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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 50-441
	)	
(Perry Nuclear Power Plant, Units 1 and 2)	)	

APPLICANTS' DIRECT TESTIMONY OF EILEEN M. BUZZELLI, JOHN D. RICHARDSON, KEVIN W. HOLTZCLAW, ROGER W. ALLEY, BERNARD LEWIS, BELA KARLOVITZ AND G. MARTIN FULS ON THE PRELIMINARY EVALUATION OF THE PERRY NUCLEAR POWER PLANT HYDROGEN CONTROL SYSTEM (ISSUE #8)

I. INTRODUCTION AND OVERVIEW

Q1: Please state your name and describe your present position.

A1: (Buzzelli) My name is Eileen M. Buzzelli. I am employed as a senior licensing engineer with The Cleveland Electric Illuminating Co. (CEI). My responsibilities include managing the resolution of licensing issues relating to the Perry Nuclear Power Plant (PNPP). In this capacity, I am responsible for all licensing issues relating to the PNPP Hydrogen Control System (HCS), including preparation of the PNPP Preliminary Evaluation Report "The Cleveland Electric Illuminating Company Preliminary Evaluation of the Perry Nuclear Power Plant Hydrogen Control System". (Applicants' Exhibit 8-1)

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(Richardson) My name is John D. Richardson. I am currently employed by Enercon Services, Inc. as Vice President in charge of Atlanta-based operations. Enercon provides technical consulting services primarily to the nuclear industry. In my present capacity, I serve as a technical consultant to CEI on a number of licensing issues, including those involving the PNPP HCS.

(Holtzclaw) My name is Kevin W. Holtzclaw. I am a principal licensing engineer in the General Electric Safety and Licensing Operation of the Nuclear Energy Business Operation.

(Alley) My name is Roger W. Alley. I am employed by Gilbert/Commonwealth, Inc. (G/C), the Architect-Engineer for the Perry Nuclear Power Plant (PNPP). I am currently the Manager of the G/C Structural Engineering Nuclear Section, which has the overall responsibility for the PNPP structural design. I supervise approximately 25 G/C structural engineers. In addition, I am the G/C Project Structural Engineer for PNPP, which means that I have the management responsibility at G/C for all structural work at PNPP.

(Lewis) My name is Dr. Bernard Lewis. I am currently the President of Combustion and Explosives Research, Inc. (Combex). Combex, founded in 1953, provides consulting services to government, research institutes and industry on the fundamentals of combustion, flames, ignition and explosions of gases, liquids and solids.

(Karlovitz) My name is Bela Karlovitz. I am Secretary-Treasurer of Combex, and was a co-founder of the company. A significant amount of my consulting activity is in the field of turbulent flames.

(Fuls) My name is G. Martin Fuls. I am currently self-employed and am President of GMF Associates. I am performing consulting work primarily related to hydrogen combustion analysis for nuclear power plants.

Q2: What is your educational and professional background?

A2: (Buzzelli) A copy of my qualifications is attached. I hold a Bachelors Degree in mechanical engineering from Ohio University (1976) and a Masters Degree in Business Administration from Case Western Reserve University (1984). I joined CEI as a nuclear engineer at PNPP in 1976, and have held a variety of positions at the plant since that time. My responsibilities included work in design and construction engineering prior to transferring to licensing work in 1980. In 1984, I became the Senior Licensing Engineer in charge of all PNPP licensing issues.

(Richardson) A copy of my qualifications is attached. I have worked in the nuclear engineering field since 1974, when I joined Westinghouse Electric Corporation. I worked for Westinghouse from 1974 until 1976. My responsibilities included work as an engineer and shift supervisor at the dual reactor naval prototype (A1W) in Idaho.

I was employed by Mississippi Power & Light Co. (MP&L) from 1976 to 1983 in a variety of positions in connection with the design, licensing and operation of MP&L's BWR 6/Mark III Grand Gulf Nuclear Station (GGNS). Among other positions, I served as Manager of Nuclear Safety & Licensing. In that capacity, I handled all licensing and technical issues relating to the GGNS Hydrogen Ignition System (HIS). While with MP&L, I served for four years as Chairman of the Mark III Hydrogen Control Owners Group (HCOG), which includes MP&L, CEI, Gulf States Utilities and Illinois Power Corp. HCOG was formed in 1981 to address generic hydrogen control NRC licensing issues common to BWR 6/Mark III nuclear plants.

Since joining Enercon in 1984, among other activities, I have worked with CEI on its preliminary evaluation of the PNPP HCS. I participated in the preparation of the PNPP Preliminary Evaluation Report.

(Holtzclaw) A listing of my educational and professional qualifications is attached. I have spent nearly 17 years as an engineer in the nuclear power industry, and have worked for General Electric (GE) since 1969. Since 1980, I have been responsible for defining and planning GE programs related to NRC degraded core rulemaking. Since March, 1982 I have served as the GE Program Manager of the GE Severe Accident Program. This has entailed managing GE programs including the BWR/6 standard Plant Probabilistic Risk Assessment and associated severe accident submittals, evaluations of hydrogen event risks and assessment of BWR fission product retention capability. I have been the GE representative on the Industry Degraded Core Rule-making (IDCOR) Technical Advisory Group.

(Alley) A copy of my qualifications is attached. I am a registered professional engineer with over 16 years experience in structural design of nuclear power plants (principally PNPP and the Davis-Besse Nuclear Power Station).

I worked for Bechtel Corporation from 1969-1972 on the design of the Davis-Besse plant. During that time, I performed design of reinforced concrete and steel structures, seismic analysis of the reactor building complex, and reviews of calculations for the ASME containment vessel and components.

Since 1972, I have worked for G/C in connection with its design of PNPP. I was involved in developing the original ASME containment design specification for PNPP. I have been responsible for both finite element and stress analyses of PNPP's shell-type structures, including the containment, drywell and shield building.

(Lewis) A copy of my qualifications is attached. I hold a Bachelor of Science degree in chemical engineering from Massachusetts Institute of Technology (1923), a Masters degree in physical chemistry from Harvard University (1924), a Ph.D. in physical chemistry from Cambridge University, England (1926), and an ScD Honorary degree from Cambridge University (1953).

Prior to starting Combex, from 1929 to 1953 I worked in the U.S. Bureau of Mines, and served as Chief of the Explosives and Physical Sciences Division of the Bureau. In that capacity, I directed all combustion, flame, and explosives research activities of the Bureau.

In 1951 and 1952, I also served as Director of Research on propellants and explosives for the U.S. Army Ordinance Corps.. I invented the present standard U.S. Army Hand Grenade Fuze during the Second World War.

As indicated in my resume, I have served on numerous committees and organizations in the area of combustion. I founded, and am currently Honorary President of, The Combustion Institute (International Scientific Society). I have authored over 250 numerous publications, chiefly on the subjects of combustion, flames, explosions and explosives, including two books on "Combustion, Flames and Explosions of Gases," which are used throughout the world.

I began specific research into hydrogen combustion in 1930. I have studied the subject intensively since that time, and have authored a number of publications on the detonation, combustion and ignition of hydrogen.

(Karlovitz) A copy of my qualifications is attached. I hold mechancial (1926) and electrical (1928) engineering degrees from universities in Budapest, Hungary and Zurich, Switzerland, respectively.

Among my professional activities prior to starting Combex, I conducted experimental research at Westinghouse on magnetohydrodynamic (MHD) power generation from flames from

1938 to 1947, and was the Chief of the Flame Research Section of the U.S. Bureau of Mines from 1947 to 1953. Among my professional accomplishments, I originated the concept of MHD power generation. MHD involves the conversion of flame energy directly into electrical energy, without the application of a steam plant. I also originated and developed the electrically augmented flame, and designed and patented a clean burning spark ignition engine based on fundamental turbulent flame theory.

I have authored numerous publications, particularly in the area of turbulent flames.

(Fuls) A copy of my qualifications is attached. I hold a Bachelor of Science degree in mechanical engineering from Carnegie Mellon University (1956), a Master of Science degree from Carnegie Mellon (1957), and a Ph.D. in mechanical engineering from the University of Pittsburgh (1958).

I have over 20 years experience in the development of analytical techniques for analyses of nuclear plant systems. Much of my experience has been in the area of computer program analysis. I first became involved with digital computers by programming an IBM-650 for my Masters project to perform dynamic analysis of radio chasses. After joining Bettis Atomic Power Laboratory in 1960, I first developed computer programs for the solution of problems in the navy nuclear propulsion program. This work led to the development of three computer programs which were part of the approved design basis for new navy nuclear plants.

I worked for Bettis until 1976, when I joined Offshore Power Systems (Westinghouse). I continued developing and modifying computer programs relating to nuclear power plants. After the TMI-2 accident, I began developing analytical methods to evaluate the consequences of a hydrogen burn in a reactor plant containment. This led to my development of the CLASIX and CLASIX-3 programs. These programs have been used extensively by U.S. and foreign nuclear utilities to address hydrogen burning issues raised after TMI.

Q3: What is the overall purpose of Applicants' direct testimony on Issue #8?

A3: (Buzzelli) The overall purpose of Applicants' direct testimony is to address the following contention of Ohio Citizens for Responsible Energy (OCRE):

> The Perry hydrogen control system is inadequate to assure that large amounts of hydrogen can be safely accommodated without a rupture of the containment and a release of substantial quantities of radioactivity to the environment.

OCRE's contention, identified as Issue #8 in this proceeding, is set forth in the Licensing Board's Memorandum and Order (Motions on Hydrogen Control Contentions), dated March 14, 1985.

Section I of this testimony provides a general overview of CEI's selection and preliminary evaluation of the PNPP distributed igniter system for controlling large amounts of hydrogen. Section I also discusses the low likelihood that there will be an event requiring the PNPP igniter system to be used.

A more detailed description of the hydrogen igniter system, and a description of CEI's preliminary evaluation of the system, is included in Sections II, III and IV of Applicants' direct testimony. Section II describes CEI's analysis of the ultimate structural capacity of the PNPP containment to withstand hydrogen burning. Section III provides a description of the PNPP distributed igniter system design. Section IV of Applicants' direct testimony discusses the accident scenarios, containment response and equipment survivability analyses, which are covered in CEI's preliminary evaluation of the PNPP Hydrogen Control System.

Applicants' direct testimony on Issue #8 demonstrates that CEI's distributed igniter system meets the Nuclear Regulatory Commission's recently-enacted hydrogen control requirements in 10 C.F.R. § 50.44(c)(3), in that (1) CEI has provided at PNPP a hydrogen control system capable of handling, without loss of containment integrity, an amount of hydrogen equivalent to that generated from a metal-water reaction involving 75% of the fuel cladding surrounding the active fuel region (excluding the cladding surrounding the plenum volume), and (2) CEI's hydrogen control system is supported by a suitable preliminary analysis, which provides a satisfactory basis for a decision that the plant is safe to operate at full power until CEI's final analysis of the PNPP hydrogen control system has been completed. Thus, Applicants' direct testimony demonstrates that PNPP is adequately designed to accommodate safely large amounts of hydrogen without a containment rupture and release of radioactivity, contrary to OCRE's contention as set forth in Issue #8.

Q4: What led to CEI's consideration of measures at PNPP to mitigate the effects of a release of large amounts of hydrogen resulting from a postulated degraded core accident?

A4: (Buzzelli) The accident at Three Mile Island, Unit 2 in 1979 involved a metal-water reaction in the core resulting in the release of amounts of hydrogen beyond the NRC's design basis accident requirements. Following the TMI accident, the NRC required a number of design improvements to all light water reactors (NUREG-0737/TMI Action Plan). These TMI-related design improvements were for the purpose of further reducing the likelihood and effects of degraded core accidents, beyond the inherent design capability of the plants. Pursuant to these TMI-related action items, the NRC staff requested CEI to provide a description of its program to improve the hydrogen control capability at PNPP. This sequence of events following the TMI-2 accident led CEI to consider a hydrogen mitigation system as part of overall design improvements at PNPP to further reduce the likelihood and effects of degraded core accidents.

Q5: What are the inherent design featues of BWR 6/Mark III plants that reduce the likelihood and effects of degraded core events, such as occurred at TMI-2?

A5: (Holtzclaw) There are BWR/6 design features associated with the plant heat sink, the pressure relief capability, reactor water level measurement, noncondensible gas venting and natural circulation, depressurization capability, and core cooling which preclude a TMI-2 type of accident from occurring and resulting in degradation of the reactor core. A comparison of the major complications at TMI-2 with the BWR/6 design, demonstrates that the unique features and characteristics of the BWR/6 will protect against and mitigate the type of accident scenario that occurred at TMI-2.

For example, with respect to heat sink, the TMI accident began with a loss of feedwater and unavailability of auxiliary feedwater which combined to isolate the reactor from its heat sink. As shown in Attachment A, the Mark III containment at PNPP has a large suppression pool inside the containment. The suppression pool provides a passive heat sink sufficient to accommodate stored thermal energy and decay heat from the reactor for several hours with the reactor isolated from its normal heat sink. In addition, the suppression pool serves as an effective filter to radioactive releases in the event of a low probability degraded core accident. The particulate fission products released from such postulated events will be effectively trapped in the suppression pool.

Another example is in the area of pressure relief capability. The TMI-2 event became a small break LOCA when a power operated primary relief valve stuck open, resulting in the overpressurization of its quench tank, discharge of primary system water to the containment, and activation of the Emergency Core Cooling System (ECCS). The functional equivalent of a power operated primary relief valve for the BWR/6 is the Safety Relief Valve ("SRV"). The SRV's open to relieve pressure increases that occur during expected transients and during certain accident conditions. Each SRV is piped to the large suppression pool. Since the SRV's are designed for routine use in BWRs, the isolation with "stuck open" valve is a design transient that is accounted for in standard plant and containment analyses. The containment would not be significantly pressurized by SRV blowdown. Automatic initiation of the makeup systems would maintain the reactor water level. There would be no TMI-2 type complications arising from SRV actuation and subsequent failure to close; such an event even considering failure of several makeup systems would be only a minor transient at PNPP.

A third example is the area of core cooling design features. Partial uncovering of the core at TMI-2 led to inadequate core cooling and resulted in core damage. The PNPP reactors are designed with highly redundant multiple water sources and injection delivery systems to maintain adequate core cooling. The diverse and redundant water supply capability to the PNPP reactor vessels is due in part to the direct cycle BWR design, in which normal pumping systems (feedwater, control rod drive cooling), and Reactor Core Isolation Cooling (RCIC) provide makeup water to the reactor vessel. In addition, Emergency Core Cooling Systems (ECCS) assure adequate cooling during an emergency via High Pressure Core Spray (HPCS), Low Pressure Coolant Injection (LPCI), and Low Pressure Core Spray (LPCS). These systems include the capability to spray the core from above and refill it from below at both high and low pressure. The HPCS, LPCI, LPCS and RCIC systems are shown on Attachment B.

In addition to the RCIC, HPCS, LPCI and LPCS systems, further pumping capability is available from the following pumps which can also deliver water directly to the core to ensure core coverage: feedwater pumps (2 pumps), control rod drive pumps (2 pumps) and condensate booster pumps (3 pumps). It is important to note that in most instances, any one of thirteen pumps is sufficient to maintain core coverage and prevent potential core damage and hydrogen generation.

These systems (RCIC, HPCS, LPCI, and LPCS) and additional pumping capability, together with the inherent natural circulation in a BWR, provide very large decay heat removal capability and ensure that the core temperatures will be maintained for any credible BWR accident well below the levels required for hydrogen generation by metal-water reaction.

Q6: Please describe the improvements since the TMI accident which have been or will be implemented at BWR/6 plants such as PNPP which will further reduce the likelihood and effect of events beyond design basis accidents.

A6: (Holtzclaw) Several significant improvements have been or will be implemented at BWR and other nuclear power plants. Examples of these improvements include: improved containment instrumentation to monitor containment pressure and radiation level and suppression pool water level following an accident; improved post-accident sampling capability to obtain a post-accident "grab sample" of reactor water and containment air at an accessible location to facilitate assessments of core damage; improved emergency procedure guidelines to provide plant operators with concise, symptom-oriented procedures to follow during an emergency; and control room improvements which provide an improved man-machine interface in the control room and facilities for responding to an emergency.

Two other post-TMI improvements specific to BWR design are: the automatic restart of High-Pressure Core Spray System at low reactor level in the event the operator takes manual control of the system and subsequently fails to maintain water level; and adding automatic depressurization logic for non-break events (e.g., loss of feedwater) accompanied by failure of all high-pressure cooling systems.

Q7: What analyses have been made to quantify the effectiveness of the BWR/6 plant features to assess their capability to preclude offsite risk from core damage or hydrogen generation events?

A7: (Holtzclaw) There have been a number of risk analyses and reliability analyses performed to confirm the capability of the BWR 6/Mark III design to prevent or terminate potential accident sequences and minimize the offsite risk. Probabilistic risk assessments have been performed to assess potential accident progressions and to quantify the probability of those accident sequences. One such study which is currently undergoing detailed review by the NRC staff and the Advisory Committee on Reactor Safeguards is the General Electric GESSAR II BWR 6/Mark III Standard Plant Probabilistic Risk Assessment. (Docket No. STN 50-447). GE's analysis concluded that accident sequences resulting in core damage occurred with a frequency of approximately 5×10^{-6} per reactor year. Approximately one-third of these events did not involve hydrogen-related failure modes. An independent review by the NRC staff and its contractors resulted in a core damage frequency value of approximately 2×10^{-5} per reactor year. (NRC GESSAR II Supplemental Safety Evaluation Report 2, dated November 1984). These values support the conclusion that core damage events which lead to significant quantities of hydrogen generation are very low in probability. These values are well below comparable frequencies calculated in the Reactor Safety Study (WASH-1400) and are also well below the proposed safety goal value for core damage frequency of 1×10^{-4} per reactor year.

In addition, analyses of the frequency of core damage and hydrogen generation have been made as part of the Industry

Degraded Core Rulemaking (IDCOR) Program. One of the primary technical conclusions of the IDCOR Program was that the probabilities of severe nuclear accidents are extremely low. (IDCOR Technical Summary Report, Nuclear Power Plant Response to Severe Accidents, dated November 1984). IDCOR's quantification of the probability of core damage and hydrogen generation for a BWR 6/Mark III plant ranged from 1.9×10^{-6} to 8.3×10^{-6} per reactor year.

In conclusion, risk analyses have shown that the risk of core damage and resultant hydrogen generation is expected to be very low. This is expected, in light of the multiple redundant water makeup systems to ensure core coolability and to prevent core damage and hydrogen generation, as discussed earlier.

Q8: When did CEI commit to a program to improve the hydrogen control capability, and identify the hydrogen mitigation system selected for PNPP?

A8: (Buzzelli) In a letter dated March 22, 1982, and in Amendment 8 to the FSAR, CEI responded to the NRC's followup request for a description of the program to improve the hydrogen control capability at the Perry Nuclear Power Plant. CEI indicated that a distributed igniter system had been selected for hydrogen mitigation at PNPP, and that the design would be based on igniter systems that had been developed at the Grand Gulf, Sequoyah, McGuire and D.C. Cook Nuclear Stations. The distributed igniter system selected for PNPP was the same post-TMI hydrogen mitigation concept being implemented by all ice-condenser and Mark III containment plants. The igniter design concept had been thoroughly analyzed and reviewed by these utilities and the NRC at the time it was selected for PNPP.

For BWR 6/Mark III plants, research and analysis to support the use of an igniter system to accommodate large amounts of hydrogen was being performed through the Hydrogen Control Owner's Group (HCOG). CEI indicated to the NRC staff in 1982 that it was participating in the HCOG program to address the use of igniters in Mark III containments and would be submitting plant-specific information in support of CEI's operating license application. CEI's commitments were reflected in Section 6.2.7 of the NRC staff's Safety Evaluation Report (NUREG-0887), dated May 1982.

In July 1984, CEI provided an update to the NRC staff on CEI's ongoing efforts to resolve the degraded core hydrogen control issue. At that time, CEI submitted a program plan describing the generic and plant-specific activities undertaken to resolve the hydrogen control issue for PNPP.

Q9: What is CEI required to provide under the NRC's new hydrogen control requirements, contained in 10 C.F.R. § 50.44(c)(3)?

A9: (Buzzelli) The final hydrogen control rule to address improved hydrogen control capability was published at 50 Federal Register 3498 (January 25, 1985). For the PNPP Mark

III containment design, the new rule in 10 C.F.R. § 50.44(c)(3)(iv)(A) requires a hydrogen control system capable of handling, without loss of containment structural integrity, an amount of hydrogen equivalent to that generated from a 75% metal-water reaction of the active fuel cladding. The hydrogen control system must be installed and operational prior to operation in excess of 5 percent power.

Under 10 C.F.R. § 50.44(c)(3)(vii)(B), CEI is required to provide a "preliminary analysis" to support an NRC staff finding that the plant is safe to operate at full power (in excess of 5% power). The rule provides that "completed final analyses are not necessary" for such a staff determination, provided a preliminary analysis has been submitted and found acceptable by the NRC staff. Under 10 C.F.R. § 50.44(c)(3)(vii)(A), CEI is required to submit by June 25, 1985 a proposed schedule for completing the final analysis. Under 10 C.F.R. § 50.44(c)(3)(vii)((D), a final schedule for meeting the requirements of the rule will be established by the NRC staff within 90 days of receipt of CEI's proposed schedule.

Q10: What has CEI done to meet the hydrogen control system and preliminary analysis requirements of the new rule?

A10: (Buzzelli) As indicated earlier in this testimony, prior to the new rule, CEI had committed to install a distributed igniter system at PNPP. Since the rule indicates that the NRC staff is to determine acceptability of a preliminary analysis, and in the absence of details in the rule defining the required scope of a preliminary analysis, CEI obtained NRC staff clarification prior to submittal of its preliminary information. In a letter to the NRC dated February 5, 1985 (Applicants' Exhibit 8-2), CEI identified the information on hydrogen control which would be provided in a preliminary evaluation, to address the preliminary analysis requirements of the new hydrogen rule. The letter indicated that CEI's preliminary evaluation would cover a description of the igniter system, an analysis of containment ultimate capacity, a containment response analysis, and a comparison of the significant design features at PNPP to those at Grand Gulf Nuclear Station (GGNS). The NRC responded in a letter dated February 20, 1985 (Applicants' Exhibit 8-3) that the scope of the information CEI was providing in support of a full power operating license for PNPP was acceptable.

CEI submitted its Preliminary Evaluation Report on March 1, 1985 and provided supplemental information on March 21, 1985. The Preliminary Evaluation Report covers four major areas of the PNPP design and analysis. These include a hydrogen control system description, a description of the structural capabilities of containment and drywell, a containment response analysis, and a design comparison to significant plant features of Grand Gulf Nuclear Station.

First, Section 2.0 of the Preliminary Evaluation Report discusses the design of the distributed igniter system, including the design criteria, igniter locations, power supplies, testing and system operation. The next area, addressed in Section 3.0 of the Report, describes the containment ultimate capacity analysis and evaluations of the containment negative pressure capability and the drywell positive and negative pressure capabilities. The third area, covered in Section 4.0, describes the analysis of the containment pressure and temperature response to hydrogen combustion, including the PNPP results provided as Appendix A to the Report. Finally, Section 5.0 of the Report provides a comparison of the PNPP design and supporting analysis with that of GGNS, which the NRC has licensed for full power operation on an interim basis. This comparison establishes the similarity of systems and analytical results, and provides additional basis for an NRC staff decision to support full power operation for PNPP. Supplementary information to CEI's preliminary evaluation in the areas of preoperational testing, and equipment survivability, and clarification of the containment response analysis information was submitted on March 21, 1985.

Q11: What is the basis for concluding that the PNPP will be safe to operate at full power with the installed and operable HCS, as supported by the preliminary evaluation?

A11: (Richardson) As addressed in this testimony, CEI's Preliminary Evaluation Report demonstrates that the PNPP HCS, using distributed igniters, is a viable hydrogen control concept, based on well understood combustion phenomena and extensive industry and NRC supported research and analysis. In addition, conservative plant-specific analyses, and comparisons to the GGNS design and analyses, provide sufficient preliminary basis to conclude that (1) the containment ultimate capacity is far greater than the pressures resulting from hydrogen combustion following a degraded core accident; thus, the structural integrity of the containment is maintained during and following hydrogen burning; and (2) the temperature for which the equipment is qualified to perform its function is greater than the expected equipment thermal response to the environment defined in the preliminary analysis of hydrogen combustion following a degraded core accident; thus, necessary systems and components will be capable of performing their functions during and after hydrogen burning.

Therefore, in the unlikely event that the HCS would be called upon, CEI's preliminary evaluation of the system demonstrates that it would safely accommodate large amounts of hydrogen without a rupture of the containment and release of radioactivity to the environment, contrary to the contention in OCRE's Issue #8.

Q12: Briefly summarize CEI's current plans for completing the final analysis.

A12: (Richardson) As noted earlier in this testimony, the new hydrogen rule does not require completed final analyses to be submitted prior to full power operation, provided that a preliminary analysis is submitted by CEI and approved by the NRC staff. In accordance with the new rule, CEI does intend to submit to the NRC staff by June 25, 1985 a proposed schedule for meeting the final analysis requirements. CEI's plans for completing a final analysis are consistent with those of Grand Gulf and the other participants in HCOG. HCOG is conducting an integrated program of analysis and testing to provide additional definition of the environmental conditions following hydrogen combustion resulting from a degraded core event producing large amounts of hydrogen. As the results of the HCOG program activities become available, CEI will evaluate and address their applicability to PNPP as necessary. The PNPP final analysis will be completed on a schedule consistent with the HCOG program, which is presently anticipated to be completed by mid-1986.

Q13: Please summarize the status of the HCOG program activities, and the principal areas to be addressed in the HCOG program on which CEI expects to base its final analysis.

A13: (Richardson) Since HCOG was formed in 1981, a significant amount of research and analysis has been completed on the hydrogen control issue. The activities undertaken and completed by the group include studies of postulated hydrogen generation events in response to the NRC's proposed hydrogen control rulemaking, and development of criteria for selection and design of the igniter system. HCOG also sponsored the development and verification of the CLASIX-3 computer code and sensitivity studies which support CEI's preliminary evaluation of the Mark III containment pressure and temperature response to hydrogen combustion. HCOG also sponsored hydrogen combustion tests to confirm flammability limits in steam environments and investigate Mark III combustion phenomena for different hydrogen release rates in a small (1/20th) scale facility.

HCOG is completing an integrated program of analysis and testing for final resolution of the hydrogen control issue. The principal area being addressed in the HCOG program is the further definition of the thermal environment resulting from the types of combustion phenomena in Mark III containments. The thermal environments defined under this program will be used to confirm the capability of necessary equipment to survive the environmental conditions resulting from hydrogen burning.

As previously stated, CLASIX-3 was used to calculate the pressure and temperatures to deflagration burning. Additional work is underway by HCOG to better define the thermal environment from deflagration burning produced by combustion at low hydrogen release rates. The 1/20th scale test referred to above was performed to characterize the combustion phenomena in the containment above the suppression pool and assess the thermal environment resulting from combustion of different hydrogen release rates. A larger (1/4 scale) test will be conducted to further define the thermal environment resulting from diffusion flames produced by combustion at higher release rates. The additional analysis and larger scale test results will provide an accurate prediction of the thermal environments to be used in further equipment survivability analysis.

HCOG is performing other related activities which support the thermal environment definition, such as refining the hydrogen release histories used for additional analysis, and testing and validation of the methodology used for defining the thermal environment and equipment temperature response.

II. CONTAINMENT ULTIMATE CAPACITY

Q14: What is the purpose of this section of the testimony?

A14: (Buzzelli) The purpose of this section is to describe the containment ultimate capacity analysis and drywell pressure capability evaluations which have been performed as part of CEI's preliminary evaluation of the PNPP Hydrogen Control System. These analyses are discussed in section 3 of CEI's Preliminary Evaluation Report. The analyses establish conservative pressure-retaining structural capacity values for the containment structures, using analytical techniques approved in 10 C.F.R. § 50.44(c)(3)(iv) of the new rule. When compared with the pressures predicted for a hydrogen event at PNPP, as discussed in Section IV of this testimony, the ultimate capacity analyses demonstrate that the PNPP containment structural integrity would be maintained in the unlikely event of a postulated degraded core accident generating large amounts of hydrogen.

Q15: What are the principal areas covered in the structural capacity portions of the PNPP Preliminary Evaluation Report?

A15: (Buzzelli) Section 3 of the Preliminary Evaluation Report discusses conservative internal and external pressure capabilities of the PNPP Mark III containment vessel, and separately discusses conservative internal and external pressure capabilities expected in the PNPP drywell structure. In response to the hydrogen rulemaking, which has focused on containment positive pressure, CEI requested G/C to perform a detailed analysis to calculate the internal pressure capacity of the PNPP containment. This analysis is set forth in the final report, "Cleveland Electric Illuminating Company Perry Nuclear Power Plant Units 1 and 2 Ultimate Structural Capacity of Mark III Containments," transmitted to the NRC on February 11, 1985 (G/C Report) (Applicants' Exhibit 8-4). Roger Alley was the principal author responsible for the G/C Report.

The Preliminary Evaluation Report demonstrates that the PNPP containment design is adequate to handle negative pressures following the combustion of hydrogen resulting from postulated degraded core accidents. This is primarily because PNPP has vacuum breakers which would alleviate negative pressures in the containment.

Finally, the Preliminary Evaluation Report establishes that the PNPP drywell pressure capability has substantial margin over design levels, and can adequately handle calculated pressures from hydrogen combustion (which are below design levels). This conclusion is supported by previously approved evaluations of the GGNS drywell structure, which are applicable to the PNPP drywell, as shown in the Preliminary Evaluation Report.

Q16: Is the actual pressure capacity of the PNPP containment greater than its design pressure capacity?

A16: (Alley) Yes. As demonstrated in the Preliminary Evaluation Report, the actual pressure capacity of the containment is significantly (over 3 times) greater than the design level. This is primarily because of the substantial conservatisms in the Code-allowable stress limits used in the design of the containment.

Q17: What is your understanding of the term "containment structural integrity," as used in the new hydrogen rule (10 C.F.R. § 50.44(c)(3))?

A17: (Alley) The rule indicates that containment structural integrity can be demonstrated by showing that the Service Level C Stress Limits provided in ASME Code, Section III, Division 1, Subarticle NE-3220 are met. The PNPP preliminary evaluation establishes the containment pressure at which the Service Level C limits can be met. It demonstrates that the pressures resulting from hydrogen combustion are within Service Level C limits.

Q18: Is there additional pressure-retaining capability in the containment beyond the ASME Service Level C limits referenced in the rule?

A18: (Alley) Yes. The hydrogen rule requires that containment structural integrity be maintained. The ASME Code permits higher Service Level D limits to be used where the primary intent is to assure that violation of the pressure retaining boundary will not occur. Also, the ASME code states that Service D limits are appropriate for extremely low probability postulated events. The postulated degraded core accident addressed in the hydrogen rule would fall in this category. Therefore, use of Service Level C limits to define the PNPP pressure-retaining capability, rather than the higher, more realistic Service Level D limits, represents a conservative approach to assuring that containment integrity will be maintained.

Q19: What are the principal areas addressed in the G/C Report on PNPP internal pressure capacity of the containment?

A19: (Alley) The Report addresses the internal pressure capacity of the PNPP containment vessel, including all components such as penetrations, personnel access airlocks, the equipment hatch, and penetration bellows. The containment vessel pressure capacity was evaluated using actual material strengths. The pressure capabilities of the penetrations were evaluated using stress concentration, finite element, and plastic analysis techniques. For most of the penetrations, as an additional conservatism, minimum specified material strengths were used, rather than using actual material strengths as allowed by the hydrogen rule. Where actual material properties were used, suitable margins were included (reported values are three standard deviations below the mean strength).

Q20: What are the principal conclusions of the G/C Report?

A20: (Alley) The report indicates that, using ASME Service Level C limits, the controlling lower bound pressure capacity for the PNPP containment is 50 psig for Penetration 414. Using the more realistic Service Level D limits, the Report shows that the controlling lower bound containment pressure capacity is 57 psig for the same penetration. These conservatively-calculated limits demonstrate an ample margin above the peak pressure of 21 psig after hydrogen combustion, as discussed in Section IV of this testimony.

III. PNPP HYDROGEN CONTROL SYSTEM DESIGN

Q21: What is the purpose of this section of your testimony?

A21: (Buzzelli) The purpose of this section is to describe the PNPP Hydrogen Control System, a distributed igniter system, which is provided at PNPP to control large amounts of hydrogen. The testimony describes the principal features of the HCS design and summarizes the design description in Section 2.0 of CEI's Preliminary Evaluation Report. The testimony demonstrates that PNPP has an acceptable hydrogen control system that can handle large amounts of hydrogen during and following a postulated degraded core accident, as required by 10 C.F.R. § 50.44(c)(3)(iv) of the new hydrogen rule.

Q22: What system was selected for PNPP to control large amounts of hydrogen released during a postulated degraded core accident?

A22: (Buzzelli) CEI selected a distributed ignition system as the most viable concept to control large amounts of hydrogen released during a postulated degraded core event. The system is designed to handle, without loss of containment structural integrity, an amount of hydrogen equivalent to that generated from a metal-water reaction involving up to 75% of fuel cladding surrounding the active fuel region. This is accomplished by burning hydrogen at low concentrations, thereby maintaining the concentration of hydrogen below levels which could potentially threaten containment integrity.

Q23: What was the basis for selecting this type of system to improve hydrogen control capability at PNPP?

A23: (Buzzelli) The technical criteria used in the selection of the igniter system for PNPP considered the mitigation effectiveness, consequences of intended or inadvertent operation, reliability, testability, and availability of design and equipment. Following the Three Mile Island accident, and prior to CEI's selection of an igniter-based hydrogen control system, the use of igniters to burn hydrogen at low concentrations to control large amounts had been implemented as the hydrogen control system at ice condenser plants (McGuire, Sequoyah, D.C. Cook) and at the first NRC licensed Mark III containment plant (Grand Gulf Nuclear Station). This ignition system design concept has been studied extensively. The design concept was reviewed and accepted on an interim basis by the NRC staff in the licensing of Grand Gulf Nuclear Station. There is a general technical consensus today within the industry that an igniter system constitutes an adequate system for controlling large amounts of hydrogen.

Q24: Describe the reviews performed by Combex to confirm the viability of the igniter system design concept for controlling large amounts of hydrogen in nuclear plants.

A24: (Lewis and Karlovitz) Combex has evaluated the use of a distributed ignition system for the control of large amounts of hydrogen resulting from a postulated degraded core accident, at both PWR ice condenser and BWR Mark III containments. We have reviewed the igniter systems at the McGuire, Grand Gulf, and Perry plants. We have physically observed the containment and pertinent systems inside the containment at these plants, and have reviewed drawings, specifications and other related documentation. Our review of the PNPP igniter system indicates that it will operate in the same manner as the GGNS igniter system from the standpoint of the basic combustion phenomena that can be expected to occur, and that both igniter systems will be able safely and effectively to burn large amounts of hydrogen. We have concluded that the igniter system design concept as implemented at all of these plants is adequate to prevent significant accumulations of H_2-air mixitures. Given their established containment capacities, the plants can safely accommodate the pressures which would result from the burning of hydrogen.

The PNPP igniter system will assure that hydrogen at low (8%) concentration in the presence of air will be ignited by the large number of igniters located at various locations in the containment and burn without abrupt pressure rise. This burning would be repeated as succeeding flammable mixtures are formed. This conclusion is substantiated by experimental and theoretical data. We have evaluated the PNPP igniter itself which is identical in all three plants we have reviewed. The glow plug igniters have been extensively tested and have demonstrated reliable ignition. The igniter assembly design and locations at PNPP will ensure an adequate flow of hydrogen-air mixtures to the igniter. The number and placement of igniters at PNPP are such that local detonations are unlikely to occur. Finally, based on our evaluation, there is no danger of transition of deflagration to detonation. For these containment designs, large volumes of H_2-air mixtures with a composition within the detonable range cannot accumulate in geometrical configurations condusive to transition.

Q25: Briefly describe the PNPP distributed igniter system.

A25: (Richardson) The PNPP hydrogen control system consists of thermal glow plug igniters spaced throughout the drywell, wetwell and upper containment regions of the plant. The igniters are maintained at a high surface temperature (1700°) which assures ignition of hydrogen in a controlled manner at or near its lower combustion limit as it is released.

The igniter assemblies used in the Hydrogen Control System are divided into two components: the igniter enclosure (which partially encloses the igniter and contains the terminal block, transformer, and associated electrical wiring), and the junction box which contains the cable termination.

A sketch of the igniter assembly is attached (Attachment C). The hydrogen igniters are powered from 120VAC, 60 Hz, Class 1E power distribution panels. These power panels receive their power from Class 1E motor control centers through 15 KVA transformers and a fuse panel. The fuse panel consists of a 40 amp and 45 amp fuse in series for each line to the 120 volt distribution power panels. Each transformer is fed from a Class 1E MCC breaker from a Class 1E bus which is capable of being powered from one of the emergency diesel generators. The 102 igniters are divided into six groups of approximately equal number, three groups in Division 1 and three groups in Division II. Each group is powered from a separate distribution power panel.

Q26: What are the major design criteria implemented for the HCS?

A26: (Richardson) The PNPP hydrogen ignition system is designed with suitable redundancy to assure that no single active component failure, including power supply failures, will prevent functioning of the system. The system is designed as a safety grade system, and is capable of operating for the duration of the hydrogen generation event. The HCS igniter assemblies are classified and designed as electrical Safety Class 1E and Seismic Category 1. Each igniter is powered from a Class 1E power supply which can be powered from one of the emergency diesel generators, as noted above.

Igniter locations were based on criteria that considered potential hydrogen release locations, appropriate spacing in open areas, redundancy and potential for pocketing in enclosed regions. Based on these criteria, igniters are located in a ring above the suppression pool and throughout the containment and drywell. The igniters are located approximately every 30 feet with alternating divisional power supplies, such that a distance of approximately 60 feet may exist if only one emergency power division is available. These criteria were used except in the open area above the refueling floor in the upper containment and in the reflood region in the drywell. Two igniters, one from each power division, are located in enclosed containment areas which could accumulate hydrogen. Based on the above criteria, 102 locations in the containment and drywell will have igniter assemblies. The number and arrangement of igniter assemblies are similar to those at the Grand Gulf Nuclear Station.

Q27: Explain the operation of the system.

A27: (Richardson) The Hydrogen Control System is manually placed in service when the reactor water level reaches the top of the active fuel. Actuation at this time allows the operator sufficient time to manually energize the igniters by two ON-OFF handswitches located on a control room panel. As seen in Appendix A to the Preliminary Evaluation Report, hydrogen burning occurs no sooner than one hour after the onset of the accident, which is consistent with other generic analyses of the more probable degraded core accidents.

After manual initiation, the igniters are powered continuously for up to seven days. The system is manually de-energized by the operator turning both handswitches to "OFF" when the hydrogen generation event has passed.

Q28: What testing is conducted on the HCS to verify operability?

A28: (Richardson) Both preoperational and surveillance testing will be performed on the HCS to verify correct operability of the system. Preoperational testing will include energizing one of the two divisions from the control room and verifying that all igniters powered from the associated panel are functional. Identical procedures will be followed for the remaining igniters powered from the other division. Functional testing of the system will verify that the surface temperature of the igniters is adequate, and that the power supply transformers and the igniter transformers are providing satisfactory voltages. With the testing described above, reliable ignition of hydrogen at low concentrations will be assured using the PNPP HCS.

Q29: What role do the hydrogen recombiners play in controlling hydrogen concentration in the PNPP containment?

A29: (Richardson) The recombiners are designed to maintain the hydrogen concentration in the containment below 4 volume percent for the small amounts of hydrogen associated with a design basis accident (equivalent to a 1% metal-water reaction or less.)

Following the generation of large amounts of hydrogen from a degraded core accident (equivalent to up to 75% metal-water reaction), the HCS will be used for controlled combustion at low concentrations. The recombiners would be utilized for long-term control of the remaining hydrogen, not consumed by HCS burning, with sufficient oxygen in the containment atmosphere.

Q30: Describe the PNPP recombiner system.

A30: (Richardson) The hydrogen recombiner system at PNPP is part of the Combustible Gas Control System (CGCS), which is designed to prevent flammable hydrogen concentrations from forming in the containment or drywell following a loss-of-coolant accident (LOCA) in which hydrogen has been generated from the metal-water reaction within the reactor vessel and from long-term sources of hydrogen such as radiolysis. The recombiner system consists of two 100% capacity recombiners located in the containment, and a control panel and power supply cabinet located outside containment.

Following the postulated LOCA, the hydrogen recombiners are used to reduce the volume of hydrogen within the containment vessel. Air flows by natural convection through the recombiners, is heated above the hydrogen-oxygen recombination temperature, and the hydrogen present reacts with oxygen to form water vapor. The hydrogen recombiner system is intended for use before the containment hydrogen concentration reaches 4%.

Q31: Briefly describe the purpose of the CGCS.

A31: (Richardson) The CGCS is designed to control the concentration of hydrogen which may be generated and released following a design basis accident. This is accomplished by mixing volumes of high concentration with those of low concentration and recombining the hydrogen and oxygen to control the long-term buildup of hydrogen in the containment. This system is designed to meet requirements in 10 C.F.R. § 50.44 (unaffected by the recent amendments to the rule) for combustible gas controls following a design basis loss-of-coolant accident. As described in FSAR Section 6.2.5, the CGCS consists of the following subsystems: the hydrogen mixing or drywell purge subsystem, the hydrogen recombination subsystem (described above), the hydrogen analysis subsystem and a backup containment purge subsystem.

IV. ACCIDENT SCENARIOS, CONTAINMENT RESPONSE, EQUIPMENT SURVIVABILITY, AND BURN PHENOMENA

Q32: What is the purpose of this section of your testimony?

A32: The purpose of this section is to explain the scenarios and computer models used in the PNPP HCS preliminary evaluation to predict hydrogen releases, combustion, and the resulting thermal and pressure environments and effect on equipment survivability that can be expected in a postulated degraded core event producing large amounts of hydrogen.

Q33: What events should be considered in a preliminary evaluation of a postulated degraded core accident?

A33: (Holtzclaw) Industry and NRC considerations of accident event indicators have been divided into two general categories -- those resulting from postulated pipe breaks, and those caused by plant transients compounded by multiple failures. The design basis analysis (DBA) approach focused on the DBA loss-of-coolant accident (LOCA), because of the bounding consequences from that class of events. Based on analyses performed after the Reactor Safety Study (WASH-1400), especially those performed after TMI, there has been a marked trend to place more emphasis on the more probable events, which are those initiated by plant transients. This emphasis is supported by risk analyses, which conclude that the scenarios which dominate plant risk are transient-initiated.

Q34: What scenarios were selected in the preliminary evaluation to evaluate the consequences of hydrogen combustion in the PNPP containment?

A34: (Richardson) As described in Section 4 of the Preliminary Evaluation Report, two different events were analyzed for PNPP to investigate the temperature and pressure response of the reactor containment to a postulated degraded core event with hydrogen generation, release and combustion. One event represents safety relief valve actuation during a reactor pressure transient and subsequent failure of the valve to close. No Emergency Core Cooling System (ECCS) flow is assumed in order to simulate a situation which would result in significant core degradation. Just prior to reaching a metal water reaction equivalent to 75% of the active fuel cladding, recovery of coolant makeup systems is assumed to occur and the transient is terminated. In the second scenario, the release of hydrogen results from a small steam-line break in the drywell, rather than a stuck open relief valve. After the initiating event, the same assumptions about ECCS unavailability, and core recovery, are made as were applied for the first scenario.

Q35: What is the basis for choosing the stuck open relief valve and small break in the drywell events for the preliminary analysis?

A35: (Richardson) As discussed above, the two degraded core events CEI chose for its preliminary evaluation were: (1) a small break in the drywell (DWB) LOCA with extended failure of ECCS injection, and (2) a transient with a stuck-open relief valve (SORV) accompanied by a extended failure of the ECCS. The SORV was chosen as the base case recoverable degraded-core event, because of risk studies showing the higher core melt frequency of transient-initiated events versus the LOCA events. For transient-initiated events which result in a postulated recoverable degraded core, all releases from the reactor system are directly into the containment through the safety relief valve lines into the suppression pool. The small break LOCA was included for evaluation in order to consider the potential consequence of hydrogen release directly to the drywell. The PNPP preliminary evaluation utilized scenarios which are representative of the reactor system for a postulated accident resulting in a degraded core, and have been accepted by the NRC staff in previous plant licensing reviews. The combination of mass and energy releases from these postulated events is representative of a wide variety of postulated degraded core situations in which hydrogen generation may be a factor.

Q36: What was the basis for the reactor coolant system response and hydrogen release rates used in the PNPP preliminary evaluation?

A36: (Fuls) The steam and hydrogen releases from the reactor coolant primary system resulting from a postulated degraded core accident are input into the containment response analyses. The PNPP analysis was based on the reactor coolant system response and release using results from the MARCH computer code. The MARCH computer program was developed by Battelle-Columbus for the NRC.

MARCH models the release of hydrogen with the steam from whatever openings in the primary system may be appropriate for the scenario. The specific transient analyzed using MARCH was the SORV event. For this event, the hydrogen and steam releases are directly introduced into the suppression pool through the safety relief valves. The PNPP preliminary evaluation of the DWB event modeled the SORV hydrogen and steam releases as entering the drywell through the break, and into the suppression pool through the safety relief valves. Using the mass and energy releases calculated for the SORV event in the DWB evaluation is equivalent to assuming a break the size of the SORV opening. This is conservative since the SORV opening is larger than the normally analyzed small line break LOCA which is the basis for the DWB case.

The transient analyzed using MARCH was not mitigated. After the water in the core region was boiled away, the core continued to heat up until it severely melted. Since this transient analysis went well beyond a recoverable degraded core, the MARCH results were modified to simulate core recovery. The quenching and recovery were not mechanistically calculated. Instead, the hydrogen release rate was held constant at the previous peak rate, and the hydrogen reaction was terminated when 75% oxidation of the active cladding was reached. In the PNPP containment response analysis the MARCH results were modified to be consistent with the hydrogen rule.

Q37: Describe the computer code used for analyzing the PNPP containment response to hydrogen combustion.

A37: (Fuls) The computer program used for analyzing the PNPP containment response to hydrogen combustion was the CLASIX-3 program. Following the TMI-2 accident, the original CLASIX Code was developed at Westinghouse, Offshore Power Systems, Inc., to analyze the effects of hydrogen combustion on the ice condenser containment. CLASIX-3 was developed by modifying the original code to represent the BWR 6/Mark III containment design. CLASIX-3 has the capability to model Mark III containment plant features (including the suppression pool, refueling pool, vacuum breakers, and drywell purge system) while tracking the distribution of the atmosphere constituents, i.e., oxygen, hydrogen and steam. The CLASIX-3 code also has the capability of modeling containment sprays and structural heat sinks.

CLASIX-3 applies the laws of thermodynamics and uses standard engineering equations for the conservation of mass and energy, chemical reactions, heat transfer and fluid flow. It also uses standard engineering assumptions and practices such as treating non-condensible gases and highly superheated steam as perfect gases. The code also utilizes finite differential heat transfer equations, perfect mixing and a single temperature within each control volume, and a single temperature in each finite element in passive heat sinks.

Q38: Describe the model used in CLASIX-3, including the key assumptions for analyzing the consequences of hydrogen combustion of PNPP.

A38: (Fuls) A diagram of the Mark III containment and a schematic diagram of the PNPP CLASIX-3 model used in the analysis are shown in Attachments D and E. There are three compartments in the model: the drywell, wetwell and containment. Also included in the model are the key Mark III features of the suppression pool, containment spray system, upper pool, and drywell purge system. The basic CLASIX-3 model for the PNPP analysis is identical to that used for the preliminary GGNS analysis.

Mass and energy released to the contaiment atmosphere in the form of steam and hydrogen are provided as input to the code. The burning of hydrogen is calculated in the code with provisions to vary the conditions under which hydrogen is assumed to burn and conditions at which the burn will propagate to other compartments. In the stuck open relief valve case, all reactor system releases are assumed to enter the bottom of the suppression pool on the containment side. At twenty minutes into the transient, the igniters and two Combustible Gas Control System compressors are manually activated and begin pumping gas from the containment into the drywell. At 30 minutes into the transient, the upper pool begins dumping water into the suppression pool and at 6500 seconds into the transient, drawdown of the suppression pool to refill the reactor vessel begins.

In the drywell break case, all reactor system releases are initially to the drywell. At twenty minutes into the transient, the Automatic Depressurization System is assumed to be manually actuated and 50% of all subsequent releases are assumed to exit the reactor system through the safety relief valves directly into the suppression pool. All other assumptions are the same as for the stuck open relief valve transient.

Both transients are continued until hydrogen equivalent to 75% of the clad in the active fuel region has been reacted. At this time, the stuck open relief valve transient is terminated. In order to assess the consequences of the large accumulation of hydrogen in the drywell at the end of hydrogen release, the drywell break transient was continued until at least one burn occurred in the drywell.

Q39: What was the basis for selecting the CLASIX-3 model for the PNPP analysis?

A39: (Fuls) The basic model in CLASIX-3 for analyzing hydrogen combustion in a Mark III containment was developed for the Grand Gulf Nuclear Station. The original model developed for GGNS was based upon consideration of the postulated events being analyzed, hydrogen combustion phenomena, Mark III plant design features applicable to the hydrogen generation event, and the Mark III containment design and geometry.

Sensitivity analyses conducted with the GGNS model demonstrated that the basic three compartment model provided conservative results when compared to models with more or different compartments. In addition, these sensitivity studies showed that the key design features modeled and key assumptions were applied in an acceptable manner.

Because of the similarity of plants, CEI used the identical model in the PNPP containment response analysis. The input parameters were modified as necessary to account for differences in design features or design values.

Q40: What is the basis for the input parameters used in the PNPP CLASIX-3 analysis?

A40: (Fuls) The input parameters for the ignition and combustion of hydrogen were based on existing experimental data. The remaining parameters are plant design values (such as spray flow rate, heat transfer areas and compressor flow rate), or based on engineering judgement (such as flow loss coefficients) or handbook values (thermal conductivity, emmissivity and densities).

Q41: Does Combex agree that the analysis in the Preliminary Evaluation Report is consistent with the theoretical and experimental data relative to the burning of hydrogen?

A41: (Lewis and Karlovitz) Yes. Given an ignition source, hydrogen in concentrations above 8.0% will burn in a propagating manner. Therefore, for the analysis delineated above, the assumption of propagating burns is justified. The pressures calculated from the hydrogen burns in the analysis are conservative, because experimental results show that theoretical pressures are not realized for burns of hydrogen below about 12% concentration. Based on our review of the plant, the PNPP igniters will ignite mixtures which are within the flammable limit.

The burn parameters used in the analysis and presented in Table 4 of Appendix A to the Preliminary Evaluation are consistent and conservative with respect to the theoretical and experimental data relative to hydrogen combustion.

Q42: What basic characteristics of hydrogen behavior are judged by Combex to support the conclusions contained in this testimony?

A42: (Lewis and Karlovitz) There are two important characteristics: distribution characteristics and burning characteristics of hydrogen. With respect to distribution characteristics, hydrogen rapidly mixes with other gases. Mixing results primarily from entrainment of air and hydrogen by jets originating from a high energy line break, from natural convection flow generated by density differences, and from turbulance. A high degree of turbulance promotes rapid and complete mixing. Such turbulance will result from many sources such as the blowdown from a high energy line break, containment sprays, and small localized burning of hydrogen. Hydrogen mixing by the above described processes will result in uniform concentrations in any given compartment as assumed in the analysis.

Flammability limits and burning velocity are the important burning characteristics relevant to the PNPP preliminary evaluation. Deflagration is the propagation of a slow flame through a flammable mixture. The hydrogen concentration range over which burning can occur is bounded by the limits of flammability. At ordinary temperature and pressure, the lower deflagration limit (ordinarily referred to as the lower flammability limit) of hydrogen in air is 4.0% hydrogen by volume for upward propagating flame, about 6.5% hydrogen for horizontal propagation, and about 8.0% hydrogen for downward propagation. The PNPP CLASIX-3 analysis conservatively assumes ignition and propagation of hydrogen at 8.0%.

Burning velocity is the rate at which the flame propagates into a quiescent mixture at right angles to the flame surface. Burning velocity is a fundamental parameter of flammable mixtures, which define the characteristics of propagating flames. The PNPP CLASIX-3 analysis assumes a flame speed of 6 ft/sec., which is conservative for the hydrogen concentration and burning conditions which are expected to exist.

Q43: What do the results of the PNPP CLASIX-3 analysis show?

A43: (Fuls) The CLASIX-3 results for PNPP show that frequent, periodic burns (deflagrations) occur in the wetwell region of the containment for both cases analyzed. In the SORV case, two burns occur in the containment volume. In the DWB case there are no spontaneous burns in the containment volume. Near the end of the analysis of the DWB, the hydrogen concentration in the containment was still below the specified criteria for ignition, but for conservatism, the criteria were lowered to produce a burn in order to evaluate the consequences of a burn in the containment.

During the initial portion of the DWB, only steam is released. For the first twenty minutes, all of the release is directed to the drywell resulting in the drywell being purged of all air. Subsequently, the compressors return some air to the drywell, but the concentration of oxygen is well below the ignition limit. After the termination of the releases, it is assumed that the operation of the compressors continues until the ignition limit of oxygen is reached and combustion occurs in the drywell.

During both transients, there are many brief temperature excursions in the wetwell on the order of 800°F, and pressure excursions on the order of 6 psig. For a few burns, the temperature in the wetwell reaches approximately 1760°F and pressures reach approximately 21 psig for a short period of time due to coincident wetwell and containment burning.

Q44: How do the PNPP results compare to other MARK III analyses?

A44: (Richardson) A summary of results for both PNPP and GGNS are presented in Table 18 of Appendix A to the PNPP Preliminary Evaluation Report. The comparison shows the plant responses to be generally similar except for differences due to differences in plant geometry, plant design features, and sequences of events.

The peak containment pressures and peak drywell to containment differential pressures for the two cases analyzed are comparable but somewhat lower for PNPP than for GGNS. The peak temperature for the PNPP analyses are also comparable in magnitude to those of the GGNS analyses, with the exception of the wetwell peak temperature in the SORV case. The PNPP wetwell temperature is higher due to a coincident containment and wetwell burn, which did not occur in the GGNS case. This coincident burn is due to plant geometry differences modeled in the analysis.

The higher peak wetwell temperature at PNPP will not have significant effect on the overall equipment temperature rate. This is because the burns are of short duration when compared to the time required for heat transfer to the equipment and the resulting temperature increase. In addition to peak burn temperature, the number of burns and timing between burns are also important parameters which affect equipment tempertures. Although comparable to GGNS, the number of burns at PNPP is lower and the time duration between burns at PNPP is larger than at GGNS, which is expected to result in lower equipment temperatures.

Q45: What preliminary work has been done to show that the PNPP equipment will survive a hydrogen burn environment?

A45: (Buzzelli) As indicated in Section 4 of the Preliminary Evaluation Report, the thermal environment at PNPP produced by deflagrations has been preliminarily defined using the CLASIX-3 computer code. The deflagration thermal environment at PNPP should be less severe than the corresponding environment at GGNS, based on the number of deflagrations which occur, and the time duration between the deflagrations.

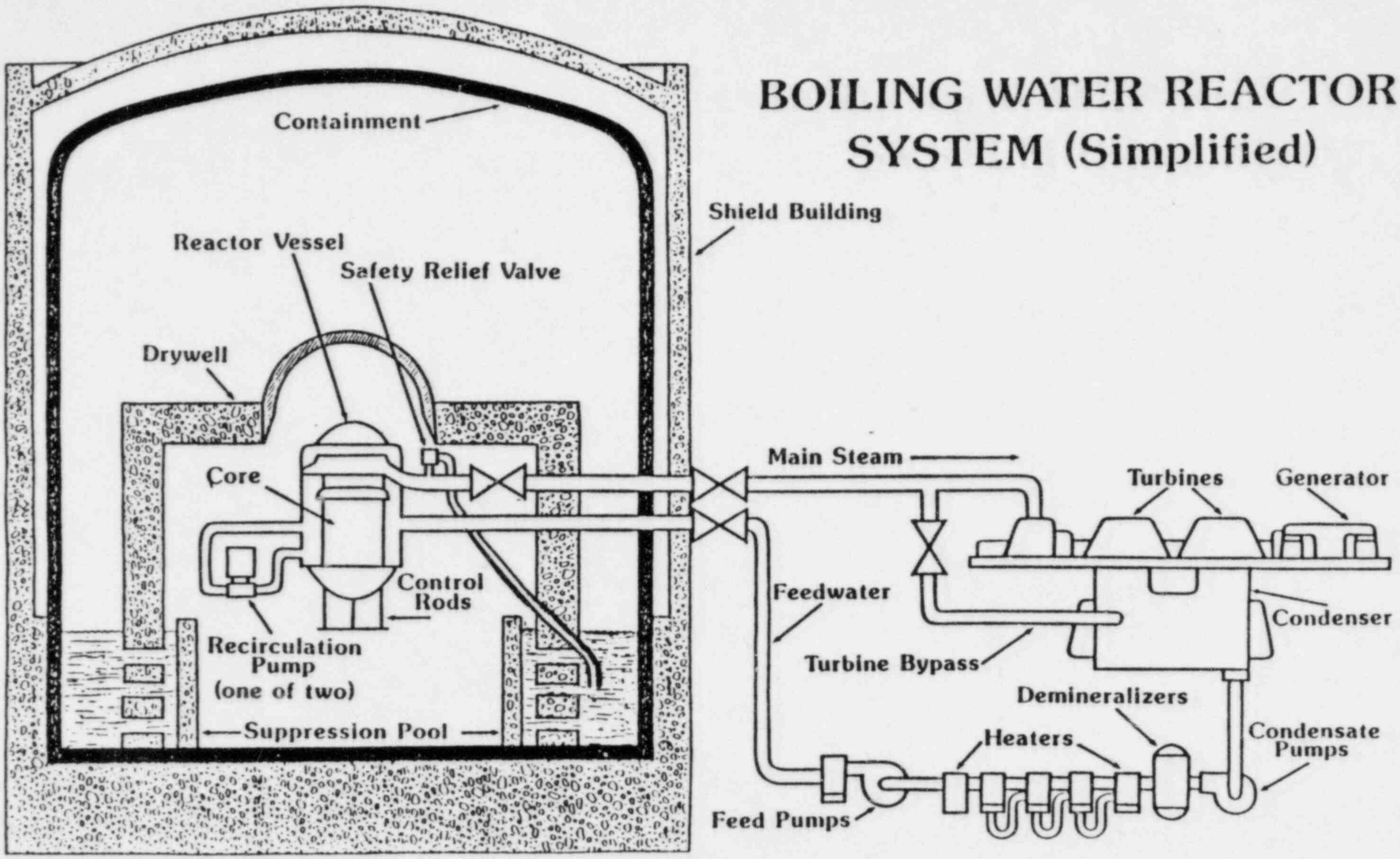
CEI has completed an analysis to verify that the PNPP CLASIX-3 temperature profile will result in lower equipment temperatures than do the GGNS CLASIX-3 temperature profile. The analysis compared the igniter assembly thermal response using the PNPP CLASIX-3 temperature profile, with the thermal response previously calculated by MP&L using the GGNS CLASIX-3 temperature profile. The same igniter assembly heat transfer model and assumptions were used to analyze the response of the igniter assembly to the PNPP CLASIX-3 profile. This analysis demonstrated the thermal response of the igniter to the PNPP temperature profile was lower than the response of the igniter to the GGNS temperature profile. MP&L previously completed an extensive program to evaluate the ability of equipment to survive hydrogen combustion. This program evaluated the ability of equipment to survive the thermal environment predicted by the CLASIX-3 computer code. The surface temperatures of the equipment at the end of the hydrogen combustion transient were shown to remain below the equipment environmental qualification temperature or the subcomponent most sensitive to thermally induced failure was shown to remain below the environmental qualification temperature.

As described in the Preliminary Evaluation Report, a preliminary identification and evaluation has been performed of equipment required to survive a hydrogen burn. The identification of the PNPP equipment required to survive was based on its function during and after postulated degraded core accidents. The criteria for selection of equipment included those systems needed for: mitigating the consequences of the accident, maintaining the integrity of the containment pressure boundary, maintaining the core in a safe condition, and monitoring the course of the accident.

These criteria are consistent with the criteria used for selecting equipment for the analysis to support the operating license for GGNS. Using these criteria, equipment in the containment and drywell which must function during and after a hydrogen burn were identified. The following is a preliminary list of the general systems and components: Containment structure, penetrations, locks and hatches; Hydrogen Control System; Combustible Gas Control System; Emergency Core Cooling System (HPCS, LPCS, LPCI, ADS); RHR Containment Spray; Containment and Reactor monitoring instrumentation; and associated instruments, controls and cable.

The preliminary list of equipment required to survive combustion is very similar between PNPP and GGNS. Many of the specific components on the two survivability lists are identical. The analysis of the igniter assembly which has been completed demonstrates that the equipment at the PNPP will experience less severe thermal response to hydrogen burning than the equipment at the GGNS. Since the equipment at PNPP has been demonstrated to survive the CLASIX-3 thermal environment predicted for GGNS, it is appropriate to conclude for purposes of the preliminary evaluation that the equipment at the PNPP would survive hydrogen deflagration burning.

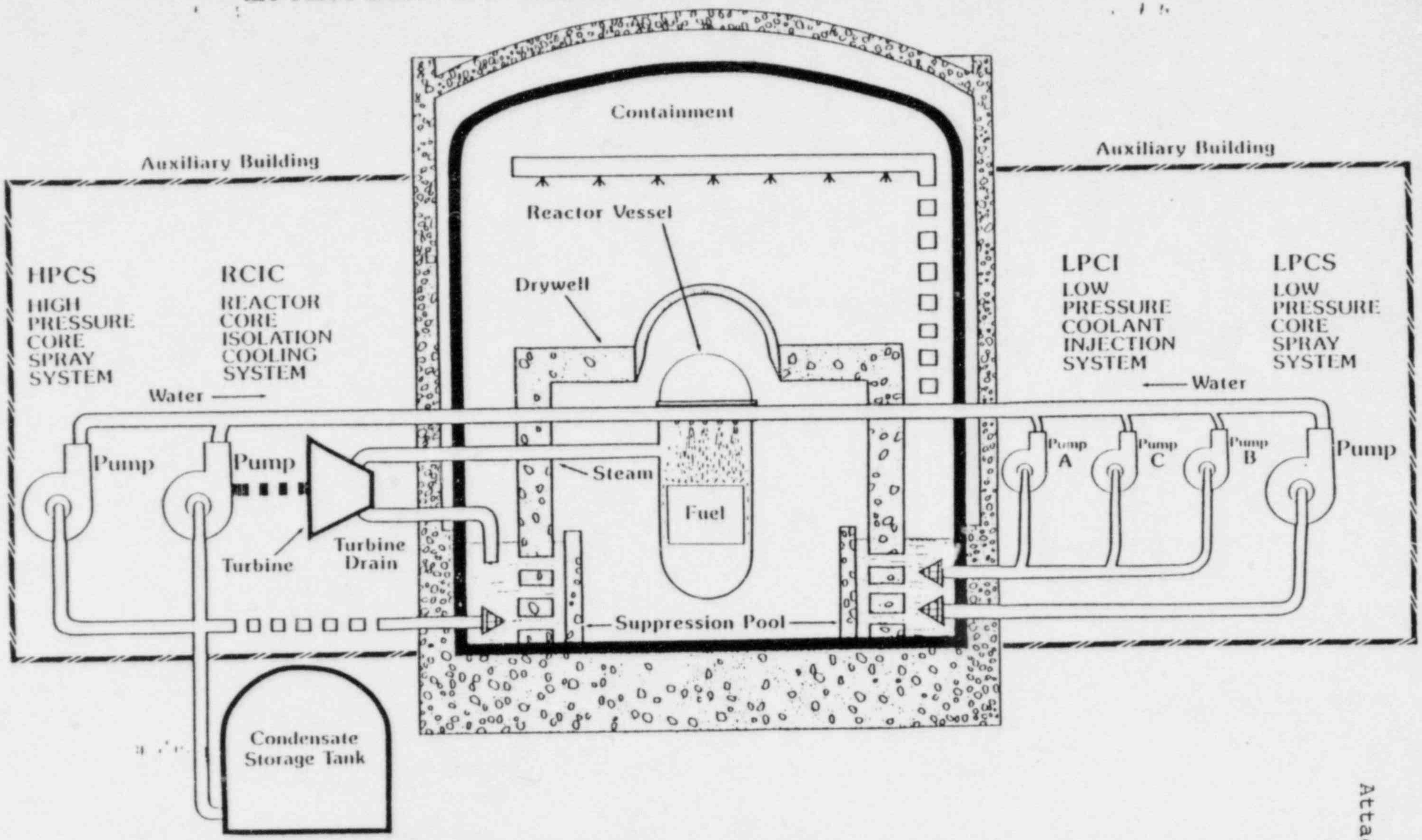
BOILING WATER REACTOR SYSTEM (Simplified)

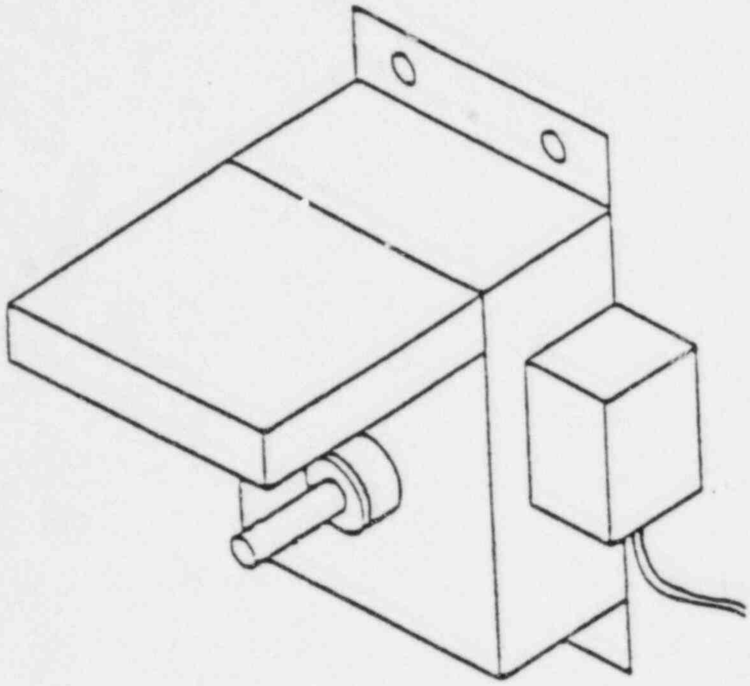


Attachment A

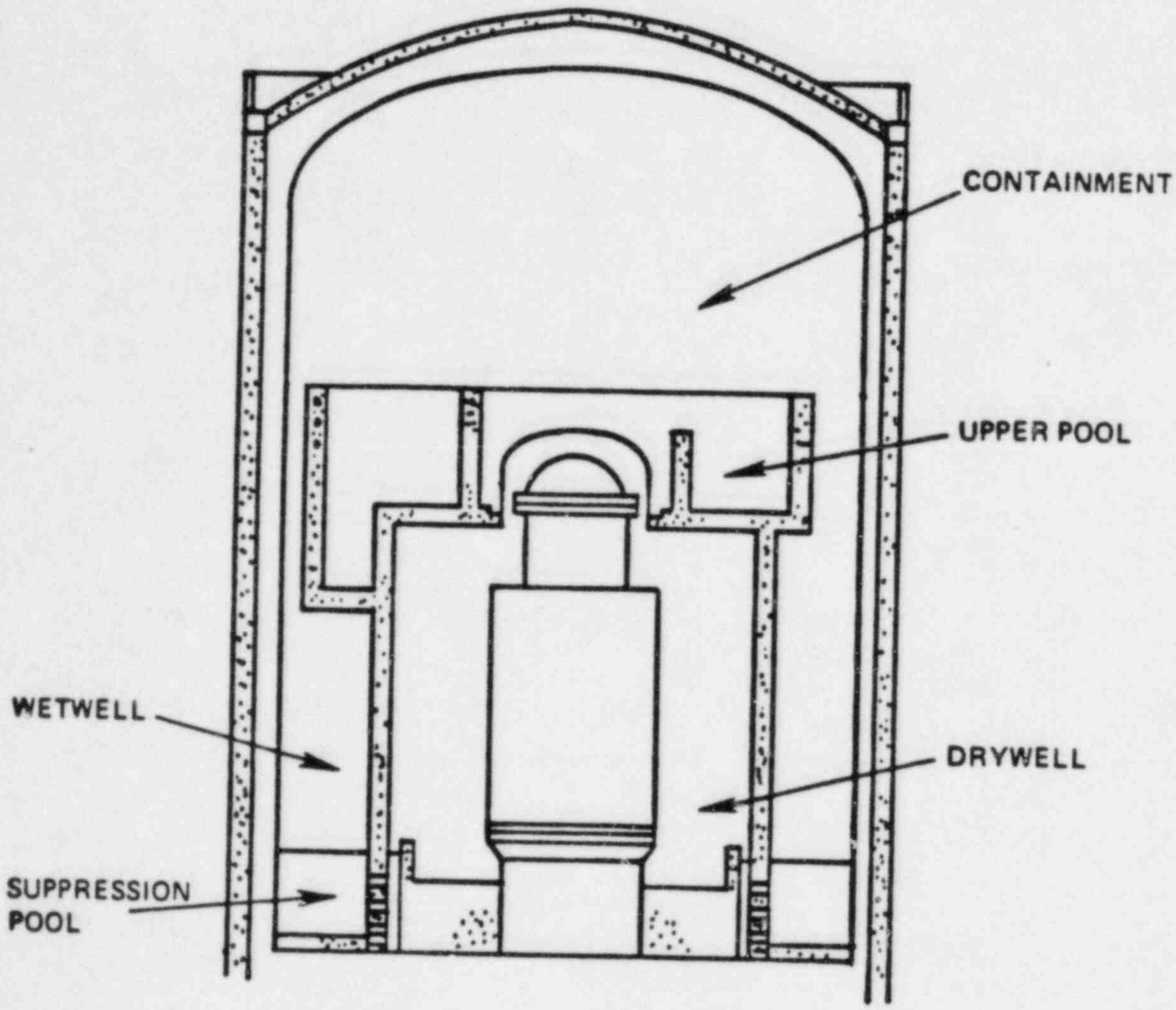
Attachment B

EMERGENCY COOLING SYSTEMS (Simplified)



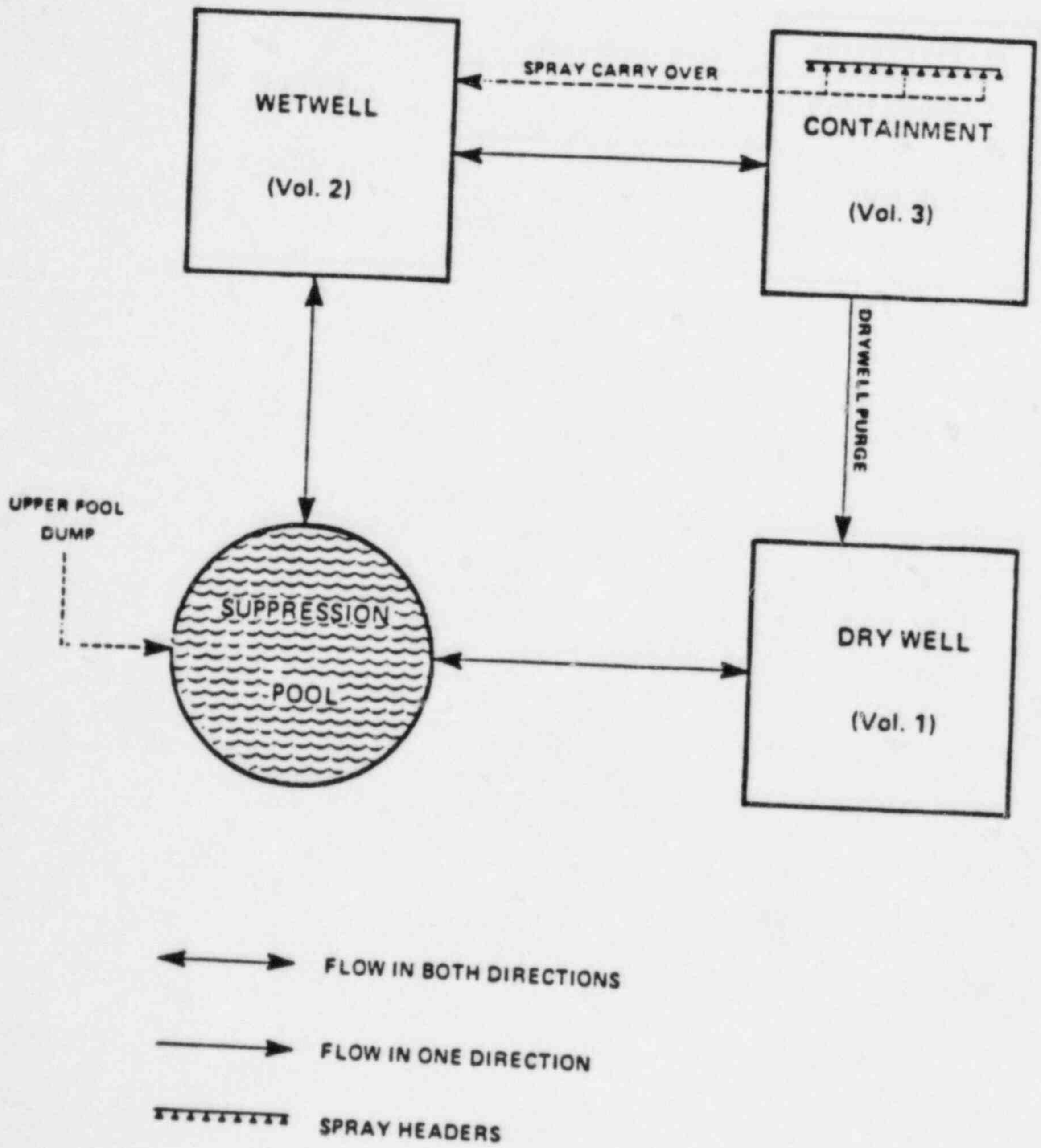


HYDROGEN IGNITER ASSEMBLY INSTALLATION



MARK III CONTAINMENT

PERRY CLASIX-3 MODEL



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1976-1978: Junior Engineer

Responsibilities included PNPP auxiliary systems design review and equipment procurement specifications.

1978-1980: Associate Engineer

Responsibilities included PNPP equipment and construction specifications, included fire protection system design and installation, and piping insulation.

1980-1984: Licensing Engineer

Responsibilities included NRC inspection and enforcement at PNPP and technical resolution of Final Safety Analysis review questions.

1984-Present: Senior Licensing Engineer

Responsibilities include coordination of all licensing activities and the resolution of technical issues with the Nuclear Regulatory Commission necessary to obtain the operating license for Perry.

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1977 - Dresden BWR Technology, BWR Operator Training including SRO certification
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1974-1975 - Westinghouse Nuclear Plant Engineer School, EOOW Prototype Training, Shift Supervisor Training School

Experience: Twelve years professional engineering and engineering management experience including experience in licensing, engineering, and safety analysis, and nuclear plant operations management experience (EOOW qualified and SRO certified).

1984 to Present: Enercon Services, Inc. Vice President of 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 reviews, engineering evaluations and analysis for hydrogen combustion in a Mark III containment.

1976 to 1983: Mississippi Power & Light Company. 1979-1983 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.

1977 to 1979: Operations Supervisor. Overall responsibility for the initial staffing and training of the Operations Section and the development of the operations procedures and program for Grand Gulf Nuclear Station. Overall management of Operations Section concerning all matters of plant operation through initial preoperational testing. 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.

1974 to 1976: Westinghouse Electric Corporation, Naval Reactors Facility. 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.

1973 to 1974: Motorola, Inc., Government Electronics Division. Electronic Design Engineer - Design , fabrication and testing of state-of-the-art satellite communications systems.

Memberships: Institute of Electrical and Electronic Engineers
National Society of Professional Engineers

KEVIN W. HOLTZCLAW

Education:	B.S. Mechanical Engineering (Nuclear Option), San Jose State University, M.S. Mechanical Engineering, University of California, General Electric Advanced Courses in Engineering
Experience:	March 1982 to Present: Principal Licensing Engineer, Program Manager of the GE Severe Accident Program February 1980 to March 1982: Senior Licensing Engineer BWR Systems Licensing (GE) June 1974 to February 1980: Technical Leader, Fuel Applicants & Thermal Design (GE) January 1971 to June 1974: Enginer, Fuel Applications and Thermal Design (CE) July 1969 to January 1971: Program Enginner Full Performance & Applications (GE) June 1968 to July 1969: Engineer - Nuclear Power Department (San Francisco Bay Naval Shipyard)
Licensing Experience:	Approximately 17 years engineering experience in the nuclear plant power industry. Since 1980, concentration on BWR licensing issues. As a senior licensing engineer through 1982, for defining and planning programs related to NRC degraded core rulemaking. Responsible for the safety & licensing program management of the Limerick Probabilistic Risk Analysis. GE representative on AIF Industry Degraded Core Rulemaking Technical Advisory Group. Since March 1982, GE Program Manager of the GE Severe Accident Program. This has entailed managing the BWR/6 standard plant Probabilistic Risk Assessment and Severe Accident submittals relating to evaluations beyond current design bases. Continued as the GE representative on the Industry Degraded Core Rulemaking (IDCOR) Technical Advisory Group. Responsible engineer in the GE Safety and Licensing organization for the GE Fission Product Retention Program and Severe Accident Source Terms and for programs relating to hydrogen generation and control. Numerous presentations to domestic and foreign regulatory groups and nuclear societies.

Additional Work Experience

Engineer and and technical leader in General Electric's Fuel Design Department from 1969 to 1980 responsible for performing Reload and Initial Core Fuel Thermal and Thermal-Hydraulic fuel design and safety analyses. Principal responsibilities in development of thermal analysis methods, the design and licensing of 8x8 fuel and extended exposure fuel designs, and in defining acceptance criteria for fuel thermal-mechanical fuel integrity properties and capabilities. Mechanical design engineer in the nuclear power department of the San Francisco Bay Naval Shipyard.

ROGER W. ALLEY

Education:

B.S.C.E., Virginia Polytechnic Institute, 1969
Post-Graduate Study, Engineering Administration, George Washington University, 1970-72

Additional Courses:
- Seminar, Finite Element Methods in Structural Analysis, Virginia Polytechnic Institute
- Elementary Theory of Structures and Computer Aided Structural Analysis, Catholic University
- Introduction to Environmental Protection, Gilbert/Commonwealth
- Structural Design, Analysis, and Testing of Nuclear Plant Equipment and Structures, Case Institute of Technology
- Construction Management, Gilbert/Commonwealth
- Stress Analysis of Shells, Lehigh University
- Practical Steel Design for Building 2 - 20 Stories, AISC
- Foundations of Management, Gilbert/Commonwealth
- Written Communications, Gilbert/Commonwealth
- Persuasive Communications, Gilbert/Commonwealth

Experience:

Fifteen years of professional experience in the structural analysis and design of major nuclear power generating facilities in the United States.

1981 to Present: Section Manager of Structural Engineering Nuclear and Project Structural Engineer for Cleveland's Perry Nuclear Station. Responsibilities as Section Manager include providing technical and administrative supervision to assigned personnel, developing and maintaining project staffing and scheduling plans, providing salary administration and performance appraisals for assigned personnel, providing assistance with corporate proposal activities, determining requirements for and participation in training programs.

1979: Project Structural Engineer for Cleveland Electric Illuminating Company's Perry Nuclear Power Plant, Units 1 and 2, 1200 MW each, BWR. Responsibilities as project engineer include providing technical and administrative supervision for a team assigned to the project. Act to ensure that production of drawings, specifications, reports, and analyses meet scope, schedule, cost, and quality requirements.

1978 to 1979: Lead Engineer for Fuel Handling Intermediate Buildings for Cleveland Electric Illuminating Company's Perry Nuclear Power Plant, Units 1 and 2, 1200 MW each, BWR. Responsibilities included: providing supervision and technical guidance to the engineers assigned to this area, defining scope of work and manhour estimates for the work, scheduling work and drawings, designing and design review of both structural steel and reinforced concrete structures, and FSAR preparation.

1972 to 1978: Structural Engineer. Designed reactor building complex for Cleveland Electric Illuminating Company's Perry Nuclear Power Plant, Units 1 and 2, 1200 MW each, BWR. Responsibilities included: preparation of PSAR, preliminary design and preparation of bid documents, preparation of specification, final design of structural steel and reinforced concrete structures, review and evaluation of bids, scheduling, working with the Site Organization with regard to resolution of construction problems and questions, and FSAR preparation.

1969 to 1972: Bechtel Corporation, Gaithersburg, Maryland. Structural Engineer - Designed reactor building complex for Cleveland Electric Illuminating Company's and Toledo - Edison Company's Davis-Besse Nuclear Power Station, 800 MW, PWR. Responsibilities included assistance in preparation of PSAR and specifications; performance of seismic analysis of reactor building complex, coordination and check of vendor drawings and calculations for ASME Containment Vessel; checking of vendor drawings and calculations on Reactor Polar Crane; final design of reinforced concrete and structural steel structures; writing responses to questions of the NRC; and working with Construction Management Group with regard to resolution of construction problems and questions.

Memberships: Professional Engineer - Pennsylvania, 1973
American Society of Civil Engineers
American Concrete Institute

DR. BERNARD LEWIS

Education:	B.S. in Chemical Engineering, Massachusetts Institute of Technology (1923) M.A. Physical Chemistry, Harvard University (1924) PhD. Physical Chemistry, Cambridge University (1926) ScD. Honorary, Cambridge University (1953) National Research Fellow (National Academy of Science), 1926-1928 Research Guest, University of Berlin, 1928-1929
Experience:	1953 to Present: President of Combustion and Explosive Research, Inc. (Combex), 1016 Oliver Building, Pittsburgh, Pennsylvania 15222. Consultant on fundamental combustion problems for government agencies, industries, and research institutes. This activity involves work on, among other things, safety of test facilities handling hydrogen, piloting of hydrogen flames, and control of flame instabilities. 1954 to 1966: The Combustion Institute (International Scientific Society). Now Honorary President. 1955 to 1965: Commission on High Temperature in Gases, I.U.P.A.C. 1929 to 1953: U.S. Bureau of Mines. Chief, Explosives and Physical Sciences division. Director of all research activities in Bureau of combustion, flames, explosion, and explosives, including the experimental mine, fire research, ignition by static electricity and other sources; gas and dust explosions; explosion pressures, damage effects; burning of a solid, liquid and gaseous fuels. 1951 to 1952: Ordinance Corps, U.S. Army. Director of Research (Propellants and Explosives). For 3 1/2 years in Army in charge of explosives loading, research and development, artillery ammunition, mines, grenades during World War II, Lt. Colonel. Inventor of present standard U.S. Army Hand Grenade Fuze. U.S. Government. Consultant for Army, Navy, Air Force and National Bureau of Standards, Bureau of Ships, National Advisory Committee for

Aeronautics, N.A.S.A., Fire Research Conference of National Academy of Science, Technical Advisory Committee U.S. Department of Interior, A.E.C., consultant on Polaris, Poseidon and Trident (Navy).

N.A.C.A. Chairman, Combustion Subcommittee

NATO: Scientific representative to Advisory Group for Aeronautical Research and Development (Combustion and Flame).

Consultant to numerous industries and research institutes, development of advanced jet engines; explosion hazard prevention in industry; Nuclear Power Plant Safety.

Investigated numerous disasters involving explosions and fires.

Publications: Author of over 250 scientific articles published in scientific journals in several countries, mostly on combustion, flames, explosions and explosives.

Author of two books on "Combustion, Flames and Explosions of Gases". (Translated into Japanese and Russian and used as a textbook throughout the world.)

Author of five other books on combustion.

U.S. Editor of Journal, " Combustion and Flame".

Honorary Editor, Journal, "Oxidation Communications", Hungarian National Academy of Sciences.

Some published articles relating to H_2 burning by Bernard Lewis:

(1) "The Chain Reaction Theory of the Rate of Explosion in Detonating Gas Mixtures", Bernard Lewis, J. Amer. Chem. Soc., Vol. 52, 193, p. [illegible].0.

(2) "Explosio[illegible] in Detonating Gas Mixtures. I Calculatio[illegible] of Rates of Explosions in Mixtures of Hydrogen and Oxygen and the Influence of Rare Gases", Bernhard Lewis and J. B. Friauf, J. Amer. Chem. Soc., Vol. 52 (1930), p. 3905

(3) "Kinetics of Gas Explosions. III Influence of Hydrogen on the Thermal Decomposition of Ozone Sensitized by Bromine Vapor and the Determination of the Explosion Temperature", W. Feitknecht and Bernhard Lewis, J. Amer. Chem. Soc., Vol. 54 (1930) p. 3185

(4) "Kinetics of Gas Expolsions. IV. Ozone Explosions Induced by Hydrogen", Bernard Lewis, J. Amer. Chem. Soc. Vol. 55 (1933) p. 4001

(5) "The Efficiency of the Reaction $OH + H_2 = H_2O + H$ and its Bearing on the Reaction Between Hydrogen and Oxygen" G. von Elbe and Bernhard Lewis, J. Amer. Chem. Soc. Vol. 54 (1932) p. 552

(6) "On the Theory of Flame Propagation", Bernard Lewis and G. von Elbe, J. Chem. Phys. Vol. 2 (1934), p. 537

(7) "Determination of the Speed of Flames and the Temperature Distribution in a Spherical Bomb from Time - Pressure Explosion Records", Bernard Lewis and G. von Elbe, J. Chem. Phys. Vol. 2 (1934) p. 283.

(8) "Temperature and the Latent Energy in Flame Gases", Bernard Lewis and G. von Elbe, The Engineer (UK), Vol. 159 (1935) p. 230.

(9) "The Sodium Line - Reversal Method of Determining Flame Tempratures", Bernard Lewis and G. van Elbe, Engineering (UK) vol. 159 (1935), p. 168.

(10) "The Experimental Determination of the Theoretical Calculation of Flames Temperatures and Explosion Pressures", Bernard Lewis and G. von Elbe, Philosophical Magazine, Vol. 20, (1935), p. 44

(11) "The Reaction Between Hydrogen and Oxygen Above the Upper Explosion Limit," G. von Elbe and Bernard Lewis, J. Amer. Chem. Soc. Vol. 59, (1937) p. 656

(12) "The Steady-State Rate of a Chain Reaction for the Case of Chain Destruction at Walls of Varying Efficiencies", G. von Elbe and

Bernard Lewis, J. Amer. Chem. Soc., Vol. 59 (1937) p. 970

(13) "Kinetics of Explosive Reaction Between Hydrogen and Oxygen Sensitized by Nitrogen Peroxide", G. van Elbe and Bernard Lewis, J. Amer. Chem. Soc., Vol. 59 (1937) p. 2022

(14) "Theory of Flame Propagation", Bernard Lewis and G. van Elbe, Chem. Reviews, Vol. 21 (1937), p. 347

(15) "Mechanism of Complex Reactions and the Association of H and O_2" G. von Elbe and Bernard Lewis, J. Chem. Phys. Vol. 7 (1939) p. 710

(16) Physics of Flames and Explosions of Gases", Bernard Lewis and G. von Elbe. J. Applied Phys. Vol. 10 (1939) p. 344.

(17) "Flame Temperature", Bernard Lewis and G. von Elbe, J. Applied Physics, Vol. II (1940) p. 698

(18) "Reaction of Hydrogen and Oxygen in the Presence of Silva. The Third Explosion Limit", Harold R. Heiple and Bernard Lewis, J. Chem. Phys. Vol. 9, (1941) p. 120

(19) The Reaction Between Hydrogen and Oxygen: The Upper Explosion Limit and the Reaction in its Vicinity", G. von Elbe and Bernard Lewis, J. Chem. Phys. Vol. 9 (1941) p. 194

(20) "The Reaction Between Hydrogen and Oxygen: Kinetics of the Third Explosion Limit", Harold R. Heiple and Bernard Lewis, J. Chem. Phys. Vol. 9 (1941) p. 584

(21) "Mechanism of the Thermal Reaction Between Hydrogen and Oxygen", G. von Elbe and Bernard Lewis, J. Chem. Phys. Vol. 10 (1942) p. 366.

(22) "Stability and Structure of Burner Flames", Bernard Lewis and G. von Elbe, J. Chem. Phys. Vol. 11 (1943) p. 75

(23) "Ignition of Explosive Gas Mixtures by Electric Sparks I. Minimum Ignition

Energies and Quenching Distances of Mixtures of Methane, Oxygen and Inert Gases", M. V. Blanc, P. G. Guest, G. von Elbe and Bernard Lewis, J. Chem. Phys., Vol. 15 (1947) p. 798.

(24) "Ignition of Explosive Mixtures by Electric Sparks II. Theory of Flame Propagation of Flame from an Instantaneous Point Source of Ignition", Bernard Lewis and G. von Elbe, J. Chem. Soc. Vol. 15 (1947) p. 803

(25) "Mechanism of the Initiation of Chains in the Thermal Reaction Between Hydrogen and Oxygen", Bernard Lewis and G. von Elbe, Revue de l'Institut Francais du Petrole et Annales des Combustibles Liquides, Vol. 4, (1949) p. 363 (in French)

(26) "Theory of Inflammation, Extinction and Stabilization of Flames", G. von Elbe and Bernard Lewis, Revue de L' Institut Francais du Petrole et Annales de Combustibles Liquides, G. von Elbe and Bernard Lewis, Vol. 4 (1949) p. 374 (in French)

(27) "Ignition of Explosive Gas Mixtures by Electric Sparks III, Minimum Ignition Energies and Quenching Distances", M.V. Blanc, P.G. Guest, G. von Elbe and Bernard Lewis, 3rd Int. Symposium of Combustion (1948), p. 363

(28) "Mechanism of Chain Initiation in the Thermal Reaction Between Hydrogen and Oxygen", Bernard Lewis and G. von Elbe, 3rd Int. Symposium on Combustion, (1948) p. 484

(29) "Ignition and Flame Stabilization in Gases", Bernard Lewis and G. von Elbe, Frans. Amer. Soc. Mech. Eng. Vol. 68 (1948) p. 307.

(30) "Burning Velocity Measurements in a "Spherical Vessel with Central Ignition", John Manton, G. von Elbe and Bernard Lewis, 4th Int. Symposium on Combustion, (1952) p. 358.

(31) "Transition from Deflagration to Detonation", S. R. Brinkley and Bernard

Lewis, 7th Int. Symposium on Combustion (1959) p. 807.

(32) "Fundamental Principles of Flammability and Ignition", Bernard Lewis and G. von Elbe, Advances in Chemistry Series, Vol. 20 (1948) p. 15.

(33) "Modern Concepts of Combustion Phenomena", Plenary Lecture, Bernard Lewis, 7th World Petroleum Congress, Mexico City, April 1967, Proceedings of the Congress, p. 225.

(34) "Use of Characteristic Parameters to Describe Initiation and Stability of Combustion Waves". Bernard Lewis and G. von Elbe, Academy of Science, USSR, Semenow Aniversary Volume 1966 (in Russian).

(35) Book: "Combustion, Flames and Explosions of Gases", Bernard Lewis and G. von Elbe, Cambridge Univ. Press, 1938, 415 pp.

(36) Book: Combustion, Flames and Explosions of Gases", Bernard Lewis and G. van Elbe, Academic Press, 1951, 795 pp. Second Edition 1961, 731 pp.

(37) Editor: Three volumes of Inst. Symposium Combustion, 1948, 1952, 1954.

(38) Editor Book: High Speed Aerodynamics and Jet Propulsion, Vol. II "Combustion Processes", Bernard Lewis et al Princeton University Press 1956 (pp. 216-311).

(39) Editor Book: High Speed Aerodynamics and Jet Propulsion, Vol. IX, "Physical Measurements and Gas Dynamics and Combustion", Part II, Bernard Lewis, et al, Princeton University Press, 1954.

<u>Lectures</u>: Lectured in United States, England, Ireland, France, Germany, Italy, Mexico, Sweden, Hungary, Japan, Belgium, Canada, Russia on Combustion, Flame, Explosions, Ignition.

<u>Memberships</u>: The Combustion Institute
National Fire Protection Association
American Chemical Society
American Physical Society (On Executive

Board of Fluid Dynamics Division)
Fellow of New York Academy of Science
Fellow of American Institute of Aeronautiacs and Astronautics
Fellow of American Institute of Chemists
Review Board of Instrument Society of America, R.P.-12 Committee
Scientific Advisory Committee, Ballistic Research Laboratories, U.S. Army, 1940-1970
Formerly Chairman of Combustion Subcommittee of N.A.C.A.
Commission on High Temperatures in Gases, International Union of Pure and Applied Chemistry
Fire Research Conference of National Academy of Science
Technical Advisory Committee, U.S. Department of the Interior
Sigma Xi
Cosmos Club, Washington, D.C.
Harvard-Yale-Princeton Club, Pittsburgh, Pennsylvania

Awards:

Medal of Legion of Merit, U.S. Army, 1946
Gold Medal, The Combustion Institute, 1958
Special Citation from President Eisenhower, 1960, for contributions on interior ballistics
Gold Medal, Associazione Termotecnica Italiana, Milan, Italy, 1962
American Chemical Society, Pittsburgh Award, 1974
Medal of City of Orleans, France, 1975

BELA KARLOVITZ

Education: Mechanical Engineer, Technical University, Budapest, Hungary (1926)

Electrical Engineer, Federal Institute of Technology, Zurich, Switzerland (1928)

Experience: 1953 to Present: Member of Combustion and Explosive Research, Inc. (Combex), 1016 Oliver Building, Pittsburgh, Pennsylvania 15222. Consultant on fundamental combustion problems for government agencies, industries, and research institutes. This activity involves work on, among other things, safety of test facilities handling hydrogen, piloting of hydrogen flames and control of flame instabilities.

Originated and developed the electrically augmented flame. Designed and patented a clean burning spark ignition engine based on fundamental turbulent flame theory. Investigated numerous industrial explosions, participated in the safe design of chemical process system. Participated in the study of hydrogen explosion problems of the nuclear power industry, participated in the study of the explosion hazards of the Saturn rockets and the Space Shuttle for NASA.

1947 to 1953: Chief of Flame Research Section of the U.S. Bureau of Mines in Pittsburgh, Pennsylvania. Engaged in research on laminar and turbulent flame phenomena. Developed theory of turbulent flame propogation and originated the concept of flame stretch and theory of flame generated turbulence.

1938 to 1947: Conducted experimental research on magnetohydrodynamic power generation for Westinghouse Electric Corporation in U.S.

1934 to 1938: Originated concept magnetohydrodynamic (MHD) power generation; designed and patented this system in collaboration with Dr. Denes Halasz.

1928 to 1938: Electric Public Utility System, Budapest, Hungary - Section Engineer in technical management.

Publications: "Investigation of Turbulent Flames", Bela Karlovitz, D.W. Denniston, Jr., and F.E. Wells,

Journal of Chemical Physics, Vol. 19, No. 5, 541-547, May, 1951.

"Open Turbulent Flames", by Bela Karlovitz, Fourth Symposium (International) on Cumbustion, The Williams & Wilkins Company, Baltimore, 1953.

"Turbulent Flames" by Bela Karlovitz, D.W. Denniston, Jr., D.H. Knapschaefer and F.E. Wells, Fourth Symposium (International) on Combustion, The Williams and Wilkins Company, Baltimore, 1953.

"Effect of Flame-Generated Turbulence on Heat Transfer from Combustion Gases", by Bela Karlovitz, Conference of U.S. Bureau of Standards, Washington, D.C.

"Applications of the Electronic Probe to the Study of Turbulent Flames", by D.W. Denniston, Jr., J.R. Oxendine, D.H. Knapschaefr, D.S. Burgess, and B. Karlovitz, Journal of Applied Physics, Vol. 28, No. 1, 70-75, January, 1957.

"The Growth and Burn-out of Flame Surface in a Turbulent Stream", by B. Karlovitz, Seventh Symposium (International) on Combustion published by Butterworths Scientific Publications, London, W.C. 1959.

"Space Propulsion by Interstellar Gas", by Bela Karlovitz and B. Lewis, IXth International Astronautical Congress, Amsterdam 1958, pp. 307-311.

"Flames Augmented by Electrical Power", by B. Karlovitz, The Journal of the International Union of Pure and Applied Chemistry, Vol. 5, 1962, Butterworths, London.

"Augmented Flames", by Bela Karlovitz, International Science and Technology, June, 1962.

"Flow Phenomena and Flame Technology", by Bela Karlovitz, Chemical Engineering Progress, August, 1965.

"Electrical Augmentation of Natural Gas Flames", by C.W. Marynowski, B. Karlovitz, and T.J. Hirt, I & EC Process Design and Development, Vol. 6, page 375, July, 1967.

"Application of Electronic Probes to Measurements in Turbulent Flames" by Bela Karlovitz, Joint Conference on Combustion, The Institution of Mechanical Engines and The American Society of Mechanical Engineers, MIT, Cambridge, Mass. June, 1955.

"History of the K and H Generator and Conclusions drawn from the Experimental Results", by Bela Karlovitz and Danes Halasz, Third Symposium of the Engineering Aspects of Magnetichydrodynamics, The University of Rochester, Rochester, N.Y., March, 1962.

Chapters in Books

"A Turbulent Flame Theory Derived from Experiments" in "Selected Combustion Problems", AGARD Combustion Colloquim, Cambridge University, December 1953. Butterworths, London 1954.

"Combustion Waves in Turbulent Gases" in Volume II, "Combustion Processes of "High Speed Aerodyamics and Jet Propulsion." Princeton University Press, Princeton, New Jersey, 1956.

Holder of patents concerning industrial application of flames, electrically augmented flames, and emission control system for spark ignition engines. Invited speaker at several International Scientific Symposia in subjects including flame phenomena.

Memberships: American Physical Society; The Combusion Institute; and Registered Professional Engineer.

Awards: The 1970 Bernard Lewis Gold Metal of the Combustion Institute; Honarary President, Sixth International MHD Conference, Washington, D.C. 1975.

G. MARTIN FULS

<table>
<tr><td>Education:</td><td>B.S. in Mechanical Engineering, Carnegie Mellon University, 1956
M.S. in Mechanical Engineering, Carnegie Melton University, 1957
Ph.D. in Mechanical Engineering, University of Pittsburgh, 1958</td></tr>
<tr><td>Experience:</td><td>Over twenty years of experience in the development of analytical techniques for analyses of nuclear power plant systems. Recent assignments included the development of methods of analysis for compressible and incompressible fluid flow, heat transfer and finite elements, and development of computer programs for those analyses; the development of the CLASIX computer program and the technical support of licensing activities of TVA, Duke and AEP in relation to the post-TMI hydrogen burning analysis in ice condenser containment plants. A modification of this program, CLASIX-3 has been used for similar work for The Cleveland Electric Illuminating Co., Mississippi Power & Light Co., and Gulf States Utilities Co. on MARK III Pressure Suppression Containments.</td></tr>
</table>

1984 to Present: President, GMF Associates. Consulting work primarily related to hydrogen combustion analysis for nuclear power plants.

1976 to 1984: Advisory Engineer, Offshore Power Systems. Responsible for the development of methods of analysis for hydrogen combustion in all types of containments and compressible and incompressible fluid flow, heat transfer and finite elements for the floating nuclear plant.

1975 to 1976: Principal Engineer, Bettis Atomic Power Laboratory. Developed analysis methods, performed the analysis and wrote the final report on the safety related aspects of the moveable fuel element in the light water breeder reactor presently in operation at Shippingport, Pennsylvania.

1974 to 1975. Manager, Bettis Atomic Power Laboratory. Developed system and equipment specifications, performed operating tests and analysis on reactor plant system for an advance submarine nuclear plant propulsion system.

1972 to 1974: Supervisor, Bettis Atomic Power Laboratory. Performed analysis, test and design on advanced reactivity control system for naval core applications.

1969 to 1972: Principal Engineer, Bettis Atomic Power Laboratory. Developed a computer program to predict material circulation in a reactor plant and the effects of ships motion on the flow rates. Wrote a test procedure and supervised the test on an operating reactor to determine the accuracy of the program.

1960 to 1969: Senior Engineer, Bettis Atomic Power Laboratory.

Publications:

-Analysis of Check Valve (Westinghouse, 1962) (Computer program description)

-ACT-I: A Digital Program for the Analysis of the Containment Transient During LOCA (Westinghouse, 1967) (computer program description)

-Portions of T&H Design Manual (Atomic Energy Commission 1968 (thermal and hydraulics design manual)

-FLOT-1: Flow Transient Analysis of PWR Floor Coastdown (Westinghouse, 1968) (computer program description)

-Containment Transient (ASME, 1968) (ASME presentation)

-Bubble Dynamics and Heat Transfer (Ph.D. Thesis, 1968)

-Correlation Between FLOT-1 and Flow Coastdown Data (Westinghouse, 1969) (technical paper)

-Flow Transient Resulting From a Loss of Plumping Power in a PWR (Westinghouse, 1969) (technical paper)

-Effect of Bubble Stabilization of Heat Transfer, 1970) (technical paper)

-Containment Pressure Response to Hydrogen Combusion (NUREG/CR-2017, 1981) (technical paper)

-The CLASIX Computer Program (Offshore Power Systems, 1981)

-Grand Gulf Sensitivity Analysis (Offshore Power Systems, 1981)

-The CLASIX-3 Computer Program (Westinghouse, 1983)

-Containment Response to Hydrogen Deflagration (Offshore Power Systems, 1983) (technical paper)

Professional Affiliations: Member, American Society of Mechanical Engineers

Registered Professional Engineers, Commonwealth of Pennsylvania

Honors: Who's Who in the South and Southest

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. (Perry Nuclear Power Plant, Units 1 and 2)	)))))))	Docket Nos. 50-440 50-441

CERTIFICATE OF SERVICE

This is to certify that copies of the foregoing "Applicants' Direct Testimony of Eileen M. Buzzelli, John D. Richardson, Kevin W. Holtzclaw, Roger W. Alley, Bernard Lewis, Bela Karlovitz and G. Martin Fuls On The Preliminary Evaluation Of The Perry Nuclear Power Plant Hydrogen Control System (Issue #8)" were served by deposit in the United States Mail, first class, postage prepaid, and by Federal Express to the parties identified with one asterisk, this 1st day of April, 1985, to all those on the attached Service List.

Harry H. Glasspiegel

Harry H. Glasspiegel

DATED: April 1, 1985

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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