05/11/81

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
FELIX B. LITTON REGARDING FRACTURE TOUGHNESS

[TEXPIRG Contention 39]

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

A. My name is Felix B. Litton. I am a Senior Materials Engineer with the Materials Engineering Branch of the Division of Engineering. A copy of my statement of professional qualifications is attached.

Q. What is the purpose of this testimony?

A. The purpose of my testimony is to address TEXPIRG's contention relating to the fracture toughness of reactor vessel materials -- an issue identified by the NRC Staff as generic issue A-11. TEXPIRG's contention is set forth below.

> TEXPIRG contends the Staff's conclusion that ACNGS may be operated without endangering the health and safety of the public is premature with regard to reactor vessel materials. If the reactor is installed as planned now, resolution of generic issue A-11 will not be enforecable without such major disruption to service to make the plant virtually totally rebuilt. We contend that the process of embrittlement of the reactor vessel is not understood sufficiently well to justify the Staff position and that construction of the reactor vessel should not be started until the A-11 generic resolution is reached. In 44 Federal Register 18513, the NRC states it is considering amending its regulations specifying fracture toughness in 10 C.F.R. 50, Appendix G.

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Q. What requirements was the Allens Creek reactor vessel designed and constructed to comply with?

A. The reactor vessel for the Allens Creek Nuclear Generating Station, Unit 1, was designed and constructed to comply with the requirements of the ASME Boiler and Pressure Vessel Code, 1971 Edition, including Summer 1972 Addenda. Appendix G, "Protection Against Nonductile Fracture," of the Summer 1972 Addendum presents procedures for obtaining the allowable stresses during normal, upset and test conditions.

Q. What is the basis for these requirements?

A. Pressure vessels which are constructed to these requirements are expected to withstand stresses equal to or exceeding twice the normal design pressure under normal operating conditions. Appendix G of the ASME Code will be used to determine the pressure-temperature limits of the reactor vessel for all phases of reactor operation for the Allens Creek facility.

Q. What are the NRC requirements for fracture toughness?

A. The fracture toughness requirements for the reactor coolant pressure boundary are stated in Appendix G, "Fracture Toughness Requirements," of 10 C.F.R. Part 50. The purpose of Appendix G is to provide an adequate margin of safety during any condition of normal reactor operation, including anticipated operational occurrences and hydrostatic tests, to which the components in the pressure boundary may be subjected over their service lifetime.

Q. Does Allens Creek meet the requirements of Appendix G of 10 C.F.R. Part 50?

A. The Allens Creek facility is in compliance with the fracture toughness requirements of Appendix G of 10 C.F.R. Part 50 as well as all the applicable requirements of the ASME Code.

Q. Will the materials in the reactor vessel be monitored after operation commences to assure adequate margins of safety?

A. Appendix G of 10 C.F.R. Part 50 requires, in part, that the beltline materials in the reactor vessel shall have an unirradiated Charpy V-notch upper-shelf energy of 75 ft-lbs and that the properties of these materials shall be monitored by a surveillance program conforming to the "Reactor Vessel Material Surveillance Program Requirements" set forth in Appendix H of 10 C.F.R. Part 50. The measurement of the fracture toughness properties before and after irradiation provides assurance that an adequate margin of safety is maintained for reactor operation. The materials surveillance program for the Allens Creek facility is in compliance with Appendix H of 10 C.F.R. Part 50.

Q. What are the criteria for predicting irradiation damage?

A. The irradiation damage to the materials in the beltline region of the reactor vessel is related primarily to the copper content of the materials. In addition to the 75 ft-lbs Charpy V-notch upper shelf energy specified for the Allens Creek vessel, the copper content of the plate and weld metal is specified not to exceed 0.12 and 0.10 per cent, respectively. An end-of-life fluence of 4.5×10^{18} n/cm^2 would result in a referenced nilductility temperature (RT_{NDT}.) shift of no more than 100°F. and an upper-shelf energy of 20%, using Regulatory Guide 1.99, "Effect of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials," criteria for the predictions. The predicted irradiation damage to the Allens Creek reactor beltline materials is acceptable. The actual damage will be determined by the materials surveillance program.

Q. What caused the concern labelled as generic issue A-11?

A. An increase in the copper content of ferritic materials has been shown to decrease the fracture toughness properties of materials under neutron irradiation. The copper content of the weld metal in some of the older operating reactor vessels was inadvertently increased by the use of copper-coated welding wire electrodes during their construction. Specimens from the surveillance test program show these welds to contain up to 0.5 per cent copper and a reduction in the Charpy V-notch upper-shelf energy. The fracture toughness requirements of Appendix G of 10 C.F.R. Part 50 have not been violated in these reactors.

Q. If all the older reactors met the applicable requirements of Appendix G to 10 C.F.R. Part 50, what is the purpose of generic issue A-11 and is it applicable to Allens Creek?

A. The purpose of Task Action Plan A-11, "Reactor Vessel Materials Toughness," is to provide procedures for assessing the safety margin for failure prevention in reactor vessels possessing Charpy V-notch upper-shelf energies below the fracture toughness requirements of Appendix G of 10 C.F.R. Part 50. We conclude that the resolution of Task Action Plan A-11 is not a necessary requirement for the licensing of the Allens Creek Nuclear Generating Station, Unit 1. The Allens Creek facility is in compliance with the requirements of Appendices G and H of 10 C.F.R. Part 50 and Section III of the ASME Code, 1971 Edition, including Summer 1972 Addenda.

PROFESSIONAL QUALIFICATIONS

FELIX B. LITTON

I am a Senior Materials Engineer in the Materials Engineering Branch of the Office of Nuclear Reactor Regulation, Nuclear Regulatory Commission. I am assigned to the Inservice Inspection and Component Integrity Sections and my duties involve the review and evaluation of materials and processes used in the construction and operation of components in the nuclear power industry.

My education consists of a B. S. (1936) and M. S. (1937) degree in Physical Chemistry from Virginia Polytechnic Institute, Blacksburg, Va. I have completed additional study in Material Science at the University of New Mexico and have taken special courses in Fracture Mechanics and other job oriented courses at Union College and George Washington University.

Prior to joining the Nuclear Regulatory Commission, my experience consists of metallurgical research related to the preparation, fabrication and alloy formation of new structural materials for nuclear, advanced aircraft and high temperature application. I have published in technical journals on the environmental behavior, thermodynamic stability and mechanical properties of uranium, plutonium, vanadium, zirconium, tetanium, hafrium and silicon and their alloys. Although my primary experience in ferrous metallurgy has related to the cause of material failure in service, I have managed metallurgic research on welding and welding processes.