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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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

NRC STAFF SUPPLEMENTAL TESTIMONY OF
VINCENT T. H. LEUNG REGARDING SEISMIC CATEGORY I
CONTROL RODS AND CONTROL ROD DRIVE RETURN LINE

[ASLB Question 8 and Doherty Contention 48]

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

A. My name is Vincent T. H. Leung. I am a system engineer in the Auxiliary Systems Branch of the Division of Systems Integration. A copy of my statement of professional qualifications is attached.

Q. What is the purpose of this testimony?

A. The purpose of this testimony is to respond to ASLB Question 8 and to Doherty Contention 48. These issues will be addressed separately below.

Q. What is the question posed by ASLB Question 8?

A. ASLB Question 8 asks whether the control rods, control rod drives, and the hydraulic control units should be designed to Seismic Cateogry I requirements.

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Q. In direct response to the Board's question, are the control rods, control rod drives, and the hydraulic control units designated as Seismic Category I for Allens Creek?

A. Yes. All of the above components are designed to Seismic Categor;y I and Safety Class 2 requirements.

Q. To respond further, should any of the above components be designed to Safety Class 1 requirements?

A. As indicated in the Allens Creek PSAR, Safety Class 1 design requirements will be applied to components of the reactor coolant pressure boundary (RCPB) as defined by 10 C.F.R. § 50.2(v). Since portions of the control rod drive system are directly connected to the reactor vessel and the effects on RCPB due to system pipe break can be significant, those portions of the system which penetrate the primary containment (including the outermost containment isolation valve) should be designed to Safety Class 1 requirements.

Q. Do the hydraulic control units and control rods have to be designed to Safety Class 1 requirements?

A. No. The NRC Staff has reviewed the design and safety criteria of these components and concluded that control rod failure would not breach the RCPB directly. The standby liquid control system is a "backup" system to the control rods and will be activated to perform its safety function upon failure of the control rods. Failure of the hydraulic control unit's piping will have negligible effect on the RCPB because of the small piping size involved. Based on the above review, the Staff concludes that the design and safety criteria for the hydraulic control units and control rods are acceptable.

Q. With regard to Doherty Contention 48, what does it allege?

A. Doherty Contention 48 alleges that:

> ACNGS should be designed with a control rod drive return line (CRDRL), because this source of high pressure water functions as an additional safeguard against events where there is water loss from the reactor vessel yet pressure remains high.

Q. What is the function of the control rod drive return line?

A. The control rod drive return line (CRDRL) was designed to provide a reactor pressure reference to the control rod drive (CRD) system and to return to the reactor vessel exhaust water from CRD movement and water in excess of system requirements.

Q. What problems have been reported with respect to this system?

A. In April of 1975, a GE task force investigating cracking in austenitic stainless steel piping reported unexpectedly high top-to-bottom thermal gradients in CRDRL nozzles. Cracking initiation susceptibility was cited. Operating experience has proven this susceptibility in that cracking has been found to be widespread.

Q. What recommendations have been made to solve the problems?

A. A subsequent GE study of the CRDRL nozzle cracking problem resulted in a series of recommendations to various licensees. The staff has reviewed each GE recommendation and has determined and reported in NUREG-0619 that: (1) valving out of the return line is acceptable only as an interim measure, (2) rerouting the return line to another system which connects to the reactor vessel is preferable, and (3) only certain BWR classes may implement the third and last of the GE recommendations, to cut and cap the line and nozzle without rerouting, and then only after specific testing has been completed.

Q. How do these recommendations affect BWRs under construct'on, in review, or operating?

A. Several BWRs under construction do not have CRDRLs; neither line nor nozzle will appear in any future GE GWRs. The Allens Creek pressure vessel will have a CRDRL nozzle since it was fabricated before the cracking problem was discovered. However, it will not have a CRDRL and the nozzle will be capped. The recommendation to remove the CRDRL was based on the need to prevent nozzle cracking and on GE's determination that the line had never been necessary in order to attain an acceptable CRD reference pressure to the reactor vessel. Reference pressure for proper operation of the system may be obtained by system adjustments in operating reactors.

Q. Does the Staff agree with that portion of the contention that asserts that an additional source of high pressure water should be provided?

A. Yes.

Q. Can this be provided without the control rod drive return line?

A. Yes.

Q. How will that source be provided in the Allens Creek design?

A. In the NUREG-0619, section 8, entitled Staff Positions and Implementation, the Staff further discusses the acceptability of alternatives proposed by GE. One of the acceptable alternatives proposed by GE, i.e. item (4) of Section 8.1, NUREG-0619, states that:

> Only licensees of the following classes of BWRs will be permitted to immediately implement the GE recommendations to cut and cap the CRDRL nozzle

without rerouting the CRDRL (the option remains open to other licensees who can prove satisfactory system operation, return flow capability, and two-pump operation if necessary):

(a) 218-inch BWR/6 (see Appendix D)

(b) 251-inch BWR/6 (see Appendix D)

(c) 183-inch BWR/4 (see Appendix D)

(d) 251-inch BWR/4 (see Appendix D)

(e) 238-inch BWR/6 (based on BE letter MFN-285-79 dated November 27, 1971

(f) 218-inch BWR/4 (also based on GE letter MFN-285-79)

(g) 251-inch BWR/5 (based on GE letter MFN-089-80 dated May 2, 1980 - two-pump operation required)

Each of the applicable licensees will be required to demonstrate, by testing, concurrent two-CRD-pump operation (if necessary to fulfill required flow capacity), satisfactory CRD system operation, and required return-flow capacity to the vessel. Finally, each of these licensees, and those electing to reroute the CRDRL with subsequent valve-out, will be required to install the following modifications:

(a) Equalizing valves between the cooling water header and the normal drive movement exhaust water header.

(b) Flush ports at high and low points of the normal drive movement exhaust water header piping run if carbon steel piping is retained.

(c) Replacement of carbon steel pipe in flow stabilizer loop with stainless steel and rerouting directly to the cooling-water header.

In addition, all licensees and applicants, regardless of the particular type of modification

> selected, must establish operating procedures for achieving CRD flow to the reactor vessel equal to or greater than the boiloff rate of the base case discussed in Section 7.3 of NUREG-0619.

Q. Is the above Staff position applicable to Allens Creek?

A. Since Allens Creek are of 238-inch BWR/6 design having redundant CRD pump, the above Staff position is applicable. We therefore conclude that the proposed modification of the CRDRL for Allens Creek is acceptable pending implementation of the Staff's position and acceptance criteria outlined in NUREG-0619.

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

A. The Staff's resolution of NRC Generic Technical Activfity A-10 as applied to Allens Creek assures that the capability for high pressure injection sought by the contention is provided and the CRDRL is not needed to provide that function.

PROFESSIONAL QUALIFICATIONS

Vincent T. H. Leung

I am a system engineer in the Auxiliary Systems Branch, Division of Systems Integration, Office of Nuclear Reactor Regulation, U. S. Nuclear Regulatory Commission. I am responsible for the review and evaluation of the functional capability of the auxiliary systems employed at nuclear power plants. I have served in this capacity since April 1972.

From 1969 to 1972, I was a senior system engineer at Westinghouse Electric Corporation. I was responsible for technical plan review of steam and power conversion system, pressure vessels, piping systems, heat exchanger, stress analysis and wrote many computer programs to solve heat transfer problems relating to nuclear power plants.

From 1963 to 1969, I was a senior mechanical engineer at Avondale Shipyard, Inc. in New Orleans. I was responsible for floating vessels design, marine propulsion plants, offshore drilling platform and equipment design, piping system design and stress analysis. I also wrote many computer programs to perform heat balance evaluation for marine power plane, heat transfer and stress analysis problems relating to Marine Engineering.

I received a Bachelor of Mechanical Engineering degree in Mechanical Engineering from Royal Melbourne Institute of Technology, Australia in 1960 and a Master of Science degree in Mechanical Engineering from Tulane University in 1963.

I am a member of the American Society of Mechanical Engineers.