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July 19, 1983

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: LaSalle County Station Units 1 and 2
NPF-11 License Condition 2.C.(19)
Additional Instrumentation and Control
Concerns
NRC Docket Nos. 50-373 and 50-374

Reference (a): C. W. Schroeder letter to H. R. Denton
dated July 6, 1983.

Dear Mr. Denton:

On June 21, 1983, Commonwealth Edison representatives C. W. Schroeder and George Crane, et al met with Dr. Bournia, et al of your staff to discuss potential multiple control system failures due to High Energy Line Break events. As a result of that discussion, Commonwealth Edison Company provided revised responses to NRC Questions 031.290 and 031.292 in Reference (a).

Following further discussion with Dr. A. Bournia, et. al, we have further revised our response to Question 031.292 to address their concerns. The revised response is enclosed for your review.

To the best of my knowledge and belief the statements contained herein and in the attachment are true and correct. In some respects these statements are not based on my personal knowledge but upon information furnished by other Commonwealth Edison and contractor employees. Such information has been reviewed in accordance with Company practice and I believe it to be reliable.

Enclosed for your use are one (1) signed original and forty (40) copies of this letter and enclosures.

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If there are any further questions in this matter, please contact this office.

Very truly yours,

C W Schroeder 7/19/83

C. W. Schroeder
Nuclear Licensing Administrator

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Enclosures

cc: NRC Resident Inspector - LSCS
Dr. A. Bournia (Fed. Express)

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July 19, 1983 (Revised)

Revised Response to NRC Question 031.292

2) Common Impulse Lines

The limiting non-safety control-system instrument line failure is postulated to be the common sensing line containing two of the three reactor differential pressure (level) transmitters for feedwater control (C34-N004B, C34-N004A). This failure occurs in the annular open area of the reactor building (zone 4A). A failure of this instrument line causes the transmitters to read low and it is assumed that the high water level (L8) is non-operative, thus main turbine trip and feedwater turbine trips are disabled. Failure of both these feedwater control channels would not affect feedwater flow if it were operating on transmitter C34-N004C, however, a worst failure assumption requires that feedwater failure occurs on either N004B or N004A. The failed transmitters would output a minimum water level, thus resulting in a feedwater control system demand to increase flow to the maximum.

With these worst case assumptions, there would be no high water level trip of the main turbine nor of the turbine-driven feedwater pumps. The main turbine is not affected directly by this failure of an impulse line in the reactor building because it is in the turbine building. The turbine control system is located in the auxiliary electric equipment room (zone C-1) and the steam flow sensors controlling the turbine bypass system (EHC) are located in the basement of the turbine building. Consequently, with this separation of equipment, as power is increased due to higher feedwater flow (and water level), the turbine bypass valves will open due to the steam flow mismatch, thus bypassing 25 percent of rated steam flow directly to the condenser.

The maximum turbine power is limited in this non-controlled flow situation to 95 percent rated power by the APRM neutron flux thermal power trip which has a maximim limit of 120 percent power; ie, 120 percent limit minus the 25 percent bypass energy (flow).

As the feedwater flow continues and the water level in the vessel reaches into the main steam lines, the main turbine begins to vibrate, thus causing a turbine trip via turbine stop valves. A turbine stop valve trip also scrams the reactor.

The two turbines driving the feedwater pumps also trip due to high vibration from fluid carry-over. With the trip of the turbine-driven feedwater pumps, the feedwater flow coasts down to a preset level of about 30 percent (motor driven feedwater pump). The operator will have data available from the third feedwater level sensor (C34-N004C) indicating an increase in water level that is both unexpected and continuous due to the feedwater controller failure. The operator can attempt to runback feedwater or if that fails, he can manually trip the turbine to protect it from damage due to the increase in carry-over. This operator action would most likely occur prior to the main turbine trip due to high turbine vibration. The main turbine trip will initiate scram and recirculation pump trip (RPT) via safety-related logic. Closure of the turbine stop valves results in a RPV pressure spike with consequent opening of the safety-related safety/relief valves (SRV) to control vessel pressure. SRV action assures pressure control of the vessel. No fuel failures result from this transient which is fully treated in Section 15.1.2 of the FSAR.

With the L8 failure, the motor-driven feedwater pump continues to raise the water level, assuming no operator actions after scram to terminate feedwater flow or to depressurize the vessel, until it reaches the main steam nozzles and starts to fill the steam lines. Such water would eventually be discharged through the SRV's into the suppression pool if operator action were not taken to control water level and/or vessel pressure. High water level alarms (L8) from the HPCS level sensors and from the RCIC level sensors are available independent of the L8 alarm from the feedwater control system to inform the operator about the water level condition. Station procedures call for operator action to backup the high level trip at L8 for positive water level control. Detailed Emergency Procedural Guidelines (EPG's) address liquid overfill conditions and appropriate operator corrective actions: termination of liquid injection and removal of excess liquid (recover to less than L8 level) via the steam-line drains and the Reactor Water Cleanup System.

A probabilistic evaluation of high pressure liquid challenge to the Crosby SRV and piping was performed for the BWR Owners Group of which LaSalle was a member, in April 1981 by Science Applications Inc. That report (SAI-245-81-PA by F. Leverenz and D. Harris and others) was discussed at an NRC meeting on March 10, 1981; see NRC Memo T. Speis from W. Hodges and memo from T. Speis to R.J. Bosnak dated May 13, 1981. It showed that the probability of getting some liquid to the steamline and hence to the SRV's is approximately 10^{-2} per reactor year; and, given the existence of subcooled water upstream of the SRV's, that the

probability of rupturing the discharge line is conservatively 7 x 10^{-4} per event. The combined probability is therefore 7 x 10^{-6} for the rupture of the entire discharge line, which is no greater than for a steamline break inside containment. For only that part of the SRV discharge line within the wetwell volume of primary containment, the combined rupture probability is reduced further to about 2 x 10^{-6} because only part of the discharge piping is within this volume.

This hypothetical event is essentially a Feedwater Flow Controller Failure - Maximum Demand event (FSAR Chapter 15.1.2) that is bounded by the Turbine Trip Without Bypass case.

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