July 20, 1981

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	§	
	§	
HOUSTON LIGHTING & POWER COMPANY	§	Docket No. 50-466
	§	
(Allens Creek Nuclear Generating Station, Unit 1)	§	
	§	

DIRECT TESTIMONY ON BEHALF OF
HOUSTON LIGHTING & POWER COMPANY:

(1) JAMES C. ELLIOTT - TEXPIRG ADDITIONAL CONTENTION 53/NON-CONDENSIBLE GASES

(2) ROBERT C. CHENG - BOARD QUESTION 4B/ CONTAINMENT DESIGN BASIS

(3) MELVYN D. WEINGART AND CLOIN G. ROBERTSON - BOARD QUESTION 4A/COMBUSTIBLE GAS CONTROL

(4) PATRICIA A. RANZAU - TEXPIRG ADDITIONAL CONTENTION 28/POST ACCIDENT MONITORING

(5) CLOIN G. ROBERTSON - TEXPIRG ADDITIONAL CONTENTION 52/OUTSIDE CONTAINMENT SAMPLING

Q. Would each of you please state your name, your current position and describe your educational and professional experience?

A. My name is James C. Elliott and I am employed by the General Electric Company (GE) as Principal Engineer in the GE Nuclear Power System Engineering Department. My educational and professional background is described in

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Attachment JCE-1.

My name is Robert C. Cheng. I have previously explained my position and background in connection with my testimony on Board Question 10.

My name is Melvyn D. Weingart and I have previously described my position and background in connection with my testimony on TexPirg Additional Contention 30.

My name is Cloin G. Robertson and I have previously described my position and background in connection with my testimony on Doherty Contention 8.

My name is Patricia A. Ranzau and I am employed by Houston Lighting & Power Company (HL&P) as Lead Engineer, Power Plant Engineering Department, Instrument & Controls. A description of my educational and professional qualifications is described in Attachment PAR-1.

Q. Mr. Elliott, what is the purpose of your testimony?

A. The purpose of my testimony is to address TexPirg AC 53 which states:

> "Applicant should commit to a system to ascertain accurately how much non-condensible gas is in the reactor vessel, to assist in estimating the possible explosion hazard in the vessel during an ECCS. The need for this information was demonstrated at Three Mile Island, Unit 2, during its recent incident. Petitioner contends that inability to know accurately the amount of non-condensible gas in the reactor increases the chance of an explosion and damage to the fuel geometry and/or physical breaking of fuel rod clad."

In the deposition of Mr. Clarence Johnson taken on February 27, 1980, he states that TexPirg's concern is about ACNGS's ability to monitor hydrogen during a small break LOCA. From this it is inferred that when the contention says non-condensible gases, that the chief concern is hydrogen and that "during an ECCS" really means "during a small break LOCA".

Q. Does the Applicant have a system which monitors the amount of non-condensible gas in the ACNGS reactor vessel?

A. The Applicant has the capability of monitoring the potential for formation of non-condensible gases in the ACNGS reactor by using the reactor water level indicators. It is a necessary condition that the reactor core become uncovered before any large scale metal-water reaction can occur to produce hydrogen. The reactor water level instrument at ACNGS gives a direct reading of the reactor water level and thus would give the reactor operator the information needed in order to take appropriate actions. These actions are described in the operator emergency procedure guidelines, wherein the operator is directed to depressurize the reactor using the safety/relief valves if an automatic depressurization has not occurred, and if the operator is unable to restore reactor water level with the high pressure pumps (feedwater, RCIC, HPCS, CRD). This action delays core heatup by inducing

steaming cooling, and allows low pressure pumps (4 condensate pumps, 3 RHR pumps, 1 LPCS pump) to begin injecting water into the reactor. In the event water level is not restored, any hydrogen generated would be vented immediately to the suppression pool. At TMI, neither reactor water level information nor the ability to rapidly depressurize and vent the reactor vessel, was available to the operator.

Q. Is it important to know the amount of hydrogen in the vessel in order to avoid "explosion hazards" in the reactor?

A. No. In general, all appropriate actions can be taken based on reactor water level. Additionally, knowing the exact amount of hydrogen is not necessary since there will be insufficient oxygen in the reactor vessel to support a hydrogen combustion. This conclusion is consistent with the final analyses of the NRC and other experts relative to the TMI accident. For a small break LOCA the primary source of oxygen would be due to radiolysis. Radiolysis is the decomposition of water into hydrogen and oxygen under the influence of high energy radiation. In the presence of an excess of hydrogen, the recombination reaction is faster than e decomposition reaction, so no net production of hydrogen or oxygen occurs. Thus the free oxygen concentration would not increase due to radiolysis, nor

would it reach the flammability limit of 4% nor the detonation limit of 9%[1]. Also, it must be kept in mind that hydrogen can be vented at any time during operation of the plant.

Q. Please describe how the Allens Creek reactor vessel provides for venting of non-condensible gases?

A. A discussion of reactor vessel vent paths is provided in Appendix O of the ACNGS PSAR in response to Item II.B.1 of NUREG-0718, "Licensing Requirements for Pending Applications for Construction Permits and Manufacturing License". Venting capability of the ACNGS reactor vessel is addressed in two parts:

1. Up to the main steam line nozzles: Venting this region can be accomplished by opening any one of the 19 safety relief valves on the main steam lines, some or all of which may already be open depending on the mode of core shutdown cooling in use. These valves and their operators are safety grade, seismically and environmentally qualified for accident conditions, have position indication in the control room (see testimony on Doherty Contention 42), and are powered from the onsite electrical system and operable from the

1/ NSAC-80-1, March 1980 - Analysis of TMI-2 Accident.

control room. Eight of the valves, which are part of the ADS, have redundant safety related air and electrical supplies. In addition, this region of the vessel can be vented through the RCIC steam supply line which connects to main steam line A, without opening the SRV's. This path is through the RCIC steam turbine exhaust, which discharges to the suppression pool.

2. Above the main steam nozzles: There are two means of venting this space:

a. Venting this area can be accomplished via the normally open reactor head vent line and valve B21-F005, which discharges to main steam line A. The non-condensible gas can then be vented to the suppression pool/containment via any one of three safety relief valves; two of which have safety related air supplies.

b. Venting this area can also be accomplished via the normally closed reactor head vent line and series valves B21-F001 and B21-F002, which discharge to the drywell high purity drain tank. This vent path is normally only used during shutdown and startup. These valves are safety grade and their operators are Class IE, and are seismically and environmentally qualified. They are operable from

the Main Control Room.

All of the above venting paths lead to the containment via the suppression pool. The control of hydrogen in the containment is discussed in the response to Board Question #4A.

Q. Mr. Cheng what is the purpose of your testimony?

A. The purpose of my testimony is to address Board Question 4B wherein the ard requested HL&P to present evidence that Allens Creek will meet current NRC requirements with respect to GDC-50, Containment Design Basis and to evaluate small pipe break consequences in the ECCS. Small pipe breaks in the ECCS were discussed in HL&P testimony by Mr. Pappone on Board Question 4B, on June 1, 1981. Mr. Weingart and Mr. Robertson will address Board Question 4A which deals with the question of whether ACNGS will meet the NRC requirements for combustible gas control.

Q. With respect to Board Question 4B, what does General Design Criterion (GDC) 50 require?

A. In general, the current revision to GDC-50 (dated October 27, 1978), requires the following:

CRITERION 50-Containment Design Basis

The reactor containment structure and associated support systems shall be designed so that the containment structure can accomodate without exceeding the design leakage rate, the

calculated pressure and temperature conditions resulting from any loss-of-coolant accident. The design margin shall reflect consideration of (1) the effects of potential energy sources that have not been included in the determination of the peak conditions such as required by 50.44 (energy from metal-water) and other chemical reactions that may result from degraded emergency core cooling functioning; (2) the limited experience and data available and (3) the conservatisms in the calculations.

Q. Does the Allens Creek PSAR make a commitment to meet General Design Criterion 50?

A. Yes. The Allens Creek PSAR Section 3.1.2.5.1.1 commits to meet the requirements of General Design Criterion 50. The design temperatures and pressures set forth in PSAR Section 3.8.2 exceed the pressures and temperatures associated with accidents described and evaluated in Chapters 6 and 15 of the PSAR, including the requirements of 10 CFR 50.44.

Q. What are your conclusions concerning Board Question 4B, compliance with GDC-50, Containment Design Basis?

A. The ACNGS design addresses and complies with the requirements of GDC-50.

Q. Mr. Weingart, with respect to Board Question 4A, what regulation governs the current requirements for combustible gas control?

A. The requirements for combustible gas control are found in 10 CFR §50.44.

Q. What are the requirements of 10 CFR §50.44 and how are these requirements met for Allens Creek?

A. In general, the requirements of 10 CFR §50.44 are the following:

1. A means shall be included in the plant design, for the control of hydrogen gas generated after a LOCA due to the metal-water reaction of the fuel cladding and coolant, the radiolytic decomposition of coolant, and the corrosion of metal.

2. The plant design shall include, following a LOCA, a means to measure hydrogen concentration in the containment, a means to insure a mixed containment atmosphere, and a method of controlling combustible gas concentration.

3. Evaluations shall be performed to verify that following a LOCA but prior to operation of the combustible gas control system that an uncontrolled hydrogen-oxygen recombination would not take place or the plant could withstand the consequences of uncontrolled hydrogen-oxygen recombination.

4. The amount of hydrogen contributed by the metal-water reaction resulting from a degraded emergency core cooling function shall be assumed to be either five times the total amount calculated in 10 CFR §50.46 (b) (3) or the amount resulting from a reaction of all the metal in the outside surfaces of the cladding cylinders surrounding the fuel (excluding cladding around the plenum volume) to a depth of 0.00023 inches, whichever is greater.

5. A combustible gas control system, such as recombiners, shall be required. A purging system shall not be the primary combustible gas control means.

Q. Does the ACNGS design comply with 10 CFR §50.44?

A. Yes. PSAR Section 6.2.5.1 commits Allens Creek to

meet the requirements of Regulatory Guide 1.7, Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant-Accident. This regulatory guide provides requirements which are to be followed in the calculations, analyses and design features pertaining to post-LOCA hydrogen generation.

As described in PSAR Sections 6.2.5.2.2 and 7.5.1.4.2.11 (d), ACNGS will have the capability to sample and measure the hydrogen concentration throughout the drywell and containment during an accident. This system consists of redundant sample and return lines, isolation valves, hydrogen analyzers, sample cylinders and pumps. The hydrogen volume percent is recorded and alarmed in the control room.

A drywell mixing system is described in PSAR Section 6.2.5.2.3. PSAR Sections 6.2.5.3.3.3 and 6.2.5.3.3.4 describe the turbulent convective mixing process of hydrogen that occurs following a LOCA. These two methods of mixing ensure there is an adequately mixed atmosphere in the containment and drywell following a postulated LOCA. Combustible gas concentrations in containment following a LOCA are controlled with an electric hydrogen recombiner system. This system as described in PSAR Sections 6.2.5.2.1, 6.2.5.2.4 and 6.2.5.2.5, operates on the principle of natural convection of warm air rising through the recombiner. As a

secondary or back-up system to the hydrogen recombiners, a hydrogen purge system consisting of fans, ductwork and filters, is available in the ACNGS design as an additional containment hydrogen control means.

As required by Regulatory Guide 1.7, a 4 percent or lower by volume hydrogen in air or steam-air atmosphere shall be maintained post-LOCA in containment. The hydrogen recombiner system is designed to maintain the containment below this 4 percent lower flammability limit.

As described in PSAR Section 6.2.5.3, design evaluations have been performed by GE to determine the quantity of hydrogen released post-LOCA due to the metal-water reaction.

Q. Mr. Robertson, the question presented in Board Question 4A was raised prior to the TMI incident. Are there any requirements pertaining to combustible gas control that have evolved as a result of the TMI incident?

A. Yes. As NRC guidance for compliance with 10 CFR §50.34 (e), a proposed rulemaking, NUREG-0718, Item II.B.8, parts (3), (4)b and (4)c require that a hydrogen control system be provided to control the amount of hydrogen generated by a 100 percent active fuel-clad metal-water reaction that has been postulated to occur from a degraded core accident.

Q. Has HL&P taken steps to assure that this guidance

has been incorporated into the ACNGS design?

A. HL&P has committed to provide a hydrogen control system in the ACNGS design. Currently a number of different methods are being considered throughout the industry and it is expected that these efforts will produce valuable data upon which to select an optimum means of hydrogen control. Further, it is expected that the pending rulemaking on degraded cores will determine the necessity for such a system. For the purposes of meeting the stated requirement, a post-accident inerting system using CO_2 as an inerting agent is being designed. However, for the reasons given above the need for this system will be under continuing review.

Q. Mr. Robertson, while we are on the subject of TMI, TexPirg Additional Contention 52 alleges that because of the TMI incident ACNGS should be designed so that reactor coolant samples can be taken from outside the containment. Does the Allens Creek design include the capability to sample reactor coolant from outside containment?

A. Yes. As indicated in Appendix O, page O-79 to the PSAR, Allens Creek will have the capability to collect liquid samples from the reactor coolant and suppression pool from outside the containment building. This commitment was made in response to Item II.B.3 of NUREG-0718, "Licensing Requirement for Pending Applications for Construction Permits

and Manufacturing License," which requires us to be able to collect reactor coolant samples assuming extensive fuel damage and contamination of the containment building.

Q. Would you please briefly describe the sample system to be used?

A. Final design for the post-accident sampling system is not required until the OL stage but as it is now planned the sampling system will be a manual system that can be operated to collect a liquid sample. The sampling station will be located in the Reactor Auxiliary Building near the Reactor Shield Building wall, in order to make the sample lines as short as possible. Shielding will be provided to ensure that doses to those personnel collecting the samples are maintained within GDC 19 as specified in NUREG-0737.

Q. Ms. Ranzau, what is the purpose of your testimony?

A. The purpose of my testimony is to address another TMI related contention. I will address TexPirg AC 28 contention which alleges that:

> "The control room design and the post-accident display instrumentation for Allens Creek plant are not sufficient to ensure that the operators can safely control the plant under all accident conditions. As at Three Mile Island, the operator may make one or more critical mistakes because of defective instruments or their location in the control room."

Q. Has the layout of the control room for Allens Creek been designed to consider the "location" of instruments?

A. Yes. The location of instrumentation was an explicit part of the design evaluation process for the main control room boards. As described in the post-TMI requirements, Appendix O of the ACNGS PSAR, Item I.D.1, the control room utilizes human factors principles in the design. ACNGS will use the General Electric Company NUCLENET/1000 Control Complex. NUCLENET is a generic design utilizing CRT based displays and is the most advanced state-of-the-art design. The design was based on a methodology virtually identical to that set out in Appendix B to NUREG-0659, which reflects current NRC recommendations for control room design. The design process included an analysis of all functions necessary to operate the plant safely, an allocation of functions between operator and machine, and a qualitative verification of the functional allocation. GE assembled a team of experts to design the NUCLENET/1000 Control Complex which included experts in controls and control systems design, computer technology, industrial design, operator training, power plant test and operations, and behavioral science. The premise upon which the design is based is that optimum control is achieved when there is an allocation of control functions between the operator and machine which recognize that each performs certain functions better than the other, and that, once the allocation is made, the design permits efficient and effective manipulation of controls

by the operator. The balance of plant controls were also integrated into the NUCLENET complex on the basis of guidelines developed by GE.

Q. Has HL&P reviewed the ACNGS design to assure compliance with current regulatory guidelines?

A. Yes. A design review was conducted of a full-scale mockup of the control room with an interdisciplinary team of instrumentation and control engineers from HL&P, Ebasco and Brown & Root. We also had a human factors consultant from MIT and two plant operating personnel from HL&P.

Q. Could you explain how this design review was performed?

A. As provided in Appendix O, Item I.D.1, of the ACNGS PSAR, the Allens Creek Control Room Evaluation Task Team performed a preliminary assessment of the ACNGS Unit 1 control room for the purpose of identifying additional human factors engineering design conditions that could provide a basis for further improvements. HL&P's instrumentation and controls engineers assigned to Allens Creek built a full size mockup of the front row panels of the NUCLENET/1000 Control Complex. A human factors engineering evaluation was performed on the mockup. The control room was evaluated as is, without regard to planned changes in layout and changes required as a result of Three Mile Island. The evaluation consisted of

the application of human factors engineering design checklists. The checklists were developed and prepared by the General Electric Boiling Water Reactor (BWR) Control Room Owners Group following the TMI incident.

Q. What were the conclusions drawn from this preliminary review?

A. The Allens Creek control room uses advanced technology which incorporates human factors engineering principles in design and operation. A control room design review has been conducted based on a full-scale mockup, deficiencies identified, and enhancement activities are being undertaken. Completion of the enhancements will result in a control room which meets the intent of NUREG/CR-1580 and NUREG-0659. Results of these actions will be forwarded to NRC upon completion.

Q. Have any specific design changes been made in the control room to assist the operator in responding to accident conditions?

A. We have added mimics, additional lines of demarcation, uniformity of abbreviations and nameplate locations to the control room panels. We have added an additional system called the Safety Parameter Display System (SPDS). The SPDS was added in direct response to NUREG-0718, Item I.D.2.

Q. Could you briefly describe the SPDS?

A. The SPDS is described in detail in ACNGS PSAR Section 7.5.1.6. The system will have the capability of displaying the full range of important plant parameters and data trends on demand. The system will also indicate when plant parameters are approaching or exceeding process limits. The SPDS will be designed in conformance with the guidance of NUREG-0696, February 1981, Section 5. The SPDS will be a computer-based system of high quality and reliability. It will be capable of functioning properly in the environments that are present during transient and accident conditions.

Q. Now that we have discussed how the control room and SPDS have been designed to include not only post-TMI requirements, but specifically human factors, how do these two main safety parameter areas meet current NRC requirements for instrumentation reliability?

A. The current NRC requirements with regard to safety-related instrumentation reliability are described in Regulatory Guide 1.97, Revision 2. This regulatory guide provides the current post-TMI instrumentation requirements. Conformance with this regulatory guide is discussed in the ACNGS PSAR Appendix C and also in Appendix O, Item II.F.3. As discussed in Appendix C of the ACNGS PSAR, safety-related instrumentation used in the ACNGS design will consist of reliable, state-of-the-art devices.

James C. Elliott

I graduated from the University of Idaho in 1960 with a Bachelor of Science Degree in Mechanical Engineering. I received a Master of Science in Mechanical Engineering in 1966 from Stanford University. I am a professional engineer in the State of California.

I served in the U.S. Navy from June 1960 to September 1963. From November 1963 to September 1965 I was employed by the General Electric Company as a Turbine Generator Startup Engineer. In this capacity I provided technical direction and supervision for installation, maintenance, and repair of steam turbine-generators, stationary gas turbines, and marine propulsion turbines. From September 1966 to June 1968 I was employed by GE as a Warranty Service Engineer. In this capacity I administered nuclear power plant, nuclear fuel, and consulting contracts with utilities having operating GE nuclear reactors. From June 1968 to November 1972 I was employed by GE as a plant test engineer. During this period I first worked for five months as an instructor at the BWR Training Center in Morris, Illinois. Following that, I worked as the Simulator Specialist at the BWR Training Center. Next, I was Shift Supervisor for the startup of the KKM Nuclear Power Plant in Switzerland. From November 1972 to March 1976 I was employed by GE as a Senior Project Engineer. During this time I was the Resident Project Engineer in Birmingham, Alabama on the Barton Nuclear Power Plant project. Since March 1976, I have worked in various engineering capacities in the GE Nuclear Power Systems Engineering Department. Initially I was responsible for developing and communicating recommendations to architect-engineers and utilities pertaining to BWR shutdown oxygen control and feedwater quality. In this position I developed a system for oxygen control in BWR's during shutdown, and co-authored several publications on BWR shutdown oxygen control and feedwater quality. In my current capacity, I am responsible for management of GE's "Post-TMI Hydrogen Control" program. I assisted in the development of a "Post Accident Inerting" hydrogen control system for Mark III containments.

Patricia A. Ranzau

I received a BS in Math from Texas A&M University in 1972. I received a BS in Industrial Engineering from Texas A&M in 1973.

I have been employed by HL&P since September 1973. I have worked entirely as an engineer in the Instrument and Controls group of the Power Plant Engineering Department. I have had various responsibilities within that group and I am now Lead Engineer in the group. I also serve as I&C Team Leader for Allens Creek with responsibility for design of all nuclear and balance of plant instrumentation and control.

In addition to my normal duties I have participated in several workshops and courses in control room design with particular emphasis on human factors design. I also participate in regular reviews of NRC and industry regulations and guidelines.