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 OF DAVID A. HAMON ON BEHALF OF HOUSTON LIGHTING & POWER CO. ON TEXPIRG ADDITIONAL CONTENTION 55 RAPID DEPRESSURIZATION - STEAM BREAK

Q. Would you state your name and place of employment?

A. My name is David A. Hamon and I am employed as a Technical Leader in the ECCS Engineering Unit at the General Electric Company.

Q. Would you describe your professional qualifications?

A. A copy of my professional qualifications is given in Attachment DAH-1 to this testimony.

Q. What is the purpose of your testimony?

A. This testimony responds to TexPirg's Additional Contention 55 which postulates a reactivity insertion accident initiated by a guillotine break in a main steam line. TexPirg also mentions a break in a recirculation line, but the governing phenomena is the same in either case for reasons that are apparent in the explanation of the actual

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events.

TexPirg asserts that the reactor coolant pressure decrease at the point of the break will draw coolant water up into the core. This upward flow of water supposedly will push steam bubbles out of the core, collapsing voids, and cause a positive reactivity insertion. TexPirg then asserts that reactivity will increase dangerously before the reactor SCRAMS. TexPirg bases this scenario on tests conducted in June 1970 by the Idaho Nuclear Experimental Laboratories (the so-called Special Power Excursion Tests (SPERT), particularly those reported in IN-1370).

Q. Will you describe the actual sequence of events which occur after the accident hypothesized by TexPirg?

A. Under normal operating conditions, coolant water flowing into a BWR core is at a temperature and pressure close to the saturation point. Dome pressure for an operating BWR/6-238, like Allens Creek, is about 1040 psia; dome temperature is 549°F. Core inlet temperature is approximately 533°F; hence, coolant entering the core is approximately 16°F subcooled.

After a guillotine break in the main steam line coolant pressure in the reactor core will fall rapidly. (A guillotine pipe break is a pipe rupture in which the break area exposed is equivalent to two pipe diameters.) The

depressurization rate averages 20 psi per second for the first 30 seconds after the break occurs and slowly decreases thereafter. Since the pressure at the break will be significantly lower (initially atmospheric pressure) than the water and steam inside the vessel, the water and steam in the rest of the vessel will move toward the break. Simultaneously, the rapid depressurization of the reactor vessel, caused by the escape of steam out of the break, will cause the water in the core to flash to steam rapidly. The rapid change into steam drastically decreases the effectiveness of the coolant as a moderator and, therefore, introduces a large amount of negative reactivity.

Q. How does General Electric calculate the core thermal hydraulics after the main steam line break?

A. General Electric's computer code called "LAMB" calculates among other things, the physical state of the reactor coolant after a large pipe rupture. The LAMB code is used to analyze the short-term thermodynamic and thermal-hydraulic behavior of the coolant in the vessel during a postulated loss-of-coolant accident (LOCA). In particular, LAMB predicts the core flow, core inlet enthalpy, and core pressure during the early stages of the reactor vessel blowdown. A more detailed description of the model is given in General Electric Company Analytical Model for

Loss-of-Coolant Analysis in Accordance with 10 CFR 50, Appendix K, NEDO-20566.

This code has verified that after a large rupture, the coolant will flash to steam so rapidly that a surge of cooler water up through the tortuous paths into the core -- generated by differential pressures lasting only seconds -- simply does not occur.

The General Electric LAMB code has been verified by the TLTA tests described in Mr. G. L. Sozzi's testimony. The TLTA tests were performed for large recirculation line breaks instead of large steamline breaks, but they still demonstrate that the coolant will rapidly flash to steam during a rapid depressurization. For these tests the water level dropped below the elevation of the recirculation line in about 10 seconds, uncovering the break. Once uncovered the break was equivalent to a large steamline break -- rapid depressurization began and the coolant began to rapidly flash to steam.

Q. What do you understand to be the source of TexPirg's contention?

A. TexPirg has apparently interpreted IN-1370 to purportedly show results contrary to the sequence of events described above. This is an understandable mistake. IN-1370 (p. 104) contains the untested and unsubstantiated conjecture that:

. . . If a pipebreak occurs, in a BWR, it appears very likely that the resultant pressure relief could cause a significant amount of bubble generation and growth in the water surrounding the core region. At the same time the depressurization could cause the water moderator level in the core region to rise. An increase in water level in the core region would result in a reactivity accident. . . ."

The assertion that an increase in moderator water level in the core would result in a reactivity accident is plainly incorrect. Although the void generation in the core may indeed briefly increase the volume of the moderator -- thus raising the water level in the vessel -- the creation of large voids will significantly reduce moderator density resulting in a large negative reactivity insertion. This result has been conclusively confirmed by exacting calculations and appropriate tests.

Q. What are your conclusions concerning TexPirg Additional Contention 55?

A. Significant negative reactivity is immediately introduced following a steam line break in a BWR, and the reactor begins to shut itself down even before the control rods are automatically inserted. The Reactor Protection system SCRAMS the reactor seconds after the break occurs by which time the fission rate has already been reduced by the loss of moderator density. Hence, the actual consequences of a rapid depressurization are the opposite of what TexPirg

supposes.

David A. Hamon

I received a B.S.E. degree in Mechanical Engineering from Arizona State University in 1976. This degree was accompanied by an Outstanding Graduate award from the Faculty of Mechanical Engineering. Subsequently, I attended the University of California at Berkeley where I received an M.S. degree in Mechanical Engineering in 1979.

Following my graduation from Arizona State University in 1976 I went to work for the General Electric Company as Program Engineer in ECCS System Design. I later worked as a Program Engineer in ECCS Engineering. I have also worked as a Program Engineer in Fuel Rod Thermal and Mechanical Analysis Unit and in a unit doing unique nuclear analyses. In my present position at General Electric I am involved in the administration of the CHASTE and LAMB/SCAT computer codes, which are used extensively in LOCA analyses performed by General Electric on behalf of all BWR projects. I have also performed several small break analyses and prepared reports for the BWR Owners' Group in response to post-TMI concerns. I have also been involved in the recirculation flow valve closure analysis, and in responding to NRC concerns about GE's fuel clad swelling and rupture models.