4-20-81

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 No. 1)	)	

TESTIMONY OF EUGENE C. ECKERT ON BEHALF OF HOUSTON LIGHTING & POWER CO. ON DOHERTY CONTENTION 7 - LPCI COLD SLUG

Q. Please state your name and job position.

A. My name is Eugene C. Eckert. I am employed by General Electric as Manager of NSSS Transient Performance Engineering. My business address is 175 Curtner Avenue, San Jose, California.

Q. Please state your educational background, work experience and professional qualifications.

A. A statement of my qualifications is attached to this testimony as Exhibit ECE-1.

Q. What is the purpose of your testimony?

A. The purpose of my testimony is to address Mr. Doherty's Contention 7, which alleges that:

> The design of obtaining Low Pressure Coolant Injection (LPCI) core spray water from the suppression pool following exhaustion of the condensate storage tank during Loss of Coolant Accident (LOCA), Reactivity Insertion Accident (RIA), or Transient Without Scram (ATWS) is an unnecessarily high risk to Petitioner's safety and environment interests because suppression pool water is colder than reactor coolant; hence when sprayed in the core it will increase core reactivity causing high temperature and increase possibility or actuality of fuel melt and formation of a critical mass.

Q. What is your understanding of the real concern expressed in this contention?

A. My understanding of Mr. Doherty's concern is that during either a Loss of Coolant Accident (LOCA), a Reactivity Insertion Accident (RIA), or an Anticipated Transient Without Scram (ATWS) the following are postulated to take place:

1. Reactor vessel pressure is reduced while the High Pressure Core Spray (HPCS) sprays condensate storage water onto the core.
2. The reactor pressure will be reduced sufficiently to allow the low pressure Emergency Core Cooling Systems (ECCS) to inject and/or spray on the core cold water taken from the suppression pool.
3. Core reactivity increases because the suppression pool water is colder than the reactor vessel water. Reactor power, in turn, increases causing high enough fuel temperatures to produce fuel melting.

Q. As a preliminary matter, will you briefly describe the Condensate Storage System?

A. The Condensate Storage and Transfer System (CSTS) mainly functions as a storage and transfer medium for the reactor and turbine generator primary fluid during normal, abnormal and shutdown operating conditions. During abnormal operating conditions, which are of interest in this contention, the system provides total reserve condensate capacity required for operation of the Reactor Core Isolation Cooling (RCIC)

system and the HPCS system and to provide adequate suction pressure for the RCIC, HPCS, and Control Rod Drive (CRD) pumps. The system is designed to operate in the temperature range of 40°F to 120°F. The temperature of the water in the system is chiefly determined by outside ambient temperature.

Q. Will you also describe the function of the Suppression Pool System during abnormal accident conditions?

A. The suppression pool contains a sufficient supply of water to satisfy the cooling water requirements of the low pressure ECC systems and the Residual Heat Removal System (RHR) during all modes of operations. The suppression pool also serves as a heat sink for steam release from the Nuclear Pressure Relief System (NPRS). The in-place volume of water in the suppression pool is designed to provide the short term energy sink for a LOCA.

Under normal plant operating conditions, the suppression pool water temperature will be in the range of 90°F, approximately the same as the containment vessel atmospheric temperature. During a LOCA, the pool water temperature may rise to 170°F shortly after the initial release of energy to the pool from a postulated large circumferential break of a line inside the drywell. The long term release of core decay heat after this initial release of energy may result in pool temperatures as high as 185°F.

Q. When the low pressure ECC Systems are called upon to inject water in the vessel, is the suppression pool water hotter than the condensate storage water?

A. The temperature range of operation for the two water sources overlap extensively. However, since the suppression pool serves as a heat sink for steam release before the low-pressure ECC systems operate, the suppression pool would generally be above the temperature of the condensate storage water at the time the low pressure ECC system operates.

Q. Will you explain how ECCS injection affects reactor physics?

A. An understanding of how water injection changes reactor physics must begin with the concepts of reactivity and criticality. For the fissioning process to be self-sustaining, as a minimum, the number of neutrons born from each fission reaction and surviving to cause another fission must be constant with time. When this minimum condition is achieved the reactor is said to be critical with an effective multiplication factor, K_{eff}, equal to one. When $K_{eff}>1$ the reactor is said to be supercritical and the neutron population is increasing with time. For $K_{eff}<1$, the reactor is said to be subcritical with the neutron population decreasing.

The shutdown margin of the core is derived from the term K_{eff}. Shutdown margin (SDM) = $1 - K_{eff}$ and is used as a measure of how subcritical the core is. Shutdown margin can also be expressed in terms of how much reactivity must be added to reach criticality. Reactivity is a measure of the effect on the neutron population of any change to the reactor core. The addition of reactivity increases neutron population. The removal of reactivity decreases neutron

population.

For a BWR, a value is placed on the minimum amount of shutdown margin which must be maintained when the reactor is subcritical. This minimum is determined once the final core enrichments are determined and is based on the core remaining subcritical at cold shutdown even with the control rod producing the greatest reactivity addition completely withdrawn.

Q. Please explain how the temperature of the water being injected during ECCS operation effects reactivity.

A. The four parameters which affect reactivity in a BWR are the position of the control rods, the temperature of the core coolant water, the temperature of the fuel, and the steam void concentration of the core. The water being injected by ECCS operation is mixed at its injection location and subsequently affects the temperature of the core coolant water and the void concentration in the core. These effects result in either a reactivity increase or decrease depending on reactor conditions. For most reactor operating conditions, initiation of the ECCS will cause reactivity to decrease due to a reduction in steam leaving the vessel, resulting in a small reactor pressure decrease, an increase in core voids, and a power reduction. For a narrow range of low power reactor conditions, ECCS injection can cause a reactivity increase due to a decrease in reactor coolant inlet temperature. This increases the density of the steam-water mixture, generally referred to as the moderator, and thus results in an increasing neutron population by reducing the number of

neutrons which leak from the fuel cell and undergo non-fissioning absorption in a control rod or some core structure material. The result is a reactivity increase.

Q. When is moderator density maximum?

A. Moderator density is maximum when the reactor is in the cold shutdown mode of operation (Temperature = 70°F, Pressure = 1 atmosphere). Any ECCS injection prior to normal cooldown to cold shutdown and after scram only serves to accelerate the rate at which cold shutdown is reached. The reactor is designed to maintain subcriticality ($K_{eff}<1$) even when ECCS injection accelerates the cooldown process. For those events in which ECCS operates without the reactor being scrammed, a analysis must be done to quantify the reactivity insertion rate and the effect on fuel temperature.

Q. Has GE performed such an analysis?

A. Yes, GE has done such a detailed analysis.

Q. For three events listed in the contention, would you describe the sequence of events affecting reactor operation and fuel temperature?

A. For LOCA, the core is brought to a subcritical state ($K_{eff}<1$) by insertion of the control rods before any ECCS injection starts. The addition of cold ECCS water aids in reflooding the core and makes core reactivity less negative since it increases moderator density. However, the moderator density is still less than its value at cold shutdown, thus the shutdown margin is maintained and the core remains subcritical. The reactor is brought to a cold shutdown state

after core reflood using the Residual Heat Removal (RHR) System. This System circulates water from the suppression pool to the vessel and maintains water level while removing decay heat in the RHR heat exchangers.

For RIA, which is any rapid increase in reactivity other than those increases due to expected plant conditions, there is no ECCS operation. The most limiting RIA for Allens Creek is the Rod Drop Accident. For this accident the reactor will either be in a cold shutdown condition, in which case the reactor will be subcritical with sufficient shutdown margin to remain so even if the maximum worth control rod is rapidly withdrawn from the core, or there will be sufficient forced circulation from the recirculation system and water supply from the feedwater system to handle any reactivity insertion without the aid of any cold water ECCS injection.

For ATWS, which is an event which combines the failure of the normal scram function of the control rod drive system with an anticipated transient, the reactor is shutdown by tripping the recirculation pumps in combination with either an alternate method of control rod insertion or by the injection of boron into the vessel via the standby-liquid control line. Only in the case of boron injection would the ECCS operate before the reactor was brought to a subcritical state and then only the HPCS would be utilized since the vessel pressure is well above the pressure injection point for the low pressure ECC Systems. General Electric has

analyzed such a case with boron injection and found that the reactivity increase due to HPCS injection is negligible. (See NEDO 24222.) For HPCS injection after the reactor has been brought to a subcritical state, the amount of reactivity increase would be insufficient to cause criticality for the same reasons discussed above concerning LOCA. Long term cooling and eventual cold shutdown is provided by the RHR System. In this mode, the System withdraws reactor water from the vessel, passes it through heat exchangers to remove core decay heat, and then returns it to the vessel. This mode of RHR operation is not part of ECCS and does not involve any cold water injection.

Q. Is there any event in which injection of ECCS water occurs without the reactor being scrammed other than an ATWS?

A. There is no plausible event which would cause this to happen. However, General Electric has done an analysis in which they have considered inadvertent initiation of ECCS during normal low power operation. Having the reactivity insertion at low power maximizes the reactivity insertion rate and thus the rate of fuel temperature increases. (ECCS injection at high power will result in a reactivity decrease.) This injection results in gradual drop in the core coolant inlet temperature and a simultaneous increase in core power. When the power level reaches the Intermediate Range Monitor (IRM) Scram setpoint, a scram is initiated which terminates the power increase and prevents fuel damage. This event is

much less severe than the design basis rod drop accident (See PSAR Section 15.1.35 and previous discussion on RIA) since the rate of reactivity insertion due to the cold water injection is much less than that due to a rod drop event.

Q. What are your conclusions?

A. The ECC Systems do inject water which is colder than the reactor coolant and therefore in some cases can increase reactivity by increasing moderator density. However, in no case will core reactivity be increased in an amount to cause fuel damage.

Exhibit ECE-1

EDUCATION AND PROFESSIONAL QUALIFICATIONS

Eugene C. Eckert

Mr. Eckert received a Bachelor of Science Degree in Electrical Engineering from Valparaiso University in Indiana in 1958. During the next year, he attended Standford University under an Oak Ridge Fellowship and received the Master of Science Degree in Engineering Science in August, 1959.

Immediately upon joining General Electric Company in September, 1959, Mr. Eckert participated in a company-wide engineering training program. His work assignments in this program included large jet engine control design, aircraft nuclear propulsion control analysis, nuclear submarine kinetic and control analysis, and industrial control simulation analysis at GE's Research and Development Center. After completing this program in 1962, Mr. Eckert joined General Electric's Nuclear Energy Division to work on Boiling Water Reactor (BWR) simulation and dynamic analysis. He has been responsible for design and licensing documentation of the dynamic analysis for several GE BWR's and has participated in initial startup testing of many of the units. He led the dynamic design efforts which established the BWR/4 product line, culminated in 1974 by the startups of the Browns Ferry (TVA), Peach Bottom (PECO) and Fukushima-2 (Japan) units. Since then, his design and analysis capabilities have been applied in all BWR product lines. He has been lead total plant design engineer and, since 1971, manager of transient analysis for BWR's.

In his current position, he is responsible for establishing

the simulation requirements of the computer models needed to perform transient analyses, development of design procedures evaluation of BWR stability, and evaluation and specification of the functional protection systems required for reactor abnormal transient protection. Included is the analysis and mitigation of transients with postulated failure of reactor scram (ATWS). Plans for new product lines are evaluated, and all projects are carried through plant startup until turnover to the utility.