UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
HOUSTON LIGHTING & POWER COMPANY	)	Docket Nos. 50-466
	)	
(Allens Creek Nuclear Generating Station, Unit 1)	)	

NRC STAFF SUPPLEMENTAL TESTIMONY OF SAI P. CHAN RELATIVE TO CONTAINMENT BUCKLING AND REACTOR PEDESTAL

[Doherty Contentions 9 and 27]

Q. Please state your name and position with the NRC.

A. My name is Sai P. Chan. I am employed at the U.S. Nuclear Regulatory Commission as a Senior Structural Engineer in the Structure Engineering Branch.

Q. Have you prepared a statement of educational and professional qualifications?

A. Yes. It is attached to this testimony.

Q. What is the purpose of your testimony?

A. The purpose of my testimony is to respond to Doherty Contentions 9 and 27 which state as follows:

Doherty Contention 9

That Intervenor's health and safety interests are inadequately protected because Applicant's steel containment shell is not strong enough by design to resist dynamic and static loads which may plausibly occur in the life time of the atomic plant.

Doherty Contention 27

> The concrete in the pedestal beneath the ACNGS reactor may be sufficiently weakened by heat from a design basis accident to compromise the safety of the plant after its subsequent return to operation.

Q. With respect to Doherty Contention 9, buckling of the steel containment, has that issue been identified as an "unresolved safety issue"?

A. No. This contention refers to the "Task B-5" listed in Table C.2 "List of Technical Activities," in "Safety Evaluation Report related to Construction of Allens Creek Nuclear Generating Station, Unit 1," Supplement No. 2, NUREG-0515, March 1979. The issue is listed as a Category B generic technical activity which is defined as: "Those generic technical activities judged by the staff to be important in assuring the continued health and safety of the public but for which early resolution is not required or for which the staff perceives a lesser safety, safeguards or environmental significance than category A matters." Table C-1, NUREG-0515.

Q. What is the generic concern to be addressed by Task B-5?

A. The most recent statement of the concern by the NRC Staff is the statement in "Generic Task Problem Descriptions, Category B, C and D Tasks, "NUREG-0471, June 1978. That statement is:

> Buckling Behavior of Steel Containments - The structural design of a steel containment vessel subjected to unsymmetrical dynamic loadings may be governed by the instability of the shell. For this type of loading, the current design verification methods, analytical techniques, and the acceptance criteria may not be as comprehensive as they should be. Section III of the ASME Code does not provide detailed guidance on the treatment of buckling of steel containment vessels for such loading

> conditions. Regulatory Guide 1.57 recommends a minimum factor of safety of two against buckling for the worst loading condition provided a detailed rigorous analysis, considering inelastic behavior, is performed. On the other hand, the 1977 Summer Addenda of the ASME Code permits three alternate methods, but requires a factor of safety between 2.0 and 3.0 against buckling depending upon the applicable service limits. NUREG-0471, p. B-7.

Q. What are the objectives of Task B-5?

A. As stated in NUREG-0471 the task has the following specific objectives:

1. To review and assess the assumptions and methodology presently used in the buckling analysis of steel containment shells,

2. To establish general standard design and acceptance criteria for the dynamic/static stability of steel containment shells, particularly for steel containments subjected to unsymmetrical internal or external dynamic loads,

3. To evaluate the computer programs presently used in the buckling analysis and design of steel containment shells by developing benchmark problems to verify these programs, and

4. To perform selective detailed reviews of typical containment designs to assess the effect that any new licensing requirements may have on different types of containments.

Q. Have any new licensing requirements been established?

A. No. As stated on page C-4 of NUREG-0515, Task Action Plans have not been approved by the Technical Activities Steering Committee for Category B, C and D Tasks.

Q. Has such approval been made since NUREG-0515 was published in March, 1979?

A. No.

Q. Seventeen "Unresolved Safety Issues" are listed on page C-13 of NUREG-0515. Has that list been updated?

A. Yes. The Commission has approved four new "Unresolved Safety Issues" (Letter S. J. Chilk to W. J. Dircks, Subject: SECY-80-325 - Special Report to Congress Identifying "Unresolved Safety Issues (Commission Action Item), dated December 22, 1980). Candidate issues considered by the Commission originated from concerns identified in NUREG-0660, "NRC Action Plan as a Result of the TMI-2 Accident;" ACRS recommendations; abnormal occurrence reports and other operating experience. Task B-5 continues as a Category B Task and is not classified as an "Unresolved Safety Issue."

Q. Has any new information been developed during consideration of this contention that was not previously known to the Staff, and which sheds new light on the categorization of the generic concern.

A. No new information has been provided by the Intervenor or developed by the Staff

Q. Does the Allens Creek application meet the Commission's present requirements?

A. Yes. As stated in Section 3.8.1 "Steel Containment" of NUREG-0515, the Applicant has utilized Regulatory Guide 1.57, "Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components," as the basis for the buckling criteria for the steel containment. The Commission accepts regulatory guide positions as one way of meeting its requirements.

Q. With the above noted concern with respect to containment buckling, why is it practical to proceed with construction?

A. Again, as indicated in Section 3.8.1 of NUREG-0515, we do not anticipate that the end product of this program will result in significant design changes, but rather will produce a clear and precise set of requirements for future licensing actions and that if anticipated results are not realized, design modification during construction are feasible.

Q. Why is it acceptable to proceed with construction of ACNGS and other plants if the resolution of this matter could later result in changed requirements for future licensing actions?

A. The Staff does not regard the buckling of the steel containment issue as being so critical as to warrant immediate resolution. The rationale for such a licensing approach is as follows:

1. Buckling of shells and plates has been the subject of numerous studies. Each study is usually limited to a shell of specific geometrical configuration and loading. Generally the results of such a study are at best applicable only to the particular shell configuration under the particular loading. However, the use of Regulatory Guide 1.57 related criteria is expected to be adequate and to provide ample margin of safety.

2. Stiffeners are used in the Allens Creek steel containment, and it is generally believed that the use of stiffeners will reduce the sensitivity of buckling to the shell geometrical imperfections, especially with a large shell structure as a steel containment. Use of the stiffeners, therefore, further minimizes the likelihood of buckling.

3. The steel containment of Allens Creek is designed for the loads which may give rise to its buckling. The conservatism associated with the definition of the loads is believed to compensate the uncertainty related to the buckling concern.

4. In case the prospective research program concludes that strengthening of the containment is required, it can be accomplished by welding additional stiffeners to the containment without undue difficulty even after the plant is put into operation.

Based on the foregoing, the Staff concludes that even though buckling of the containment is classified as a generic safety issue, the licensing actions and measures taken by the Applicant and reviewed by the Staff provide reasonable assurance that the health and safety of the public will be protected.

Q. Turning now to Doherty Contention 27, weakening of the pedestal concrete, can you briefly describe the purpose and characteristics of the reactor pedestal?

A. The reactor pedestal provides support for the reactor vessel by means of a support skirt anchored to the reactor pedestal and welded to the vessel bottom head. The reactor pedestal also supplies support for the reactor biological shield wall. The pedestal basically consists of two concentric steel cylinders with the annular space between filled with concrete.

Q. Is the strength of the concrete considered in the load bearing design of the pedestal?

A. No. The basic material of the pedestal is structural steel and, therefore, the strength of the pedestal depends on the steel. The concrete is non-load bearing and, accordingly, the contribution to the pedestal strength of the concrete is not considered in the design. The fill concrete is used to provide additional biological shielding. In reality, however, the concrete will also add strength to the pedestal.

Q. During postulated power excursion or loss-of-coolant accident conditions, what is the maximum temperature the reactor pedestal is designed to withstand?

A. The maximum temperature to which the pedestal will be subjected during these accidents is about 330°F. At this temperature, there is some loss of steel strength, but this has been taken into consideration in the design. Therefore, the structural integrity of the pedestal will be maintained under the postulated accident conditions.

Q. What would happen to the concrete under the postulated accident conditions?

A. The temperature of 330°F will not significantly affect the added strength of the concrete because the concrete is confined and sealed by the steel cylindrical box. This temperature will result in practically no loss-of-concrete moisture and, therefore, its inherent strength should be maintained.

Q. What is your conclusion with respect to this contention?

A. As noted above, postulated accident conditions should not result in any weakening of the reactor pedestal and, in particular, the pedestal concrete. In any event, since the pedestal concrete is not considered in the design of the pedestal strength, any weakening or cracking of the concrete will not create any safety hazard.

PROFESSIONAL QUALIFICATIONS
OF
SAI P. CHAN
STRUCTURAL ENGINEERING BRANCH
DIVISION OF ENGINEERING

I am a senior structural engineer in the Structural Engineering Branch of the Division of Engineering. I am responsible for the evaluation of seismic analysis and design of structures, systems and components of nuclear facilities assigned to the Branch.

I received a B.S. Degree in civil engineering with honor from Lingnan University, China, in 1943. I received the degree of Master of Science from the University of Illinois, Urbana, Illinois in 1950 and the degree of Ph.D (Structural Engineering) from the same institution in 1953.

I taught undergraduate students at the National Chiao-tung University, Shanghai, China from September 1943 to August 1947. From October 1947 to August 1949 I studied at the University of Paris, France under a scholarship sponsored by the Nationalist Chinese Government and worked as an architectural engineer in the Atelier Le Corbusier, Paris, France. During the years 1951 and 1952, I worked as Research Assistant at the University of Illinois where I developed numerical methods for dynamic analysis of structures.

Since 1953 I have served in the structural engineering area including research, development, design and analysis for the construction, aerospace and power industries. My experience in structural methodology and stress analysis includes development of computer programs and numerical methods for dynamic analysis of framed and shell structures; analysis of composite, laminated and anisotropic structures; structural optimization and nonlinearities; postbuckling and dynamic behavior of stiffened and monocoque shells. I also taught at the University of Denver part-time for two years in Theory of Elasticity and Theory of Plates and Shells.

My experience in seismic design and ground shock problems involves earthquake design of a fossil-fuel power plant in California; mining structures and facilities; launch towers and silos for the Titan missiles; ground shock studies for military structures; seismic design and analysis of containment structures and auxiliary buildings or nuclear power plants.

I joined the U.S. Atomic Energy Commission (now Nuclear Regulatory Commission) in 1972. As a member of the Structural Engineering Branch, Division of Engineering, I have participated in developing criteria for seismic design and instrumentation for nuclear power plants, performed evaluations of technical reports concerning structural dynamics and reviewed numerous nuclear power plants in the area of seismic and structural design.

I am a member of the American Society of Civil Engineers, Earthquake Engineering Research Institute, and the American Institute of Aeronautics and Astronautics. I am registered as Professional Engineer in the states of Colorado and Georgia. I have published technical papers in the Journal of Royal Aeronautical Society and Aircraft Engineering, and several research reports for the Lockhead-Georgia Research Laboratory.