SECY-80-107, p. 6)?

ANSWER

Yes, the staff considers manual actuation of the hydrogen control system to be acceptable.

Staff analyses of Grand Gulf sequences involving LOCA's, core uncovery, and hydrogen production due to a combination of system failures and human errors have shown that at least 30 minutes is available between a reduction in water level and the onset of significant hydrogen generation. This conclusion is based on MARCH analyses that examine the

entire sequence. With this amount of time (30 min) available, procedures for acceptable manual initiation of the igniters are practicable.

Accident scenarios of the TMI-2 type (LOCA followed by operator errors leading to core uncovery and hydrogen generation) are not analyzed in NUREG/CR-1561, the first reference cited. The information given on pp. 36 and 37 of NUREG/CR-1561 states that the reaction rate between steam and zirconium is such that 100 kg of hydrogen can be produced in 21 minutes or less at temperatures above 1400 K (2060°F). However, it does not indicate how long it would take to raise the temperature to this point. By itself, therefore, it is not an appropriate way of judging whether there is time for manual operation. MARCH calculations, which track the entire sequence, are a better way of determining the total time available. These calculations are given in "An Assessment of Postulated Degraded Core Accidents in the Grand Gulf Reactor Plant (Draft)", by R. D. Gasser of BNL, June 1982 (available in the PDR).

The second reference, (SECY-80-107, p. 6), is related to a sequence that is generally considered to be well beyond the class of TMI-2 related sequences. It describes the rapid hydrogen generation that could occur in the case of a large break LOCA with ECCS failure and adiabatic core heating. In this sequence core melting has been estimated to begin within 12 minutes of the start of the accident. Because of the low probability of a large break LOCA, however, this sequence is not estimated to contribute significantly to the risk of a degraded core, nor is this sequence considered to be a TMI-2 type accident.

INTERROGATORY NO. 6-37

List all documents relied upon in answering the above interrogatories, and list all persons responsible for the answers, along with their professional qualifications.

ANSWER

Documents relied upon in answering the interrogatories, if any, are identified in the answers to the interrogatories. (Also see the answer to Board Question No. 4.) Persons responsible for answering the interrogatories are identified in their attached affidavits. Professional qualifications of persons responsible for answering the interrogatories also are attached.

ANSWERS TO LICENSING BOARD QUESTIONS TO NRC STAFF

BOARD QUESTION NO. 1

What, if anything, has the staff done to develop different scenarios about a TMI-type accident (a loss of coolant accident, compounded by one or more human errors) that results in core uncovery and hydrogen generation?

ANSWER

In May, 1979, the NRC Staff, Bulletins and Orders Task Force (B&OTF), reviewed BWRs (including the BWR/6) to assess the potential for a TMI-2 type event in a BWR. The review revealed no TMI-2 type events which would result in core heat-up and hydrogen generation that are viewed by the Staff as credible.

The B&OTF was established within the NRC Office of Nuclear Reactor Regulation to review and direct the TMI-2 related staff activities associated with loss of feedwater transients and small break loss-of-coolant accidents for all operating plants to assure their continued safe operation.

Because the TMI-2 event involved a loss of feedwater, a stuck open relief valve and operator error, the same or similar failures were considered for the BWR. The B&OTF restricted its review to small break loss-of-coolant accidents and loss of feedwater events with numerous variations of systems assumed inoperable. The assumed inoperability of the systems could be attributed to either operator errors or system malfunctions. The transients analyzed and systems failures considered were specified by the NRC; the transient analyses were provided by

the BWR Owner's Group. In addition, each system (e.g., water level measurement system, RCIC, HPCS, LPCI, ADS, condensate system) was reviewed by the B&OTF to assess such critical parameters as control room indication, power source, water source, redundant systems and the effect of systems on the plant transient.

Plant systems parameters and transient analyses covering TMI-2 type events plus analyses of even more degraded events are reported in NEDO-24708A, "Additional Information Required for NRC Staff Generic Report on Boilding Water Reactors," Volumes 1 and 2, Revision 1, December, 1980. The Staff evaluation of this report is given in NUREG-0626, "Generic Evaluation of Feedwater Transients and Small Break LOCA Accidents in GE Designed Operating plants and near term operating license applications, January 1980. Both of these documents are available in the NRC Public Document Room.

There are several reasons why a TMI-2 type event is not viewed by the Staff to be credible for a BWR. These include: 1) the vessel water level is measured directly at the reactor vessel, not at a separate vessel (pressurizer); 2) operator training and procedures emphasize maintaining water in the normal control range if possible or above the top of the active fuel if the water level cannot be maintained in the normal control range; 3) if the high pressure systems are inoperable, the automatic depressurization system will quickly lower the reactor vessel pressure so that low pressure systems can inject water into the vessel; 4) recently approved emergency procedure guidelines are based on plant symptoms rather than specific events (one reason for operator error at TMI was that more than one simple event was occurring and the operator

took incorrect action using event-based procedures: symptom-based procedures would have directed the operator to take the correct actions); 5) BWRs are designed to operate with steam in the upper part of the reactor vessel; 6) power levels up to approximately 50 percent can be maintained with natural circulation flow.

The B&OTF review did identify improvements in systems, procedures and analysis which will make the GE-designed BWRs even less susceptible to core damage during accidents and transients that are coupled with systems failures and/or operator errors. The B&OTF recommendations were included in the TMI Action Plan (NUREG-0737) requirements. Perry has been reviewed against NUREG-0737 requirements.

BOARD QUESTION NO. 2

What, if anything, has staff done to determine whether such scenarios are credible?

ANSWER

No quantitative type probability analyses were performed for the scenarios described in NEDO-24708A to determine whether or not the scenarios are credible for Perry or a BWR/6. However, with the improvements made in systems, procedures and analysis, the Staff believes that the accidents and transients examined have a very low likelihood of causing core damage should they occur. This belief is based on the large number of systems failures and/or operator errors that must occur to cause core damage.

BOARD QUESTION NO. 3

Discuss whatever doubts the staff has about whether a TMI-type accident could occur at Perry or at similar BWR reactors?

ANSWER

As discussed in the answer to Board Question No. 1 above, BWRs are designed to operate with voids in the upper part of the vessel and are designed to operate with natural circulation of coolant through the core. The primary method to assess adequate core cooling in BWRs is to use direct measurement of reactor water level. Operators are trained to keep the reactor water level above the top of the core in all situations.

New generic, symptom based, emergency procedure guidelines have been developed for BWRs and will be used in the development of procedures for Perry. Should a SBLOCA occur in a Perry type BWR, the Staff believes that, because of the improvements made in systems, procedures and training in the aftermath of the accident at TMI-2, the likelihood of the occurrence of the large number of operator errors (and system malfunctions) that are necessary to cause core damage, generation of large amounts of hydrogen, containment failure or leakage and doses greater than 10 CFR 100 guideline values is so low that such an occurrence can be viewed as incredible.

BOARD QUESTION NO. 4

Provide documents and analyses that are not available in the docket room and have not been provided to OCRE in response to its Freedom of Information Act requests but that bear on the above 3 questions.

ANSWER

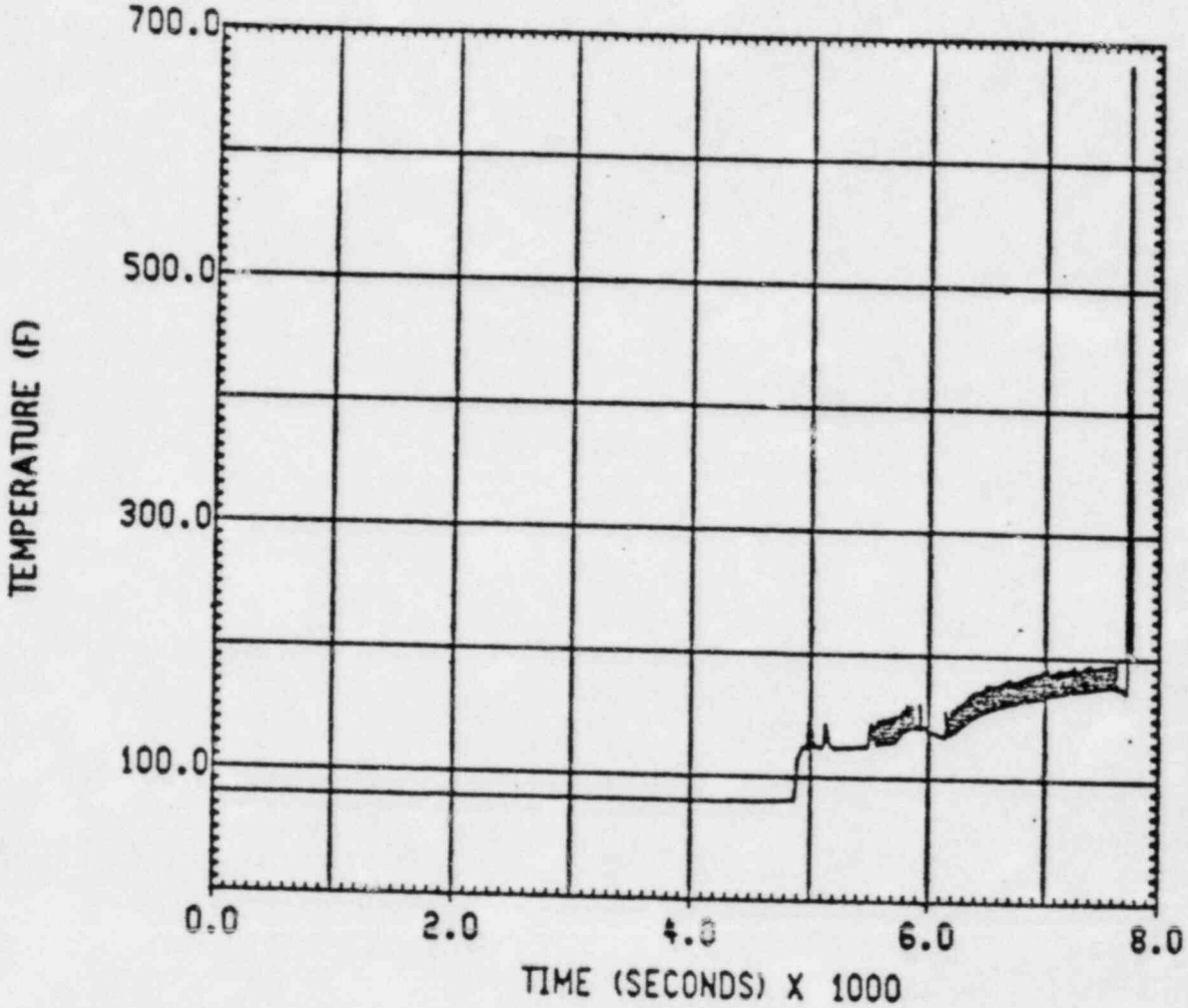
There are no documents or analyses that bear on Board Questions Nos. 1-3 and that are in the possession, custody, or control of the NRC, that are not available for inspection and copying in the NRC Public Document Room.

BOARD QUESTION NO. 5

Provide the name of any staff person or NRC consultant who, in the course of work for the NRC, prepared a memorandum or other document suggesting that there are one or more credible TMI-type accident scenarios for Perry or for similar BWR reactors.

ANSWER

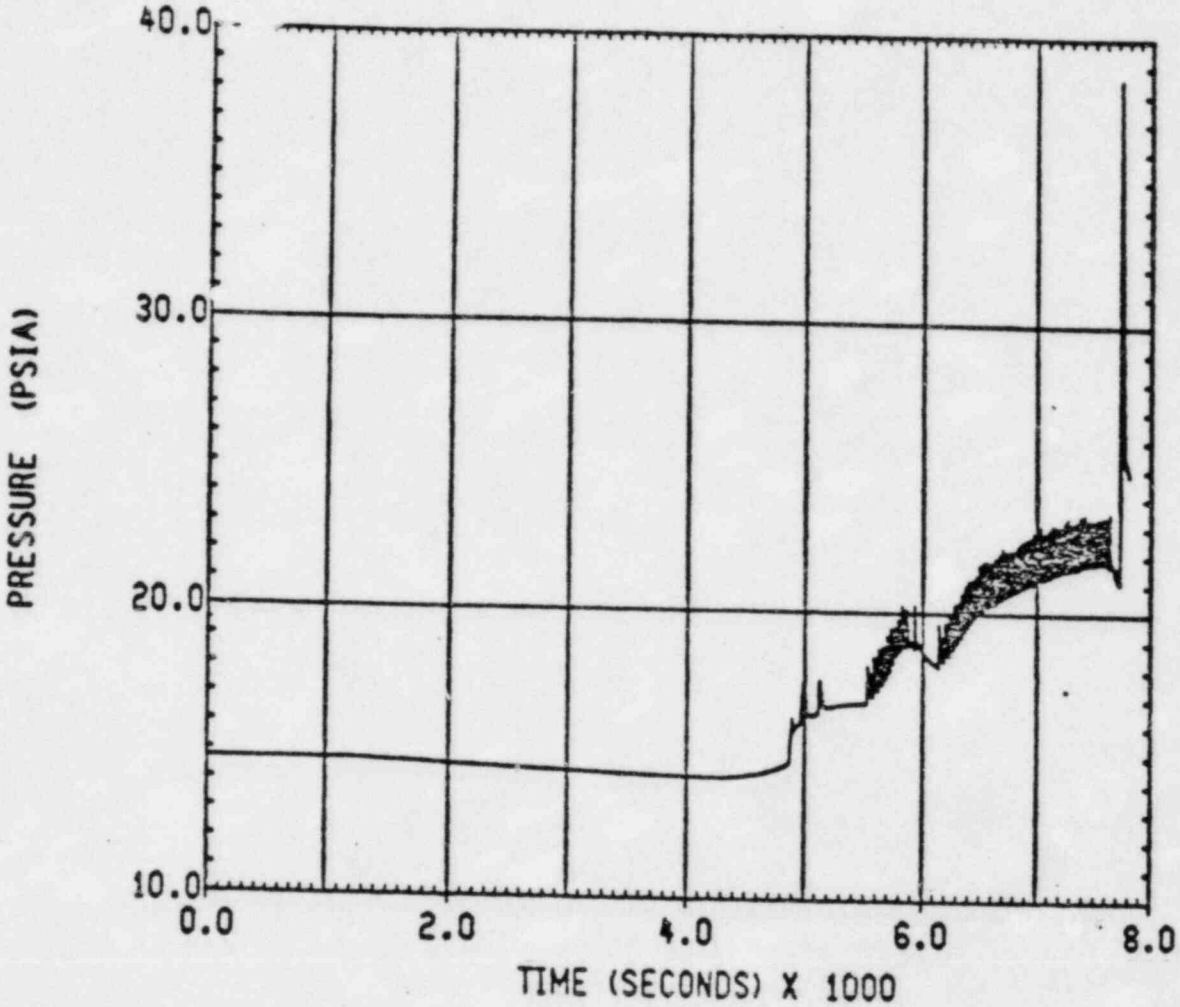
There are no NRC Staff persons who, in the course of work for the NRC, prepared a memorandum or other written document suggesting that there are one or more credible TMI-type accident scenarios for Perry or for similar BWR reactors.



GGNS BASE CASE SORV

CONTAINMENT TEMPERATURE

Figure 43



CGNS BASE CASE SORV

CONTAINMENT PRESSURE

Figure 46

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
CLEVELAND ELECTRIC ILLUMINATING COMPANY, ET AL.	)	Docket No. 50-440 OL 50-441 OL
	)	
(Perry Nuclear Power Plant Units 1 and 2	)	
	)	

AFFIDIVIT OF MARVIN W. HODGES

I, Marvin W. Hodges, being duly sworn, state as follows:

1. I am a Section Leader in the Reactor Systems Branch, Division of Systems Integration, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission.

2. I am the NRC Staff member jointly responsible with George Thomas for the answers to Interrogatories 6-1 and 6-2, of OCRE's Sixth Set of Interrogatories to the staff. I am also jointly responsible with George Thomas for the answers to Board Question 1 to 5.

3. These answers are true and complete to the best of my knowledge and belief.

Marvin W. Hodges

Subscribed and sworn to before me
this 31st day of January, 1983.

Judy L. Butts
Notary Public

My Commission expires: July 1, 1986

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
CLEVELAND ELECTRIC ILLUMINATING COMPANY, ET AL.	)	Docket No. 50-440 OL 50-441 OL
	)	
(Perry Nuclear Power Plant, Units 1 and 2)	)	

AFFIDAVIT OF JOHN K. LONG

I, John K. Long, being duly sworn, state as follows:

1. I am a Reactor Physicist in the Reactor Systems Branch, Division of Systems Integration, Office of Nuclear Reactor Regulation, U. S. Nuclear Regulatory Commission.

2. I am the NRC Staff member responsible for the answers to Interrogatories 6-3, 6-7 and 6-35 of OCRE's Sixth Set of Interrogatories to the Staff.

3. These answers are true and complete to the best of my knowledge and belief.

John K Long
John K. Long

Subscribed and sworn to before me
this 31st day of January, 1983.

Judy L. Butler
Notary Public

My Commission expires July 1, 1986

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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

AFFADAVIT OF ALLEN NOTAFRANCESCO

I, Allen Notafrancesco, being duly sworn, state as follows:

1. I am a Containment Systems Engineer in the Division of Systems Integration, Office of Nuclear Reactor Regulation, U. S. Nuclear Regulatory Commission.

2. I am the NRC Staff member responsible for the answers to Interrogatories No.: 6-6, 6-10, 6-14 thru 6-24, 6-26 thru 6-29, and 6-31 thru 6-34, of "Ohio Citizens for Responsible Energy Sixth Set of Interrogatories to the Staff".

3. These answers are true and complete to the best of my knowledge and belief.

Allen Notafrancesco

Allen Notafrancesco

Sworn and subscribed before me
this 27th day of January 1983

Raine A. Thurston

Notary Public

My commission expires: July 1, 1986

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
CLEVELAND ELECTRIC ILLUMINATING COMPANY, ET AL.	)	Docket No. 50-440 OL 50-441 OL
	)	
(Perry Nuclear Power Plant, Units 1 and 2)	)	

AFFIDAVIT OF JACQUES B. J. READ

I, Jacques B. J. Read, being duly sworn, state as follows:

1. I am a Senior Physical Scientist in the Accident Evaluation Branch, Division of Systems Integration, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission.

2. I am the NRC Staff member responsible for the answer to Interrogatory No. 6-25 of OCRE's Sixth Set of Interrogatories to the Staff.

3. This answer is true and complete to the best of my knowledge and belief.

Jacques B. J. Read

Jacques B. J. Read

Subscribed and sworn to before me
this [illegible] day of January, 1983.

Notary Public

My Commission expires: [illegible]

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the matter of	)	
	)	
CLEVELAND ELECTRIC ILLUMINATING COMPANY, ET AL.	)	Docket No. 50-440 OL 50-441 OL
	)	
(Perry Nuclear Power Plant, Units 1 and 2)	)	

AFFIDAVIT OF JOHN J. STEFANO

I, John J. Stefano, being duly sworn, state as follows:

1. I am a Project Manager in Licensing Branch No. 1, Division of Licensing, Office of Nuclear Reactor Regulation, U. S. Nuclear Regulatory Commission.

2. I am the NRC Staff member responsible for the answers to Interrogatories No. 6-9 and 6-37 of OCRE's Sixth Set of Interrogatories to the Staff.

3. These answers are true and complete to the best of my knowledge and belief.

John J. Stefano

Subscribed and sworn to before me
this _____ day of January, 1983.

_____
Notary Public

My Commission expires: _____

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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

AFFIDAVIT OF CHARLES G. TINKLER

I, Charles G. Tinkler, being duly sworn, state as follows:

1. I am a Containment Systems Engineer in the Division of Systems Integration, Office of Nuclear Reactor Regulation, U. S. Nuclear Regulatory Commission.

2. I am the NRC Staff member responsible for the answer to Interrogatory No. 6-11 of "Ohio Citizens for Responsible Energy Sixth Set of Interrogatories to the Staff."

3. This answer is true and complete to the best of my knowledge and belief.

Charles G. Tinkler

Charles G. Tinkler

Sworn and subscribed before me
this 27th day of January 1983

Claire A. Harwood

Notary Public

My commission expires: July 1, 1986

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 No. 50-440 OL 50-441 OL

AFFIDAVIT OF GEORGE THOMAS

I, George Thomas, being duly sworn, state as follows:

1. I am a Nuclear Engineer in the Reactor Systems Branch, Division of Systems Integration, Office of Nuclear Reactor Regulation, U. S. Nuclear Regulatory Commission.

2. I am the NRC Staff member jointly responsible with Marvin W. Hodges for the answers to Interrogatories 6-1 and 6-2 of OCRE's Sixth Set of Interrogatories to the Staff. I am also jointly responsible with Marvin W. Hodges for the answers of the Board Questions 1 to 5.

3. These answers are true and complete to the best of my knowledge and belief.

George Thomas
George Thomas

Subscribed and sworn to before me
this ____ day of January, 1983.

_____
Notary Public

My Commission expires: _____

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the matter of	)	
	)	
CLEVELAND ELECTRIC ILLUMINATING COMPANY, ET AL.	)	Docket No. 50-440 OL 50-441 OL
	)	
(Perry Nuclear Power Plant, Units 1 and 2)	)	

AFFIDAVIT OF LI YANG

I, Li Yang, being duly sworn, state as follows:

1. I am a Structural Engineer in the Structural Engineering Branch, Division of Engineering, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission.

2. I am the NRC Staff member responsible for the answers to Interrogatories No. 6-5, 6-12, 6-13 and 6-30 of OCRE's Sixth Set of Interrogatories to the Staff.

3. These answers are true and complete to the best of my knowledge and belief.

Li Yang

Subscribed and sworn to before me
this 27th day of January, 1983.

Notary Public

My Commission expires: July 1, 1986

Marvin W. (Wayne) Hodges

Professional Qualifications

Reactor Systems Branch

Division of Systems Integration

U. S. Nuclear Regulatory Commission

I am employed as a Section Leader in Section B of the Reactor Systems Branch, DSI.

I graduated from Auburn University with a Mechanical Engineering Degree in 1965. I received a Master of Science degree in Mechanical Engineering from Auburn University in 1967.

In my present work assignment at the NRC, I supervise the work of 5 graduate engineers; my section is responsible for the review of primary and safety systems for BWRs. I have served as principal reviewer in the area of boiling water reactor systems. I have also participated in the review of aralytical models use in the licensing evaluations of boiling water reactors and I have the technical review responsibility for many of the modifications and analyses being implemented on boiling water reactors post the Three Mile Island, Unit-2 accident.

As a member of the Bulletin and Orders Task Force which was formed after the TMI-2 accident, I was responsible for the review of the capability of BWR systems to cope with loss of feedwater transient and small break loss-of-coolant accidents.

I have also served at the NRC as a reviewer in the Analysis Branch of the NRC in the area of thermal-hydrulic performance of the reactor core. I served as a consultant to the RES representative to the program management group for the BWR Blowdown/Emergency Core Cooling Program.

Prior to joining the NRC staff in March, 1974, I was employed by E. I. DuPont at the Savannah River Laboratory as a research engineer. At SRL, I conducted hydraulic and heat transfer testing to support operation of the reactors at the Savannah River Plant. I also performed safety limit calculations and participated in the development of analytical models for use in transient analyses at Savannah River. My tenure at SRL was from June 1967 to March 1974.

From September 1965 to June 1967, while in graduate school, I taught courses in thermodynamics, statics, mechanical engineering measurements, computer programming and assisted in a course in the history of engineering. During the summer of 1966, I worked at the Savannah River Laboratory doing hydraulic testing.

PROFESSIONAL QUALIFICATIONS

OF

JOHN K. LONG

Reactor Physicist, USNRC

(April 30, 1982)

My name is John K. Long. I am a reactor physicist in the Reactor Systems Branch of the Nuclear Regulatory Commission, business address Washington, D. C. 20555.

I graduated from Columbia University with a bachelor's degree in Chemical Engineering, 1942. I was employed by the Hercules Powder Company, 1942-1945, in the manufacture of explosives. I worked at Wright-Patterson Air Force Base, Dayton, Ohio, 1945-1949, in the development of aircraft materials. I received a Ph. D. in Nuclear Physics from Ohio State University in 1953, and was employed by Argonne National Laboratory from 1955 to 1974. At Argonne I participated in and directed research on reactor critical facilities, analysis of reactor operating phenomena at EBR-2, and development of procedures for criticality control.

I have been employed by the Nuclear Regulatory Commission since 1974, and have reviewed many problems related to the licensing of sodium cooled fast breeder reactors, including problems related to the accidental generation of hydrogen and its release in containment. I was responsible for the review of aerosol and containment analyses and dose consequence analyses in the FFTF review. Since the TMI-2 accident I have participated in accident analyses involving hydrogen problems for the TMI-1, Zion, Indian Point, Sequoyah and McGuire reactors.

With the reactivation of the CRBR licensing activities, I have been given responsibility for review of those aspects of core disruptive accidents involving thermal margins beyond the design basis.

A list of recent publications is attached.

List of Publications

A. M. Broomfield, A. L. Hess, P. I. Amundson, Q. L. Baird, E. F. Bennett, W. G. Davey, J. M. Gasidlo, W. P. Keeney, J. K. Long and R. L. McVean, ZPR-III Assemblies 48, 48A, and 48B: The Study of a Dilute Plutonium-fueled Assembly and its Variants, ANL-7759 (Dec. 1970).

J. K. Long - The Effect of a Metallurgical Phase Change on the Power Coefficient of EBR-II. Nucl. Appl. 10 17 (January, 1971).

J. K. Long and E. M. Dean, EBR-II Codes for Processing Reactivity Data at Full and Reduced Coolant Flows. ANL/EBR-020 (May, 1970).

J. K. Long and D. Meneghetti, Anisotropic Scattering Calculations for a Stainless Steel Reflector. Trans. ANS 15, 2, 796 (November, 1972).

J. K. Long - Shortcomings of the Albedo Approximation in KENO Calculations. Trans. ANS 17, 269 (November, 1973)

Safety Evaluation Report, Fast Flux Test Facility, NUREG-0358, August 1978, (J. K. Long responsible for accident consequence analysis)

Final Environmental statement. Clinch River Breeder Reactor Plant, NUREG-0139, February 1977 (J. K. Long responsible for chapters 8, 9, and parts of 11)

Site Suitability Report, Clinch River Breeder Reactor Plant, March 1977, (J. Long responsible for section on containment design)

J. K. Long, A. R. Marchese, T. P. Speis, R. D. Gasser, W. T. Pratt, Radiological and Containment Analysis for a Postulated Fast Reactor Melt Through Accident With Containment Venting, Proceedings of the meeting of the European Nuclear Society on Fast Reactor Safety Technology, Seattle, Washington, Aug. 19-23, 1979

Preliminary Assessment of Core Melt Accidents at the Zion and Indian Point Nuclear Power Plants and Strategies for Mitigating Their Effects. NUREG-0850, vol. 1, U.S. Nuclear Regulatory Commission November 1981. (J. Long responsible for sections on hydrogen).

J. K. Long Analysis of the Static Power Coefficient of EBR-II with Reduced Coolant Flow, Nucl. Appl. 6, 116 (1969)

J. K. Long, R. W. Hyndman Prompt Feedback Reactivity in EBR-II, Trans. ANS.- 12, 2, 690 (December 1969)

J. K. Long, The Contribution of Fuel Expansion to the EBR-II Power Coefficient, Trans. ANS, 12 (Supplement) p. 48. (San Juan, Puerto Rico, October 1969)

W. G. Davey, A. Broomfield, P. I. Amundson, Q. L. Baird, J. M. Gasidlo, W. P. Keeney, J. K. Long, R. G. Matlock, R. L. McVean, L. A. Mountford, R. J. Palmer, R. O. Vosburgh. A Benchmark Series of Plutonium-Fueled Fast Critical Assemblies. Trans. ANS. 11, 1, 239 (June, 1968)

J. K. Long, F. S. Kirn, Effect of Reduced Flow Conditions on the Power Coefficient of EBR-II. Trans. ANS, 11, 1, 280 (June, 1968)

P. J. Persiani, J. Kallfelz, F. S. Kirn, J. K. Long, R. Smith, EBR-II Power Coefficient Variations During Operational Run 26. Trans. ANS, 11, 1, 281 June, 1968)

R. R. Smith, R. A. Cushman, R. W. Hyndman, F. S. Kirn, J. K. Long, W. B. Loewenstein, J. T. Madell, T. R. Bump, and P. J. Persiani, Recent Operating Experience with EBR-II. Brit. Nucl. Energy Soc. Conf. on the Physics of Fast Reactor Operation and Design June (1969)

R. R. Smith, T. R. Bump, R. A. Cushman, R. W. Hyndman, F. S. Kirn, W. B. Loewenstein, J. K. Long, J. T. Madell, P. J. Persiani, W. R. Wallin, The Effects of an Over-Cooled Stainless Steel Reflector on the EBR-II Power Coefficient. ANL-7544, (1969)

P. J. Persiani, T. R. Bump, R. A. Cushman, J. K. Long, D. Kucera, Analysis of the Variation in the EBR-II Power Coefficient Due to Blanket Change, Trans. ANS., 11, 1, 282, (June, 1968)

F. S. Kirn, J. K. Long, R. R. Smith, Effect of the Stainless-Steel Reflector on the Power Coefficient of EBR-II, Trans. ANS 10, 2, 648, (November, 1967)

J. K. Long, A. L. Hess, R. L. McVean, P. J. Persiani, A. J. Ulrich, Q. L. Baird, R. A. Bennett, S. L. Engstrom, FFTF Critical Experiments -- Control Rod Studies on ZPR-3. Trans. ANS., 10, 1, 269 (June, 1967).

A. M. Broomfield, P. I. Amundson, W. G. Davey, J. M. Gasidlo, A. L. Hess, W. P. Keeney and J. K. Long "ZPR-3 Assembly 48: Studies of a Dilute Plutonium Fueled Assembly". International Conference on Fast Critical Experiments and Their Analysis, ANL-7320 p. 205-214 (1966)

P. J. Persiani, T. R. Bump, R. A. Cuchman, J. Long, and D. Kucera, Analysis of the Variation in the EBR-II Power Coefficient Due to Blanket Change, in "Reactor Physics Division Annual Report; July 1, 1967 to June 30, 1968," p. 203, ANL-7410.

P. J. Persiani, J. Hallfelz, F. Kirn, J. K. Long, and R. Smith, EBR-II Power Coefficient Variations During Operational Run 26, in "Reactor Physics Division Annual Report; July 1, 1967 to June 30, 1968," p. 201, ANL-7410.

P. J. Persiani, J. K. Long, A. J. Ulrich, and D. A. Kucera, Development of the Critical Experiment Program for the Fast Fuels Test Facility in "Reactor Physics Division Annual Report; July 1, 1966 to June 30, 1967," p. 232, ANL-7210.

R. Avery, C. Dickerman, W. Y. Kato, J. K. Long, A. B. Smith, in "Fast Breeder Reactors," P. V. Evans, ed., p. 403-420. Oxford, Pergamon Press, 1967 from British Nuclear Energy Society Conference on Fast Breeder Reactors, London.

J. M. Gasidlo, J. K. Long, R. L. McVean, Critical Studies of a Dilute Fast Reactor Core (ZPR-III), Assembly 31) ANL-6338 (October, 1961).

A. L. Hess, W. Gemmell, J. K. Long, R. L. McVean, Critical Studies of a 490-liter Uranium Oxide Fast Reactor Core (ZPR-III), Assembly 29) ANL-6336, (Nov. 1960).

J. M. Gasidlo, J. K. Long, R. L. McVean, Critical Studies of a Fast Reactor Core Containing Depleted Uranium and Sodium as Diluents (ZPR-III, Assembly 36) ANL-6494 (January, 1962).

P. I. Amundson, W. Gemmell, J. K. Long, R. L. McVean, Critical Studies of Uranium-Steel and Uranium-Steel-Sodium Fast Reactor Core. (ZPR-III, Assemblies 32 and 33) ANL-6690 (February, 1963).

J. M. Gasidlo, J. K. Long, W. P. Keeney, Doppler Effect Measurements, Sodium Void Coefficients, and Critical Studies of a 5000-liter Fast Power Breeder Reactor by the Two-Zone Method. ZPR-III, Assembly 43. ANL-6838 (May, 1964).

J. K. Long, R. L. McVean, A. B. Reynolds, S. L. Stewart, A. Weitzberg, A Loridon, Critical Mass of SEFOR Mock-Up in ZPR-III. Nucl. Sci. Eng. 25, 442-4, 1966).

F. W. Thalgott, J. K. Long, W. G. Davey, W. Y. Kato, S. G. Carpenter, H. A. Karrowitz, G. H. Best, Fast Critical Experiments and Their Analysis, A/CONF. 18/P/265 (3rd Int. Conf. On Peaceful Uses of Atomic Energy, 1964).

R. O. Brittan, B. Cerutti, H. V. Lichtenberger, J. K. Long, R. L. McVean, F. W. Thalgott, M. Novick, R. Rice, Hazard Evaluation Report on the Fast Reactor Zero Power Experiment (ZPR-III) ANL-6408 (October, 1961).

J. K. Long, A. R. Baker, W. Gemmell, W. P. Keeney, R. L. McVean, F. W. Thalgott in "Physics of Fast and Intermediate Reactors, Vol. 1" p. 261-85. Vienna International Atomic Energy Agency (1962).

PROFESSIONAL QUALIFICATIONS OF
ALLEN NOTAFRANCESCO
CONTAINMENT SYSTEMS BRANCH
DIVISION OF SYSTEMS INTEGRATION
U. S. NUCLEAR REGULATORY COMMISSION

I am a Containment Systems Engineer in the Containment Systems Branch, Division of Systems Integration, Office of Nuclear Reactor Regulation, which I have held since June 1981.

I serve as a reviewer in the area of containment systems (BWRs). This involves performing review and evaluations of specific portions of the licensing applications for which the branch has responsibility to assure public health and safety.

I received a Bachelor of Science degree in Nuclear Engineering from the Polytechnic Institute of New York in 1976, and I received a Master of Science degree in Mechanical Engineering from the Polytechnic Institute of New York in 1980. Other training includes: BWR/6 Technology Course - 1982 (NRC sponsored) and BWR/6 Simulator Course - 1982 (NRC sponsored).

From 1976 to 1979, I was employed as a thermal hydraulic analyst by Ebasco Services, Inc., New York. My responsibilities included performing safety related containment analyses pertinent to the licensing of various nuclear power plants, preparing responses to NRC questions and writing appropriate sections of safety analysis report.

STATEMENT OF PROFESSIONAL QUALIFICATIONS

JACQUES B. J. READ

Accident Evaluation Branch
Division of Systems Integration
U. S. Nuclear Regulatory Commission

As a member of the Accident Evaluation Branch my duties include the performance of technical reviews, analyses, and evaluation of fission product behavior and of chemical phenomena involved in the safety of nuclear reactors. Prior to the creation of the Accident Evaluation Branch in 1980, I was a member of the Accident Analysis Branch. Within that branch my duties included the identification and evaluation of hazards to the safe operation of nuclear power plants due to accidents external to those plants, and aspects of other risk evaluations susceptable to stochastic methods. Risks from such external hazards for which I have performed or participated in analyses include munitions rail traffic near Braidwood, Illinois, tanker traffic near Waterford, Mississippi and Salem, New Jersey, and military aviation near Seabrook, Massachusetts, Boardman, Oregon, Douglas Point, Maryland and Palo Verde, Arizona. I was responsible for assessing the risks to proposed nuclear power plants from explosives, flammable gases, aircraft, and other missile impacts. I have represented the Nuclear Regulatory Commission in discussions of flammable gas hazards amongst member nations of the Organization of Economic Cooperation and Development.

I was born in Maywood, New Jersey, in 1935, and received an A.B. from Princeton in 1957 (physical chemistry), an M.S. from Yale in 1958 (statistical mechanics), and a Ph.D. from Yale in 1962 (chemistry and physics). I was employed at Oak Ridge National Laboratory during the summers of 1956 and 1957, and held post-doctoral appointments at Columbia University and the Nevis Synchrocyclotron Laboratory between 1961 and early-1964. I taught several courses in chemistry at Fairleigh Dickinson University, part-time during 1962 and 1963 and full-time during 1964. From late-1964 to 1974, I was employed by the Lawrence Livermore Laboratory, in the Radiochemistry Division prior to 1971 and in Special Projects Division thereafter. From 1966 to 1974 I held an appointment as Lecturer in the Department of Applied Science, Graduate School of Engineering, University of California. During 1973 and 1974 I was on detached assignment to the U. S. Atomic Energy Commission headquarters, under a contract between the Commission and the Regents. I resigned from the Laboratory and the Department on November 4, 1974, to assume my present position.

My baccalaureate thesis was a study of high temperature electrochemistry. At Yale, I studied optical rotation of polarized light by molecules. My eventual doctoral thesis was a study of the mechanisms of the nuclear reactions of heavy ions, and my post-doctoral studies concerned proton-induced nuclear spallation reactions, and the creation of computer programs to calculate the probabilities of rare nuclear interactions. While at the University of California's Lawrence Livermore Laboratory, I studied deuteron-induced nuclear reactions, and was involved in research in nuclear fission and fusion devices. My duties included supervision of radiochemical analysis and responsibility for the radiochemical diagnostics of certain prototype weapons. I wrote the Monte Carlo code used to reduce the data from the Gnome "neutron wheel" experiment, and performed the search for neutron-rich silicon isotopes on the Hutch Event. I was, for several years, a participant in the U. S. - U. K. Joint Working Group in Radiochemistry.

I am a member of the American Chemical Society and Sigma Xi. I have served on the Board of Abstractors, in French and English, of the American Chemical Society, and have in the past held memberships in the American Physical Society and the American Association of University Professors. I have authored or co-authored articles in Physical Review and Journal of Inorganic and Nuclear Chemistry, papers presented before the American Nuclear Society, the American Chemical Society, and the International Union of Pure and Applied Chemistry, and numerous technical reports.

PROFESSIONAL QUALIFICATIONS

JOHN J. STEFANO

Licensing Branch No. 1
Division of Licensing
U. S. Nuclear Regulatory Commission

My name is John J. Stefano. I have been employed by the U. S. Nuclear Regulatory Commission (NRC) since December 1981. In February 1982, I was assigned my current duties and responsibilities as project manager for the Perry Nuclear Power Plant (PNPP), which includes the management and coordination of environmental and safety reviews documented in the Cleveland Electric Illuminating Company's (the applicant) ER and FSAR for an operating license, ensuring that all work performed by the applicant complies with all applicable NRC rules, regulations, guidelines, schedules and the provisions of AEA and NEPA. In this capacity I serve as the principle NRC point of contact and liaison between the project review staff, the applicant and other interested parties (the public, Congress, other federal, State and local governmental agencies, the media, the Advisory Committee on Reactor Safeguards, and NRC senior management).

My accomplishments to date on the PNPP project in this capacity have included: the preparation and issuance of the PNPP Safety Evaluation Report and Supplements 1 and 2 thereto (NUREG-0887); the Draft and Final Environmental Statements for the PNPP (NUREG-0884); the coordination and preparation of responses to interrogatories received in the PNPP licensing proceeding from February 1982 to the present.

I have had over 27 years of technical engineering and management experience on a wide-range of nuclear and non-nuclear programs, since having received my Bachelor of Science Degree in Aeronautical Engineering from the University of St. Louis in 1956. Post-graduate studies have included nuclear engineering and reactor safety; industrial engineering; business administration/accounting; quality assurance and mechanical engineering attending the University of Minnesota, New York University and the Carnegie-Mellon University, over the period of 1958-1975 in the pursuit of these studies.

A summary of previous positions held follows:

1977-1981	Various engineering positions with the U. S. Department of Energy involving the development and demonstration of fuel cell technology and other alternative energy technology programs; represented the Secretary of Energy on a number of national government/industry committees on energy development and economic assessment. Authored and co-authored several papers on work managed in technical journals and publications.
1965-1977	Various engineering positions with the U. S. Atomic Energy Commission involving the design, development and construction of nuclear-fueled terrestrial and space power sources and the Liquid Metal Fast Breeder reactor. Served as technical representative for AEC directors at the sites where this work was performed.

1958-1965	The position of senior project manager for the design, development, construction and test of weapon system flight simulators for the U. S. Army, Marines and Navy.
1957-1958	Active military duty with the U. S. Coast Guard
1956-1957	The position of Aeronautical engineer with the Grumman Aerospace Corporation involving the design, test and reliability analysis of flight control systems for supersonic aircraft.

PROFESSIONAL QUALIFICATIONS

Charles G. Tinkler, Jr.

I am a Containment Systems Engineer in the Containment Systems Branch of the Office of Nuclear Reactor Regulation, U. S. Nuclear Regulatory Commission. In this position, which I have held since May 1976, I am responsible for the review and technical evaluation of containment related aspects for PWR applications for both construction permits and operating licenses. Among the plants for which I have or have had this responsibility are Yellow Creek Nuclear Plant; San Onofre Nuclear Generating Station, Units 2 and 3; Sundesert Nuclear Plant; and Bellefonte Nuclear Plant. From June 1979 to March 1980 I was assigned to the Systematic Evaluation Program Branch where I was responsible for the review of reactor and containment systems related aspects for the Yankee Rowe, R. E. Ginna, and Palisades nuclear plants. My most recent assignment is the analysis of hydrogen control in LWR plants.

From September 1973 to May 1976, I was employed as an engineer in the Nuclear Energy Systems Division of the Westinghouse Electric Corporation, Pittsburgh, Pennsylvania. My responsibilities included performing safety related analysis pertinent to the licensing of PWR plants, preparing responses to NRC questions and writing appropriate sections of safety analysis reports. I was principally involved in work related to the licensing of ice condenser containment plants where I also participated in the ice condenser test program.

From July to September 1973 I was employed by Westinghouse in the Heat Transfer Division where I was responsible for design analysis of waste heat recovery systems.

I received a Bachelor of Science in Mechanical Engineering from Virginia Polytechnic Institute in 1973.

PROFESSIONAL QUALIFICATIONS OF
GEORGE THOMAS
REACTOR SYSTEMS BRANCH
DIVISION OF SYSTEMS INTEGRATION
U.S. NUCLEAR REGULATORY COMMISSION

I have been employed as a Nuclear Engineer in the Reactor Systems Branch, Division of Systems Integration, Office of Nuclear Reactor Regulation, since October 1980.

I serve as a reviewer in the area of reactor systems (BWRs). This involves performing reviews and evaluations of those portions of the applications for Operating Licenses and submittals regarding proposed modifications in licensed nuclear power plants for which the branch has responsibility to assure public health and safety.

Since 1981, I participated in the Perry review for the Reactor Systems Branch.

I received a Bachelor of Science degree in physics from Kerala (India) University in 1963. Additional graduate and professional courses were taken in Nuclear Engineering, University of Pennsylvania and Engineers Club, Philadelphia, PA - 1975. Other educational background and training includes: Power Plant Engineering - 1976 (diploma from International Correspondence Schools, Scranton, PA); PWR Technology Course - 1980 (NRC sponsored); BWR/6 Simulator Course - 1981 (NRC sponsored); Tarapur Atomic Power Station (India) - Reactor Operators Training Program - 1969; GE BWR - Training at Tarapur by GE - 1967.

From 1967 to 1972 I served as a Reactor Operator on the Indian Atomic Energy Commission's first commercial nuclear power station, Tarapur 1 & 2 (a BWR built by Bechtel and GE). There I participated in construction tests, pre-operational tests, and normal operations of the station.

From 1973 to 1975 I was employed by United Engineers and Constructors (UE&C), Philadelphia, PA. Initially I was a Test and Start-up Engineer in the Construction Division of UE&C. In this capacity I wrote various procedures and systems descriptions for a BWR. Subsequently, I worked as a Staff Nuclear Engineer on the Nuclear Technical Staff of UE&C. I was engaged in providing technical expertise and consultation services to all nuclear projects of UE&C.

From 1975 to 1980 I was a Systems Engineer in the Power Division of Stone & Webster Engineering Corporation. I performed detailed engineering and design of reactor systems of a BWR. My duties included project interface and coordination work with the NSSS supplier (GE) and the client (utility).

LI YANG

PROFESSIONAL QUALIFICATIONS

STRUCTURAL ENGINEERING BRANCH

I am a Structural Engineer in the Structural Engineering Branch of the Nuclear Regulatory Commission. I am responsible for the review and evaluation of the adequacy of criteria used in the structural design and analysis of seismic Category I structures, systems and components of the Perry Plant.

I received a B.S. degree in Civil Engineering from Cheng Kung University, Taiwan in 1956. I received a degree of Master of Applied Science in Structural Engineering from the University of British Columbia, British Columbia, Canada in 1964.

From 1958 to 1962 I worked for Taiwan Power Company as a Civil Engineer on concrete quality control, and design of structural elements for hydro-electric power projects. From May 1965 to October 1970, I worked as a structural engineer for Swan Wooster Engineering Company on structural design and analysis of industrial and commercial buildings. From November 1970 to December 1972, I worked as a structural engineer for Donald Warren Engineers, Los Angeles, California. Since August 1973, I worked for Bechtel Power Corporation, Los Angeles, California until August 1980. During the years (1973-1980) of my association with the Bechtel Power Corporation as a structural engineer, most of my assignments were related to the structural design and analysis of Category I structures for various nuclear power plants.

In September 1980 I accepted a position as a structural engineer in the Structural Engineering Branch, Division of Engineering, Nuclear Regulatory Commission. My responsibilities included the review and technical evaluation of the safety aspects of Category I structures of nuclear power plants.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
CLEVELAND ELECTRIC ILLUMINATING COMPANY, ET AL.	)	Docket No. 50-440 OL 50-441 OL
	)	
(Perry Nuclear Power Plant, Units 1 and 2)	)	

CERTIFICATE OF SERVICE

I hereby certify that copies of "NRC STAFF ANSWERS TO OCRE SIXTH SET OF INTERROGATORIES AND BOARD QUESTIONS TO NRC STAFF" in the above-captioned proceeding have been served on the following by deposit in the United States mail, first class, or, as indicated by an asterisk, by deposit in the Nuclear Regulatory Commission's internal mail system, this 3rd day of February, 1983:

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