10/09/81

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)	)	

NRC STAFF TESTIMONY OF SHOU-NIEN HOU
REGARDING FLOW INDUCED VIBRATION [TEXPIRG'S
CONTENTION NO. 11 AND DOHERTY'S CONTENTION NO. 31]

Q. What is your name and your position with the NRC?

A. My name is Shou-Nien Hou. I am a Principal Mechanical Engineer in the Mechanical Engineering Branch, Division of Engineering, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission. A copy of my professional qualifications is attached.

Q. What is the purpose of this testimony?

A. The testimony addresses Doherty Contention 31 and TEXPIRG Contention 11 which basically allege that there is no adequate assurance that the Allens Creek reactor internal structures will be able to sustain vibrations induced by operating flow transients, especially the Local Power Range Monitor (LPRM) tubes and Feedwater Spargers. They further allege that the basis for this contention is that several operating BWRs have experienced these vibrational problems. In addition, this testimony addresses the preoperational assurance program, which includes analyses

and testing, required to verify design adequacy of reactor internals to sustain flow-induced vibration.

Q. What are the NRC regulatory requirements to ensure that reactor internal structures are safe from flow-induced vibrations?

A. In accordance with Section 3.9.2 of the NRC Standard Review Plan, a preoperational vibration assurance program in conformance with Regulatory Guide 1.20, "Comprehensive Vibration Assessment Program for Reactor Internals During Pre-operational and Initial Startup Testing," should be implemented for any reactor. For the first-of-a-kind reactor, the prototype program includes vibration assessment analysis, monitored flow tests, and visual inspection of reactor internals after the testing. For reactors similar to the prototype reactor, the non-prototype program consists of only testing and inspection. Since reactor internals are tested to over one million vibration cycles under actual flow transients, and subsequent visual inspections are conducted to find any structural degradation, the program is intended to verify design adequacy of reactor internals to sustain flow-induced vibration.

For program implementation, since testing can only be done after completing the construction of the reactor and its related systems, the Applicant is required to commit to a preoperational vibration assurance program in their construction permit application and have details of the program and test results submitted for NRC approval prior to the issuance of operating license.

Q. Does Allens Creek Unit 1 meet such requirements?

A. Yes. The Applicant has committed to comply with Regulatory Guide 1.20. Since the plant is still in its construction permit application stage, a commitment by the Applicant to implement Regulatory

Guide 1.20 is sufficient and acceptable for the reason noted above. These requirements will include "non-prototype" testing and inspection since Allens Creek is not expected to be a prototype reactor.

Q. In response to TEXPIRG Contention 11, can you describe briefly what kind of sparger failures occurred in the early operating BWRs and what may be considered an acceptable explanation to the NRC Staff regarding the cause of failure?

A. Gradually developed cracks occurred at the vicinity of the tee box junction between the thermal sleeve and the sparger arms, and wear occurred on the sparger arm end pins, brackets and thermal sleeves. Since the spargers used in early BWR's did not fit tightly into the nozzle, the Staff on review of this problem concluded that the leakage flow of feedwater through the gap which causes thermal fluctuations and vibratory stresses resulted in the sparger cracking. It should also be noted that past failures of this type were not instantaneous and catastrophic and can be detected prior to a complete severance.

Q. On what basis has the NRC Staff accepted the information regarding feedwater spargers to be used in Allens Creek plant Unit 1?

A. The Staff was informed that feedwater spargers of improved design, the GE triple-sleeve type, will be used in the Allens Creek plant. The new design modified the tee box junction by the use of the forged tee, which provides less flow restriction and local stress concentration. The use of triple sleeves, a double piston ring scale, and an improved interference fit is likely to eliminate the leakage flow. Results of full scale flow tests and performance in several operating reactors using this type of sparger have demonstrated that the vibration

levels were acceptably low for all flow and load variations experienced. So far, no failure of this type of sparger has been reported. Furthermore, as in other BWR plants, we will evaluate details of in-service inspection and monitoring programs for early detection of any failure. Such evaluations will be performed prior to the issuance of the operating license. Since improved spargers will be used at Allens Creek, sparger failure, if any, will develop gradually and be detectable. This fact, and the fact that past failures have never caused any radiation hazard to the public, leads us to conclude that the information provided by the Applicant at this stage is adequate and acceptable for a construction permit application.

Q. In response to Doherty Contention 31, can you describe briefly what kind of LPRM tube failures occurred in the early BWRs and what may be considered an acceptable explanation to the NRC Staff regarding the cause of failure?

A. Vibrating LPRM tubes impacted nearby fuel channel boxes and caused gradual wear of these impacted channel boxes. The Staff's evaluation concluded that the coolant flow through core plate bypass flow holes induced excessive vibration in the LPRM tubes. It should be noted that the damage was located in the impacted channel boxes and has never occurred in the LPRM tubes as indicated by Doherty's Contention No. 31. LPRM tubes have never failed in any BWR plant.

Q. On what basis has the NRC Staff accepted this information regarding LPRM tube integrity, which is the sole concern of Doherty Contention No. 31, in Allens Creek plant Unit 1?

A. Since the core plate design in the Allens Creek plant does not have those bypass flow holes, we believe that vibration levels of LPRM

tubes in the Allens Creek plant are likely to be low judging from actual performance of several operating plants in which these holes were plugged. Furthermore, we will evaluate details of monitoring systems prior to the issuance of the operating license to ensure that excessive LPRM vibrations or any related new vibration problems will not occur.

Since Allens Creek Unit 1 has adopted the core plate design modifications, the channel box wear, if any, will develop gradually and be detectable. In addition, past failures have never caused any radiation hazard to the public. Accordingly, the Staff concludes that information provided by the Applicant is adequate and acceptable for a construction permit application.

PROFESSIONAL QUALIFICATIONS

DR. SHOU-NIEN HOU

U. S. NUCLEAR REGULATORY COMMISSION

MECHANICAL ENGINEERING BRANCH

DIVISION OF ENGINEERING

I am a Principal Mechanical Engineer, an assistant to the Chief of the Mechanical Engineering Branch (MEB) for performing independent review of generic matters and coordinating technical position among the staff. For nine years in MEB, I have reviewed plant design criteria, plant operating problems, plant safety considerations and dynamic analysis and testing of piping, equipment, reactor internals and nuclear safety features. I also served as the leader of the Seismic Qualification Review Team for conducting plant seismic audit, and the Task Manager for developing staff position on plant safety against the postulated pipe rupture event. In addition, I am on several National Standard Committees and participated in several Regulatory Guide developments.

Born 1934 in China, I came to USA in 1957. I received the B.S. in Civil Engineering from Taiwan University in 1955, the M.S. in Structural Dynamics from Virginia Tech. in 1958, and the Ph.D. in Structural Mechanics from M.I.T. in 1968. For 26 years after the B.S., I have had various working experiences in structural design, stress analysis, research and development in space vehicle dynamics, and technical review in nuclear power plant safety. I was a visiting lecturer to universities in England (1971), and Chile (1975), and to the government in Taiwan, China (1975). I was the author of a dozen technical papers, a recipient of Apollo Achievement Award from NASA (1969) and a High Quality Performance Increase from AEC (1975), and a member of Sigma Xi, Tau Beta Pi, Chi Epsilon, AIAA and ANS.

During 1955-57 I served as a commissioned engineering liaison officer with the National Chinese Navy in various US-aid military projects, and passed the Examination of Professional Engineers.

In 1957 I came to the USA with teaching assistantship from VPI where I completed the M.S. degree in one year. The thesis was entitled "Vibration Behavior of Parabolic Arches." I received "VPI Structural Ten" award after graduation.

During 1958-64 I was a Bridge Design Engineer with Virginia Highway Department. For six years, I worked in stress analysis of steel, reinforced concrete and prestressed concrete structures. In 1960, after taking short courses with IBM, I was assigned to perform independent studies in developing computer capability in design and analysis. In 1963 I published the solution manual for "The Mechanics of Solid" authored by G. L. Rogers.

In 1964 I received a Research Assistantship from MIT where I completed the Ph.D. degree in 1968. My studies were emphasized on mechanical vibrations,

material behavior under various loading and temperature conditions, and stochastic processes in engineering applications. Based on my research in random vibration theories, I completed the doctoral thesis entitled "Earthquake Simulation Models and Their Applications." In MIT I was elected to honor societies of Sigma Xi in 1965, Tau Beta Pi in 1966, and Chi Epsilon in 1967.

During 1968-71 I worked as a member of the technical staff with the Space Vehicle Dynamics Department of Bellcomm, Inc., which was performing technical studies, system planning and analysis in the Office of Manned Space Flight within NASA headquarters, Washington, D.C. My works were related to investigation, evaluation and development of dynamic analysis and testing technology in space vehicle and missile dynamics, as well as participating in task groups for solving problems such as POGO, rover stability, wind simulations, and fuel tank sloshing etc I also had full responsibility for developing computer capability to perform large system dynamic analysis, such as modal synthesis. I received "Apollo Achievement Award" by NASA in 1970 and "Recognition of Accomplishment" by AT&T in 1971. I was a visiting lecturer to England in 1971 and published ten technical papers in this period of time.

In January 1972 I joined the U.S. Atomic Energy Commission and have remained with this organization through the transition to the U.S. Nuclear Regulatory Commission. During this time I have participated in the review of plant operating problems and design criteria and evaluation of over forty construction permits and operating license applications in the area of dynamic effects of LOCA, earthquakes, pipe rupture, and operating transients on systems, components, equipment, reactor internals and nuclear safety instrumentation. I had served as the leader of SQRT (Seismic Qualification Review Team) for conducting plant seismic audit, and Task Manager of a generic program for investigating design criteria for plant protection against postulated high energy line rupture. I am a member of industry code and standards writing bodies including: ANSI N176, "Design Basis for Protection of Nuclear Power Plants Against Effects of Postulated Pipe Rupture," and Standard IEEE-344 "Recommended Practices for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations." During my service in NRC, I have received one "High Quality performance Step Increase" and published two technical papers. In 1975 I was invited by the University in Chile to give lectures regarding seismic design of nuclear power plants and by the National Chinese government to discuss various subjects concerning nuclear power plant safety.