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) STEVEN A. HUCIK AND JOHN J. BOSEMAN ON DOHERTY CONTENTION 17 - SRV RELIABILITY

(2) ROBERT L. HUANG ON TEXPIRG ADDITIONAL CONTENTION 41 - REACTOR PRESSURE LIMIT/ SAFETY RELIEF VALVES

(3) JACK N. BAILEY ON DOHERTY CONTENTION 42 - POSITION INDICATION FOR SRV'S

Q. Panel, would each of you state your name, your position with your employer, and describe your professional experience and educational background?

A. My name is Steven A. Hucik and I am employed by the General Electric Company (GE) as Manager, Mark III Containment Engineering. My professional and educational background is described in Attachment SAH-1.

My name is John J. Boseman and I am employed by GE as a Senior Engineer in Valves and Auxiliary Equipment Design. My professional and educational background i

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described in Attachment JJB-1.

My name is Robert L. Huang and I am employed by GE as a Technical Leader of the BWR/6 Transient Design Group. My educational and professional background is described in Attachment RLH-1.

My name is Jack N. Bailey and I am employed by Houston Lighting & Power Company (HL&P) as Supervising Project Engineer of Mechanical, Nuclear and Health Physics. My educational and professional background is described in JNB-1.

Q. Mr. Hucik and Mr. Boseman, would you please state the purpose of your testimony?

A. The purpose of our testimony is to address Doherty Contention 17 which states that blowdown following a Power Excursion Accident (PEA), Loss of Coolant Accident (LOCA) or Power Coolant Mismatch Accident (PCMA) combined with a single or several relief valves stuck in either the fully open or fully closed position may cause loads which would crack the containment wall. In addition, the intervenors question the reliability of the safety relief valves used in a BWR and request that applicant be required to use the most reliable valve available.

Q. Mr. Boseman, please describe the safety relief valve design being used in Allens Creek.

A. The SRV's for ACNGS are spring loaded, direct

acting, dual function type valves. The current design used at ACNGS is based upon the experience gained in over 100 reactor years of BWR operations. The design of ACNGS SRV's, manufactured by Crosby Valve and Gauge Company, has eliminated the causes of previously experienced undesirable performances associated principally with reverse seated type multiple stage pilot operated safety relief valves. The ACNGS SRV's do not use pilot or air operator diaphragms. Instead, the ACNGS SRV's consist of a direct-acting safety valve with an electro-pneumatic actuator assembly to provide for two separate and independent modes of operation (safety and relief). This design improvement combined with existing manufacturing control of critical dimensions and clearances between all moving parts, stringent production testing and inservice maintenance and inspection will make it extremely unlikely that any of the 19 SRV's on ACNGS will stick open or closed.

Q. What has been the operating experience of Crosby SRV's?

A. Crosby SRV's have been used in both nuclear and non-nuclear applications for many years with an excellent service and performance record. The present SRV design with the previously stated improvements have undergone extensive qualification testing and are expected to have an even better service and performance record. To date the

operational history at Browns Ferry III and at Chinshan nuclear plants have shown this to be true.

Q. Mr. Boseman, please describe the operation and function of the safety relief valves.

A. Safety/relief valves (SRVs) protect against overpressurization of the Reactor Coolant Pressure Boundary (RCPB) by opening automatically in either the relief or safety modes of operation when the pressure setpoints are exceeded. Allens Creek has 19 safety/relief valves. The pressure setpoints for the automatic relief function of the valves are in the range of 1103 psig to 1123 psig. Of the 19 valves, eight specially selected SRVs, which are part of the ADS system, open automatically as part of the Emergency Core Cooling System for small breaks in the RCPB where depressurization of the reactor vessel is necessary to permit operation of the low pressure coolant systems.

The present design of the system is such that the 19 SRV's open at different pressure levels via the relief function. At 1103 psig, 1 valve opens; at 1113 psig, 9 more valves open; at 1123 psig, the remaining 9 valves open. In the relief mode the valve is opened by pressurizing an air cylinder which moves an actuating lever thereby lifting the valve stem, as shown on PSAR Figure 5.2.6. The air cylinder is pressurized when either of 2

solenoid valves is energized. The solenoids are automatically energized by an Instrumentation and Control signal generated by high reactor pressure or they can be manually energized by the operator.

If the reactor pressure exceeds 1123 psig and one or more of the safety/relief valves are not open, the valves will open automatically in the safety mode of operation when the pressure underneath the valve overcomes the spring force holding the disc closed. The safety setpoints range from 1165 psig to 1190 psig. All SRVs will be open by the time reactor pressure reaches 1190 psig.

Q. Mr. Hucik, please describe how the discharge through the SRVs cause loads on the containment?

A. The SRV discharge piping routes reactor steam from the relief valves to the suppression pool. The discharge piping is arranged so that the quenchers which are attached to the end of the SRV discharge piping are uniformly distributed in the suppression pool. When a safety relief valve is opened, there is a rapid pressure build-up in the discharge pipe. This rapid compression of the column of air in the pipe caused by the release of reactor steam causes a subsequent acceleration of the water column in the submerged portion of the pipe. During this water clearing process, the pressure in the pipe builds to a peak as the last of the water is expelled. At this point, the highly compressed

cushion of air between the water slug and the reactor steam begins to leave the pipe. As the highly compressed air exits, it forms an air bubble which expands and contracts, or oscillates, as it rises to the surface of the suppression pool. This oscillation of the air bubble causes a pressure disturbance throughout the suppression pool which is transmitted as a dynamic load to the containment. This air clearing process takes about 0.75 seconds. After the air clearing process, the quencher is acting only as a condenser as the steam from the RPV is discharged into the suppression pool and condensed. This will begin to heat the suppression pool. If the relief valve sticks open, it will continue to heat the pool but will not impart any significant dynamic loadings on the containment. This heat-up of the suppression pool will be controlled by the Residual Heat Removal (RHR) System which will keep the pool temperature within acceptable limits.

Q. Mr. Hucik, would you also describe the load combinations, pertinent to this contention, for which the containment has been designed?

A. There are two load combinations pertinent to this contention. They are as follows:

1. LOCA plus single SRV actuation.
2. Automatic Depressurization System (ADS)

actuation.

For the case of a relief valve stuck open, the worst case for containment design is a combination of LOCA blowdown loads with the loads due to a single SRV actuation. The dynamic loadings due to this load case would be the same as for LOCA plus a stuck-open relief valve.

For the case of relief valves failing in the closed position, load case #2 demonstrates that only 8 of the 19 relief valves are necessary to rapidly depressurize the reactor. Thus the failure of up to eleven relief valves in the closed position will not cause the violation of a safety limit.

The pressure loadings on the containment due to load cases 1 and 2 have been provided to HL&P by GE and are part of the design basis for the containment as described in Chapters 3 and 6 of the ACNGS PSAR.

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

A. My testimony addresses TexPirg Additional Contention 41, which alleges that there is inadequate protection against overpressurization of the ACNGS Reactor Coolant Pressure Boundary (RCPB) resulting from pressure increase transients. This concern arises, according to TexPirg, because the Nuclear Pressure Relief System (NPRS) is not designed adequately to ensure that during the most severe

abnormal operational pressure increase transient, pressure is maintained below the limit allowed by the American Society of Mechanical Engineers (ASME) Boiler & Pressure Vessel Code. TexPirg's concern stems from the fact that the analysis of the most severe operational transient resulting in the highest nuclear system pressure rise assumes that the reactor is shutdown by the high-neutron flux SCRAM. TexPirg has asserted that reliance on the high flux signal as a major contributor to the termination of the pressure transient, and hence as a critical design input into the NPRS, does not provide an adequate assurance against overpressurization of the RCPB because there is a history of poor performance in the BWR flux instrumentation systems with inaccuracies of 5.4 percent.

Q. Would you briefly describe the purpose and design of the NPRS?

A. As described by Mr. Boseman, the NPRS consists of 19 safety/relief valves located on the main steam lines between the reactor vessel and the first isolation valve within the drywell. These valves protect against overpressurization of the RCPB by opening automatically upon receipt of pressure signals (relief operation) to limit a pressure rise or by self-actuation (safety operation), if not already automatically opened for relief operation. The

events that lead to actuation of the safety/relief valves result from sudden reduction of steam flow while the reactor is operating at power. Major pressurization transients are caused by the closure of the MSIVs or the turbine control valves or turbine stop valves. The closure of these valves cuts off the steam flow path and isolates the reactor vessel from the condenser while steam is still being formed. The pressure inside the vessel thus increases rapidly.

The ASME Boiler & Pressure Vessel Code requirements of Article NB-7000 on setpoints of safety/relief valves are conservatively satisfied by (1) setting all setpoints at or below the reactor vessel design pressure and (2) by setting the setpoints so that the peak vessel pressure does not exceed 110 percent of the design pressure during the limiting pressurization event. The ACNGS safety/relief valves are set to operate in the relief mode from 1103 to 1123 psig and in the safety function from 1165 to 1190 psig. This satisfies the ASME Code requirements because all valves open at less than nuclear system design pressure (1,250 psig).

Q. What is the limiting pressurization event and how does it relate to SRV capacity?

A. The pressure transient resulting from the closures of all main steam line isolation valves (MSIV) represents the most severe pressurization transient when credit is taken only for an indirectly derived SCRAM. The analysis

of this transient conservatively assumes the failure of the direct, safety-grade main steam isolation valve position SCRAM. In this event the reactor is shutdown by the backup, indirect neutron flux SCRAM. Consequently, the analysis introduces a significant delay between the initiation of the transient (MSIV closure) and the initiation of a SCRAM. The probability of this event with failure of the safety grade scram on MSIV closure is very low.

The required safety/relief valves capacity is determined by analyzing the pressure rise from such a conservatively postulated transient. The plant is assumed to be operating at 105 percent of nuclear boiler rated steam flow conditions at a maximum operating reactor vessel dome pressure of 1,045 psig. It is further assumed that only one-half of the safety/relief valves operate in the pressure relief mode (setpoints are conservatively assumed to be in the range of 1,115 to 1,155 psig), and the other half is assumed to operate in the backup safety mode (spring setpoints are conservatively assumed to be in the range of 1,175 to 1,215 psig). The analysis indicates that the design valve capacity is capable of maintaining the reactor vessel pressure well below the ASME Code allowable pressure in the nuclear system (110 percent of design pressure or 1,375 psig). The peak pressure at the bottom of the reactor vessel is 1,294 psig. Therefore, the most severe over-pressure transient is termi-

nated well below the pressure limit required by the ASME Code.

Q. What are the conservative assumptions built into this analysis?

A. Under the general requirements for protection against overpressure, as given in Article NB-7,000 of Section III of the ASME Boiler & Pressure Vessel Code, analysis of overpressure transients can consider the effects of an appropriate SCRAM from the Reactor Protection System (RPS). Thus, the above-described overpressure transient analysis could have considered the effects of a SCRAM resulting from an RPS signal initiated by MSIV closure since this automatic SCRAM qualifies as an acceptable protection device under the provisions of the ASME Code. There are four main steam lines with two isolation valves per line. Position switches mounted on the eight main steam line isolation valves signal MSIV closure to the reactor protection system. Each of the switches is arranged to provide a signal to the reactor protection system before the valves are more than 10% closed. This provides early positive indication of closure. The logic for generating a SCRAM signal from isolation valve closure is as follows: Closure of two main steam line isolation valves on the same steam line will not initiate SCRAM. (To allow for MSIV testing.) However, closure of one main steam line isolation valve (MSIV) in two or more steam lines will cause a SCRAM. In other words, one of four

steam lines may be closed without initiating SCR. ' but closure of more than one steam line will initiate SCRAM assuming any single failure.

However, even though the Code allows for consideration of the immediate SCRAM generated by MSIV closure, the General Electric Company conservatively assumes the failure of the direct, safety-grade MSIV position SCRAM signals. The General Electric analysis relies only upon the delayed SCRAM signal generated by high-neutron flux.

When the MSIV's close and the vessel internal pressure rises, the steam bubbles in the core region collapse. With more water in the core, the neutron flux increases above the high-neutron flux SCRAM setpoint. This sequence takes considerably longer than the direct MSIV position SCRAM, and its use in the overpressure transient analysis is, therefore, very conservative.

Thus, TexPirg's allegation that the General Electric analysis is nonconservative neglects completely the conservatism of taking no credit for the earlier scram.

Q. TexPirg also asserts that the safety/relief valve capacity analysis does not account for an alleged history of unreliability in high-flux SCRAM signal circuitry. Is this assertion valid?

A. No, there are 4 divisions of Average Power Range Monitors (APRMs) which measure neutron flux and thus reactor

power levels. The APRM signals are monitored by the Reactor Protection System (RPS). If the APRM measurement exceeds the setpoint, a SCRAM will be initiated. The 4 divisions of APRMs are amply redundant and are routinely calibrated and checked through heat balances at power operations.

In addition, the high APRM scram setpoint assumed in the overpressure protection analysis is at a conservatively high level above the nominal setpoint. This accounts for initial setpoint errors and setpoint drift that may occur during operation. Typically, the assumed setpoint in the analysis is about 4% above the nominal setpoint. Furthermore, the neutron flux increase in the overpressure transient caused by MSIV closure with assumed failure in the direct scram is very rapid. So, the alleged APRM uncertainty of ±5.4% could only delay the scram initiation time by less than 50 milliseconds, and its effect on the peak vessel pressure is still well below the ASME Code limit.

Finally, further indepth protection offered by the safety/relief system is demonstrated by the fact that pressure limits will not be exceeded even if the high-neutron flux scram (the second or delayed SCRAM) is assumed to fail. The SCRAM under these conditions is initiated by the high reactor pressure trip signal. The probability of the simultaneous failure of the MSIV position SCRAM and high-

neutron flux SCRAM signals is obviously extremely low. But even assuming these incredibly unlikely events, the peak reactor vessel pressure for this transient is still below the ASME Code limit of 1,375 psig. Hence, even if TexPirg's unfounded claims of high-flux signal unreliability were true, the RCPB would still be adequately protected against overpressurization.

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

A. The purpose of my testimony is to address Doherty Contention 42 which alleges that the design of the SRV position indicators is inadequate. He bases his allegation on events at TMI and the main thrust of his contention is that HL&P has not explained how it will comply with the recommendations that evolved from the TMI incident.

Q. Are you familiar with HL&P's commitments to meet the new regulatory requirements that evolved from the TMI incident?

A. Yes, I am. I was the engineer at HL&P who had primary responsibility for the preparation of HL&P's detailed response to NUREG-0718. Accordingly, I am intimately familiar with the details of the steps taken to comply with the new regulatory requirements that evolved from TMI.

Q. Turning to Mr. Doherty's contention, does the ACNGS design provide direct indication of SRV position in the main control room?

A. Yes. As indicated in Appendix O, page 0-9" of the PSAR, Allens Creek will provide direct indication of SRV position in the main control room. This commitment was made in response to Item II.D.3 of NUREG-0718-"Licensing Requirements for Pending Applications for Construction Permits and Manufacturing License", which requires a demonstration that design and implementation can be completed prior to the issuance of an operating license.

Q. Please describe the SRV position indication design to be incorporated in the Allens Creek design.

A. SRV position indication will be determined by pressure measurement in the SRV discharge pipe. Data from plants presently in operation demonstrates that this method provides an adequate indication of SRV position. The actual pressure setpoint to be used at ACNGS will be determined from a combination of analysis and field test data, and will be submitted with the FSAR. Indication in the Main Control Room will be on two light matrices, one for each division of position measurement, on the Reactor Core Cooling Systems benchboard above the manual control switches for the relief valves. The indication will be redundant, safety grade, seismically and environmentally qualified, and powered from a Class IE power source. An alarm indicating that an SRV is open will be provided.

There are no questions regarding technical feasi-

bility or state-of-the-art of the SRV position indication design, nor is there any concern that it cannot be implemented prior to OL issuance. In fact, this design concept has been approved by the NRC for use on the Hatch nuclear plant operated by Georgia Power Company. Accordingly, there is no doubt that we can demonstrate that our design concept is technically feasible.

Attachment JNB-1

Jack N. Bailey

I graduated from Georgia Tech in 1972 with a degree in Electrical Engineering. From June 1972 through November 1977, I served in the U.S. Navy. During this time I attended the Naval Nuclear Power School, served on the USS Long Beach, a nuclear-powered guided missile cruiser, and served as a supervisor in the Naval Nuclear Power Training Unit, D1G Prototype.

I have been employed by HL&P since December 1979. Initially I was a member of the ACNGS Engineering Team with responsibility for piping program organization, painting requirements, inservice inspection access requirements and other balance of plant systems. From February 1980 to August 1980 I was Chairman of HL&P's TMI Design Task Force. The task force studied the TMI accident and made recommendations for needed studies and design changes. From August 1980 to March 1981 I was Engineering Team Leader for Allens Creek with responsibility for all NSSS systems and a variety of balance of plant systems. I was also responsible for HL&P's response to NUREG-0718, "Licensing Requirements for Pending Applications for Construction Permits and Manufacturing License." I was promoted to my current position as Supervising Project Engineer in March 1981, and in this position I am responsible for supervising the mechanical, nuclear and health physics design review of ACNGS.

RESUME

John J. Boseman

Position: Senior Engineer

Employer: General Electric Co., San Jose, CA.

Principal Duties and Responsibilities:

Responsible for the design, development, qualification tests, programs, and related technological growth of equipment assigned including all necessary technical and liaison assistance to support installation, testing, inspection and maintenance of the equipment. Formulated, directed, and participated in the design, tests, evaluations and programs to improve the reliability and optimization of Safety/Relief Valve designs for BWR applications.

Background

5/68 - 1/77 - Product Engineer with General Electric Co., MAO, Schenectady, N.Y. - As cognizant engineer liaisoned and performed design, development, tests, manufacture, evaluations, installation and field support functions for various types of Naval nuclear power plant fluid components. As Product Engineer performed and provided technical direction for the design, applications, qualifications, development, installation and maintenance of assigned equipment. (e.g. - Valves (checks, gates, globes, relief, motor operated, hydraulic operated, etc.; magnetic separators, filters, pressure vessels, pipe and fittings, demineralizers, reactor viewing devices, etc.)). Assigned to participate in the Navy's 1970 Valve Design Review Task Force.

6/66 - 4/68 - Associate Engineer with Lockheed Missile & Space Co., Sunnyvale, CA. - Analyzed, proposed, designed, developed, tested and liaisoned the manufacture of advanced electromechanical microwave antenna and antenna systems for Satellite and Polaris/Poseidon Missile applications.

2/66 - 5/66 - Assistant Engineer with The Boeing Company, New Orleans, LA. - Performed environmental and simulated testing of fluid power components and sub-systems applicalbe to the Saturn IV & V Booster System. Established cause for LOX and lift check valve failures and recommended corrective action to preclude recurrence.

7/64 - 1/66 - Assistant Engineer with Delta Steamship Lines, Inc., New Orleans, LA. - Operated, tested, maintained and repaired steam and diesel power plant systems and equipment including the review and liaison of design proposals and major shipyard overhauls and repairs. Conceived and demonstrated an emergency technique to repack stern gland while underway.

Education/Training

- 1964 - B.S. Marine Engineering - U.S. Merchant Marine Academy.
- Nuclear Radiation & Environmental Effects/Navy Structural Design Basis-63/Reactor Plant Technology/ASME Pressure Vessel Codes/ Kepner-Tregoe "Problem Solving & Decision Making Course.

Publications/Articles

- ASME 80-C2/PVP-29 - OPERABILITY ASSURANCE TESTING OF ASME CODE, CLASS 1, SAFETY/RELIEF VALVES.

- EVALUATION OF 3 THERMALLY SHOCK TESTED 1/2-INCH GLOBE VALVES (MEDF #54) - U.S. Navy Document (Restricted).

- U.S. NAVY NUCLEAR VALVE DESIGN MANUAL (VDM-71) (Classified).

- Plant Equipment Design Memorandum No. 126-74 - SEALING MECHANISM FACTORS.

Robert L. Huang

I received a Bachelor of Science Degree in Nuclear Engineering from National Tsing Hua University in Taiwan, China in 1968. During 1969 to 1975 I attended Columbia University in New York and received a Master of Science Degree in Nuclear Engineering in 1970 and a Doctor of Engineering Science Degree in Nuclear Engineering in 1975.

I joined the General Electric Company's Nuclear Energy Division in September, 1974. I was responsible for BWR/6 transient safety design, which includes reactor vessel overpressure protection and reactor fuel overpower protection. Shortly thereafter, I performed transient safety analyses for the General Electric Standard Safety Analysis Report (GESSAR), and for the Grand Gulf FSAR. Since then, I have performed design and analysis studies for all BWR product lines, and led the transient design efforts which established the BWR/6 product line.

In my current position as a Technical Leader, I provide technical guidance and work direction to engineers to perform the reactor system transient safety design analyses and licensing evaluations on BWR plants. I am also responsible for the establishment and specification of reactor system hardware and reactor protection system functional requirement.

Attachment SAH-1

Steven A. Hucik

Mr. Hucik is manager of the Mark III Containment Engineering Unit of the Nuclear Power Systems Engineering Department in the General Electric Company. His employment with General Electric began in 1973 and his experience has been mainly in the containment loads area. His unit is responsible for all Mark III containment analysis and dynamic load definition for loss-of-coolant accident (LOCA) and safety/relief valve discharges.

Mr. Hucik's experience in Mark III containment includes responsibility for the dynamic loads development and application, documented in the Mark III Containment Loads Report. He was responsible for the analysis of the Caorso SRV test data used to support the SRV load reduction defined in the final Mark III Containment Loads Report.

Mr. Hucik's Mark III experience is also supported by his previous involvement in the Mark I Containment Program. His involvement included application of test data, load definition development, customer and Architect/Engineer interface, and US regulatory presentations.

Mr. Hucik has provided support to the Mark III Customers and Architect/Engineers in the load application area for both LOCA and SRV discharge events. He has also made presentations of the design and licensing bases to the US regulatory agencies, customers, and Architect Engineers.

Mr. Hucik is a 1973 graduate of Washington State University with a B.S. Degree in Mechanical Engineering. In

1976, he received an M.S. Degree in Mechanical Engineering from the University of California, Berkeley. He is also a Registered Professional Engineer in the State of California.