UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
TEXAS UTILITIES GENERATING COMPANY, et al.	)	Docket Nos. 50-445 and 50-446
	)	
(Comanche Peak Steam Electric Station, Units 1 and 2)	)	(Application for Operating Licenses)

TESTIMONY OF FRED W. MADDEN, JR.
REGARDING BOARD QUESTION ONE
RELATED TO HYDROGEN GENERATION

Q1. Please state your name, residence and educational and professional qualifications.

A1. My name is Fred W. Madden, Jr. I reside in Cleburne, Texas. A statement of my educational and professional qualifications is attached hereto as Attachment 1.

Q2. What is your current position?

A2. I am the Lead Nuclear Engineer, Technical Support Group for Texas Utilities Services, Inc. ("TUSI"). As such, one of my responsibilities is to perform engineering and technical evaluations of plant systems related to hydrogen generation and control.

Q3. What is the purpose of your testimony?

A3. The purpose of my testimony is to describe the method of handling hydrogen gas that may be generated in the CPSES containment. To facilitate understanding of this

of this matter, my testimony is divided into two sections. The first section describes the relevant hydrogen generation mechanisms at CPSES and summarizes two analyses set forth in the CPSES Final Safety Analysis Report ("FSAR") which calculate the quantity of hydrogen that the CPSES hydrogen control systems must be designed to handle. The second section decribes the systems designed to handle this amount of hydrogen. Additional discussion of this subject is set forth in Sections 6.2.5 and 6.2.5A of the CPSES FSAR (Applicants' Exhibit 3).

I. HYDROGEN GAS GENERATION

Q4. What are the methods by which hydrogen may be generated in the CPSES containment?

A4. Significant quantities of hydrogen can be generated in the CPSES containment by only four methods: (1) a zirconium-water reaction, (2) release of the free hydrogen contained in the primary coolant system, (3) radiolysis of water and (4) corrosion of susceptible construction materials in containment. FSAR §6.2.5 at p. 6.2-79 and §6.2.5A at p. 6.2-103 (Applicants' Exhibit 3).

Q5. How are these hydrogen generation mechanisms evaluated?

A5. In the FSAR, each of these hydrogen generation mechanisms is analyzed and combined using two independent methodologies to provide the total quantity and concentration of

hydrogen as a function of time necessary to be considered in the design of the combustible gas control equipment at CPSES. The two methodologies used are a Westinghouse model (discussed in FSAR §6.2.5A) and an NRC model (discussed in Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a Loss of Coolant Accident").

Q6. What are the results of those analyses?

A6. The results of the Westinghouse model analysis and the NRC model analysis are set forth in FSAR Figures 6.2.5A-6, 6.2.5A-7, 6.2.5A-8, and 6.2.5A-9 (Applicants' Exhibit 3). Based on the Westinghouse and NRC models (both assume no hydrogen control equipment), hydrogen concentrations of 8 volume percent (the concentration necessary to sustain a hydrogen deflagration throughout the containment would not be present until after approximately 100 and 75 days, respectively, had elapsed since onset of a hypothetical design basis accident ("DBA"). The two analyses, extending only to 100 days after initiation of an assumed DBA, never reach the point at which hydrogen concentrations would be in the detonable range (18-59 volume percent).

Q7. What do you mean by hydrogen deflagration?

A7. Deflagration is the propogation of a slow flame throughout a flammable mixture. In the temperature and pressure conditions relevant here, the lower deflagration limit (referred to as lower flammable limit)

of hydrogen in air is 4.0% by volume for upward propogation. Ignition of such concentrations would result in a very thin and momentary upward flame traveling to the top of containment or to some intermediate point obstructing further upward movement. There is no detectable pressure rise associated with such a deflagration. Lower deflagration limits for horizontal and downward propogation are about 6.5 and 8 volume percent, respectively.

Q8. What do these analyses assume with respect to hydrogen distribution within the containment?

A8. Hydrogen inside the CPSES containment is assumed to be uniformly distributed. This assumption is supported by the outstanding mixing characteristics of hydrogen and the configuration and systems in the CPSES containment. Specifically, hydrogen mixes readily with other gases, and once mixed will not separate in the containment environment. Mixing is promoted by convective currents created by temperature gradients in containment, containment sprays, subcompartment vents and drains, and jet-stream entrainment from the assumed break in the primary coolant system giving rise to hydrogen generation. This assumption is discussed in FSAR §6.2.5.3.2 (Applicants' Exhibit 3).

Q9. Please describe the generation of hydrogen by a zirconium-water reaction.

A9. The production of hydrogen by the reaction of water and the zirconium cladding around the fuel is described by the following exothermic chemical equation:

$$Zr + 2H_2O \rightarrow ZrO_2 + 2H_2 + \text{Heat}$$

This reaction, however, proceeds in significant quantities only in the presence of very high temperatures. Such temperature can only be achieved during a hypothetical loss of coolant accident coupled with loss of emergency cooling water from the emergency core cooling system ("ECCS"). In such a situation, the core may be exposed and excessively high temperatures may be present. This reaction is described in more detail in FSAR §§6.2.5.3.1, 6.2.5A.1 and 6.2.5A.2.1 (Applicants' Exhibit 3).

Q10. What limits are imposed on ECCS design with respect to the zirconium-water reaction?

A10. The ECCS, a safety grade system with redundant trains and power sources, is designed to assure compliance with NRC regulations limiting zirconium-water reaction following a DBA to that associated with the reaction of 1% by weight of the total quantity of zirconium in the core. 10 CFR §50.46(b)(3). This is also discussed in FSAR §6.2.5A.1 (Applicants' Exhibit 3). ECCS calculations, however, have shown that in the event of a DBA less than 0.3% of the zirconium will react. For the hydrogen generation analyses the Westinghouse and NRC models conservatively assume a 2% and 5%, respectively, zirconium reaction.

Q11. What changes have been made in ECCS operation since the Three Mile Island accident?

A11. During the Three Mile Island accident a loss of coolant accident followed by operator interference with the ECCS resulted in an exposed core and excessive hydrogen production due to a zirconium-water reaction. Subsequent to this accident Commission directives required the development of procedures to assure that such premature operator interference with ECCS operation will not occur. To comply, procedures at CPSES will require that in the event of an ECCS initiation, operators will not terminate ECCS operation absent positive indications that the core is completely covered. Core subcooling monitors will be installed to augment existing equipment and procedures, thus providing such positive indications. This is discussed in FSAR Volume XIV, §II.F.2, Response to the NRC Action Plan Developed as a Result of the TMI-2 Accident. In addition, operators receive significant class room and simulator training in this area. This training is discussed in the same section of the FSAR at §§II.A.2, II.B.4 and II.F.2 (Applicants' Exhibit 3).

Q12. Please describe the generation of hydrogen by the release of free hydrogen in the primary coolant system.

A12. The hydrogen generation analyses set forth in the FSAR assume that the maximum equilibrium quantity of hydrogen in the reactor coolant system during normal operations is immediately released into containment following a LOCA. Such quantities include hydrogen dissolved in the primary coolant and hydrogen trapped in the pressurizer gas space. This reaction is described in FSAR §§ 6.2.5.3.1, 6.2.5A.1, and 6.2.5A.2.2 (Applicants' Exhibit 3).

Q13. Please describe the generation of hydrogen by water radiolysis.

A13. Water radiolysis is a complex process in which water, in the presence of radiation, is broken down into hydrogen and oxygen in accordance with the following equation.

$$H_2O \rightleftarrows H_2 + 1/2\ O_2$$

The FSAR analyses consider the only two major sources of water for radiolysis that would be present following a DBA, i.e., the reactor coolant inventory in the reactor coolant system and the reactor containment sump water. Significantly, the radiolysis process is relatively slow, and is retarded by increasing concentrations of hydrogen which force a reverse reaction (i.e., combining hydrogen and oxygen to produce water). While the Westinghouse model takes credit for reduced yield of hydrogen due to such reverse reactions, the NRC model does not. This reaction is described in FSAR §§6.2.5.3.1, 6.2.5A.1, 6.2.5A.2.4, and 6.2.5A.3 (Applicants' Exhibit 3).

Q14. Please describe the generation of hydrogen by corrosion of susceptible construction materials.

A14. Oxidation of metals in aqueous solutions results in the generation of hydrogen gas as one of the corrosion products. Extensive corrosion testing has been conducted to determine the behavior of the various metals used in the containment during accident conditions. Metals tested include zircaloy, inconel, aluminum alloys, cupronickel alloys, carbon steel, galvanized carbon steel, and copper. The results of the corrosion tests have shown that only aluminum and zinc will corrode at a rate that will significantly add to the hydrogen accumulation in the containment atmosphere.

The corrosion of aluminum and zinc is described by the following two equations:

$$2\ Al + 3\ H_2O \longrightarrow Al_2O_3 + 3\ H_2$$

$$Zn + 2H_2O \rightleftarrows Zn(OH)_2 + H_2$$

Based on the corrosion rates and the aluminum and zinc inventory in the containment, the contribution of aluminum and zinc corrosion to hydrogen accumulation in the containment following the design basis accident was calculated and factored into the FSAR hydrogen generation analyses. To be conservative, no credit was taken for protective shielding effects of insulation or enclosures from the spray, and complete and continuous immersion was assumed. This reaction is described in FSAR §§6.2.5.3.1, 6.2.5A.1, and 6.2.5A.2.3 (Applicants' Exhibit 3).

II. HYDROGEN GAS CONTROL

Q15. What measures are taken at Comanche Peak to handle the amounts of hydroge postulated to be generated?

A15. To safely handle the amount of hydrogen assumed to be generated by the four above referenced methods, redundant, electrical hydrogen recombiners and a backup hydrogen purge system are provided in accordance with NRC Regulatory Guides 1.7, 1.22, 1.26, and 1.29; General Design Criteria 41, 42, 43, and 50; and Branch Technical Positions CSB 6-2 and APCSB 9.2. This system is described in FSAR §6.2.5 at p. 6.2-79 (Applicants' Exhibit 3).

Q16. Please describe the design and operation of the electric hydrogen recombiners.

A16. Two redundant, electric hydrogen recombiners are provided in containment as the primary hydrogen control system. Each recombiner has sufficient capacity to assure that containment hydrogen concentration levels do not exceed 4 volume percent based on the conservative hydrogen release model set forth in Regulatory Guide 1.7. The recombiners are safety related and designed to sustain all normal loads as well as accident loads including a safe shutdown earthquake (SSE) and pressure-temperature transients from a design basis LOCA. Each recombiner is powered from a separate safeguards bus. There is no interdependency between this system and the other engineered safety features systems. In operation,

hydrogen is removed from the containment atmosphere by heating in the recombiner to a temperature sufficient to cause recombination of hydrogen with the containment oxygen. The operation of the recombiners is discussed in FSAR §§6.2.5.1.2, 6.2.5.3.3 and 6.2.5.4.1 (Applicants' Exhibit 3).

FSAR Figure 6.2.5-3 (Applicants' Exhibit 3) illustrates containment hydrogen concentration as a function of time assuming operation of one recombiner started 24 hours after initiation of a DBA. The Figure shows that even for the conservative NRC model, hydrogen concentration does not exceed approximately 2 volume percent, far below even the lower flammable limit for upward flame propogation.

Q17. Please describe the design and operation of the hydrogen purge system for Comanche Peak.

A17. The hydrogen purge system, serving both CPSES containments, functions as a backup for the electric hydrogen recombiners. Like the recombiners, the purge system has the process capacity to maintain hydrogen concentration in the containment below 4 volume percent based on the conservative hydrogen generation model set forth in Regulatory Guide 1.7. The hydrogen purge system for each containment consists of two 700 standard cubic feet per minute ("scfm") blowers

for air supply, isolation valves, two atmospheric cleanup systems, and two exhaust fans. The blowers are capable of transporting 700 scfm of fresh, filtered air to the containment. The exhaust fan draws air from either containment, as required, and passes the air through high efficiency particulate and iodine filters before discharge through the plant discharge duct at levels that assure compliance with 10 CFR Part 100 guideline values. Two trains are provided for each containment, each capable of exhausting the design airflow of 700 scfm. The system components are designed for SSE loads and maximum temperature and pressure transients from a DBA.

Q18. What is your conclusion regarding the design of the hydrogen control systems at Comanche Peak?

Q18. The design of the CPSES hydrogen control systems provide a high level of assurance that in the event of an accident leading to hydrogen generation, levels of hydrogen gas in the containments will be maintained below 4 volume percent, in accordance with the conservative hydrogen generation model set forth in Regulatory Guide 1.7.

FRED W. MADDEN

STATEMENT OF EDUCATIONAL AND PROFESSIONAL QUALIFICATIONS

POSITION:	Lead Nuclear Engineer, Technical Support
FORMAL EDUCATION:	1968-1972, B.S. Engineering Physics, Texas Tech University
	1972-1974, M.S. Nuclear Engineering, Purdue University
EXPERIENCE:	
1981 - Present	Texas Utilities Services, Inc., Comanche Peak Steam Electric Station, Glen Rose, Texas, Lead Nuclear Engineer, Technical Support Group. Activities include design and engineering of TMI-related plant modifications; engineering resolution of licensing issues; and development of analytical capabilities.
1980 - 1981	Texas Utilities Services Inc., Dallas, Texas, Licensing Engineer. Activities included preparation of licensing information such as FSAR, responses to NRC questions, and interrogatories; and review and interpretation of regulatory criteria.
1976 - 1980	Brown & Root, Inc., Houston, Texas, Senior Licensing Engineer. Activities included preparation and coordination of licensing information such as SAR's, environmental reports and NRC questions; review and interpretation of regulatory criteria. Coordinator of project design review team following TMI accident.
1974 - 1976	Bechtel Power Corporation, Los Angeles, California, Engineer on Nuclear Analysis staff. Activities include accident analysis calculations; nuclear fuel cycle analyses; radiation dose calculations; and shielding design and analysis. Other project activities include system design; preparation of specifications and bid evaluation.
PROFESSIONAL:	Registered Professional Engineer (Texas and California), American Nuclear Society, Tau Beta Pi, Phi Kappa Phi, Sigma Pi Sigma.

BOB C. SCOTT

STATEMENT OF EDUCATIONAL
AND PROFESSIONAL QUALIFICATIONS

POSITION: Principle Quality Assurance Specialist

FORMAL EDUCATION: 1961-1962, Management, University of Houston
1970-1971, Management, San Jacinto College

EXPERIENCE:

1981 - Present — Ebasco Services Incorporated, Principle Quality Assurance Specialist assigned to the Comanche Peak Steam Electric Station. Responsibilities include supervision of Quality Engineering personnel; review, comment and/or approval of quality Procedures/Instructions; control and distribution of Quality Procedures and Instructions for Non-ASME functions; basic indoctrination and required technical training for Quality Control personnel; compliance of site procurement activities to established CPSES QA requirements; and, review, processing and tracking of nonconformance reports.

1979 - 1981 — General Dynamics Corporation as a Senior Quality Assurance Engineer. Assigned to the Procurement Quality Assurance Department to supervise and instruct Quality Assurance/Control Engineering Staff.

1977 -1979 — Brown & Root, Inc. as Site Quality Assurance Manager for CPSES. Responsible for establishing, implementing and assuring compliance of the CPSES Quality Assurance/Control Programs; managed an organization of quality engineers/technicians/inspectors responsible for identifying quality problems, recommending or providing solutions and verifying implementation of solutions; and responsible for the control, processing, delivery and installation of nonconforming items or unsatisfactory conditions until acceptable dispositions were accomplished.

1976 - 1977 — Brown & Root, Inc. as a Quality Assurance Supervisor for CPSES. Responsible for Records Control, Receiving Inspection, Nonconformance/Corrective Action, Training, Procurement, System Turnover Verification, and Calibration; developed

a Quality Assurance Instruction Program to train personnel in Quality requirements and practices. Appointed as Quality Instruction/Consultant by the Corporate Personnel Training and Development Department to administer quality training sessions to Construction and Inspection personnel.

1974 - 1975	Brown & Root, Inc. as a Quality Assurance Audit Section Manager. Established and implemented a comprehensive Quality Assurance Internal/External Audit Program for three (3) nuclear power plant projects. This program included planning, scheduling, staffing, personnel training and certification, procedural preparation and audit techniques for all phases of power plant construction, including design, procurement and component installation.
1969 - 1974	ILC Industries as a Quality/Reliability Supervisor and a Quality/Reliability Foreman.
1968 - 1969	E. I. DuPont Company as an Assistant Manufacturing Engineer.
1963 - 1968	Goodyear Tire & Rubber Company as a Manufacturing Technician.
1961 - 1963	Southwestern Pipe Inc. as a Quality Control Inspector.

PROFESSIONAL AFFILIATIONS:

American Society of Quality Control

American Management Society

DAVID H. WADE

STATEMENT OF EDUCATIONAL
AND PROFESSIONAL QUALIFICATIONS

POSITION:	Senior Licensing Engineer
FORMAL EDUCATION:	1971, BS Mechanical Engineering, University of Texas, Arlington
REGISTRATION:	Professional Engineer, State of Texas #47622
EXPERIENCE:	
1982 - Present	Texas Utilities Services Inc. as Senior Licensing Engineer
1981 - 1982	Texas Utilities Services Inc. as Project Mechanical Engineering Department Head. Supervised Comanche Peak Mechanical Engineering efforts.
1980 - 1981	Texas Utilities Services Inc. as Area Supervisor. Supervised Mechanical Field Engineering Activities at Comanche Peak.
1978 - 1980	Texas Utilities Services Inc. as Design Engineering Supervisor for the Field Support Design Group at Comanche Peak. Responsible for resolutions of field interference problems.
1975 - 1978	Texas Utilities Services Inc. as Comanche Peak Mechanical Engineer. Responsible for specification and procurement of piping, valves, supports and in-line components.
1973 - 1975	Dallas Power and Light Company as Associate Engineer in the Engineering Dept. Responsible for design and engineering of power plant systems and modifications to existing facilities.
1972 - 1973	Dallas Power and Light Company as Junior Engineer in the Plant Dept. Responsible for plant start-up, testing, maintenance and technical assistance to operations.

Educational and Professional Qualifications

MICHAEL J. HITCHLER
WESTINGHOUSE ELECTRIC CORPORATION

Carnegie Mellon University, M.S. Mechanical Engineering 1977

Lowell Technological Institute,

B.S. Mechanical/Nuclear Engineering 1974

Mr. Hitchler is the Manager of the Probabilistic Risk Assessment Group at Westinghouse Nuclear Center. He has current lead responsibility for a probabilistic risk study of the Sizewell B (British National Nuclear Corporation) Nuclear Station, which includes development of a risk baseline and an assessment of potential design alternatives. He has recently worked on the Zion and Indian Points risk studies, contributing extensively in the following areas: plant and containment event tree construction, systems success criteria for fault tree development, external (seismic, wind, fire, etc.) event analysis and review of the ramification sections.

Previously, Mr. Hitchler was involved in the development and implementation of strategic programs to enhance and apply risk assessment technology for use in nuclear power plant design and licensing. This included development, quantification and NRC defense of event trees for use in reviewing emergency and abnormal operating procedures as part of the Westinghouse Owners' Group response to Post TMI issues. He has assisted in the development and review of Auxiliary Feedwater System Reliability Studies for three nuclear plants. Prior to this, his responsibilities included performing accident analyses for accidents used in licensing documents. He has served as a Westinghouse interface with the NRC, architect engineers and utilities for issues concerning reactor protection system design requirements. Specific areas of specialization include core and systems response to transients initiated in the primary system, development of methodology for safety analysis of reload cores, and simulation of actual plant transients for computer verification purposes. Included was the lead responsibility for the transfer of the above technology to various utility customers. This included the structuring of classroom as well as on-the-job training for a number of utility personnel.

Mr. Hitchler is a member of the American Nuclear Society and the American Society of Mechanical Engineers. He has served on two ANS Standards committees and contributed to several AIF

and IEEE committees on Development of Risk Criteria and Utilization of PRA Approach to Licensing. He is currently a member of the PRA Methodology Procedures Handbook Committee. He is author or co-author of three reference articles, several papers and numerous project reports.

Educational and Professional Qualifications

KENNETH RUBIN
WESTINGHOUSE ELECTRIC CORPORATION

My name is Kenneth Rubin. My business address is Westinghouse Electric Corporation, P.O. Box 355, Pittsburgh, Pennsylvania 15230. I am employed by Westinghouse as an engineer in the Mechanical Equipment and Systems Licensing group, within the Nuclear Safety Department of the Nuclear Technology Division.

I am currently attending the University of Pittsburgh and will shortly receive a Bachelor of Science Degree in Applied Mathematics.

My current responsibilities include development and implementation of improved radiological consequence analysis codes, evaluation of containment spray systems for fission product removal capability, corrosive effects on materials of construction, and general post-accident hydrogen production.

I have performed numerous accident analyses for use in safety analysis reports and in support of operating plants and have served as a Westinghouse interface with the NRC, utilities and A/E's for issues concerning post-accident radioactivity releases and hydrogen production and accumulation. I have provided technical support to the Westinghouse Equipment Qualification Program regarding containment spray chemical environment. I hold a U.S. patent in the area of containment spray system testing.