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February 3, 1981

Mr. James G. Keppler, Director
Directorate of Inspection and
Enforcement - Region III
U.S. Nuclear Regulatory Commission
799 Roosevelt Road
Glen Ellyn, IL 60137

Subject: LaSalle County Station
NRC Inspection Report
50-373/80-48 and 50-374/80-30
NRC Docket Nos. 50-373/374

Dear Mr. Keppler:

In response to the subject inspection report transmitted by your letter dated January 9, 1981, attached are replies to the apparent items of noncompliance in the Notice of Violation. The attached replies include our evaluation of quality assurance program and management control system improvements which will be implemented to preclude further violations of this type.

The primary reason for the violation was inadequate followup of corrective actions identified in our reply to your previous inspection report 50-373/80-20 and 50-374/80-13. This inadequate followup occurred because the LaSalle County Project Construction Management did not recognize their responsibility to followup their contractor's design control corrective actions. This was the only LaSalle County Construction Management controlled contractor with extensive design and analysis responsibility. Design and analysis are normally handled by contractors controlled by the LaSalle County Project Engineering organization; therefore, Construction Management incorrectly assumed the design and analysis corrective actions would be followed by Project Engineering. This lack of responsibility for control of contractor design activities is unique to this specific contractor.

We agree that our followup was not adequate to assure timely corrective actions to deficiencies identified in the vendor quality assurance program by the NRC. As we stated in our meeting on January 29, 1981, Commonwealth Edison had performed an audit of the vendor in May, 1980, in which deficiencies were identified and had scheduled a reaudit of the vendor in November, 1980 to take steps to correct his inadequate response to date. Although our followup was not timely, it did not represent a breakdown in our Quality Assurance program.

Your Inspection Report does not discuss the basis you used to determine the severity level of this violation; however, in a meeting on January 29, 1981, you explained the severity level was increased for repeating a previous violation. We stated in the January 29, 1981 meeting that in our opinion the noncompliance cited should not be considered a repeat noncompliance under the new enforcement policy because it was the first occurrence during the period of applicability of the new enforcement policy. Therefore, we respectfully request your reconsideration of considering the January 29, 1981, meeting as an enforcement meeting and the appropriate reassignment of severity level.

Very truly yours,

C. Reed

C. Reed
Vice President

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Appendix A

RISK ASSESSMENT

An estimate of the core uncovery risk from a break in the SDV system piping (at a plant like BF-3) might be calculated as follows:

$$P_o = P_1 \times P_2$$

where,

P_o = Probability of Core Uncovery/Rx/Yr

P_1 = Probability of an unisolated SDV break/Rx/Yr

P_2 = Probability of core uncovery following an unisolated SDV break

where,

$$P_1 = (N \times P_{11}) \times (P_{12} + P_{13})$$

N = Number of Rx scrams/Rx/Yr

P_{11} = Probability of an SDV Break (>> sump pump cap)/Rx scram

P_{12} = Probability of not being able (RPS or control air condition) to immediately reclose scram valves after a Rx scram/Rx scram

P_{13} = Probability of not reclosing (human or procedural) or being unable to reclose (break consequences) scram valves after an SDV break.

If we assume:

$N = 2$
$P_{11} = 10^{-4}$
$P_{12} = 10^{-1}$
$P_{13} = 10^{-1}$

$P_2 = 0.25$

then $P_o = 10^{-5}$

Discussion

Based on BWR operating experience it would not be unreasonable to assume that at least two reactor scrams (from full pressure and temperature) occur every year at each plant. It might also be assumed that a break in a small line in the SDV system (downstream of the scram outlet valves and upstream of the SDV system vent and drain valves), resulting in a substantial blowdown rate*, (>> 100 gpm) can occur once in every 10,000 BWR reactor scrams. (A blowdown rate of this magnitude could result in eventual loss of the emergency makeup systems if not isolated.) For BWRs it also seems reasonable to assume that out of every ten reactor scrams, one would involve a RPS trip condition or power supply failure or a loss of control air supply such that the scam outlet valves would not be able to be reclosed for an indefinite period of time. Furthermore, should a break in the SDV system occur, the additional abnormal plant symptoms and reactor system process conditions indicated in the control room could divert and continue to occupy the control room operator's time and attention (e.g., reactor water level drop) which could result in the scram valves being left open. The break itself may also introduce additional failure modes to the break isolation arrangements (e.g., air line failure due a postulated pipe whip of a ruptured SDV system line, environmental damage to the detection equipment, damage to the scram valve teflon seating surfaces caused by prolonged blowdown). We would estimate that considerations such as these could contribute an additional one chance in ten of not isolating a break in the SDV system.

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* Note: A break from a one inch Schedule 160 vent line is capable of passing approximately 400 gpm at 1,000 psi, while a two-inch Schedule 160 drain line is capable of passing approximately 1,500 gpm.

Finally, in the event of such an unisolated break in the SDV system, we would assume that there is a 75% chance that at least some ECCS equipment in the Reactor Building basement and emergency makeup inventory will be available to keep the core covered continuously and indefinitely even though none of the equipment is qualified for environmental conditions including flooding.

Although the above point estimate is considered to be 10^{-5} /Rx/Yr, which would make this event a significant contributor to risk, the uncertainty range may be such that the uncovery probability most likely lies within the range of 10^{-3} /Rx/Yr to 10^{-9}/Rx/Yr. Consequently it is difficult to conclude on the basis of these numbers alone that the existing plant design configuration is safe, i.e., less than 10^{-6}/Rx/Yr.

If from these convolutions one were to conclude that the SDV pipe break is a significant contributor to BWR core uncovery risk, it is believed that the risk can best be reduced by decreasing the likelihood of a break in the SDV system piping by an appropriate upgrading of the SDV system mechanical integrity assurance basis. The risk can also be reduced in a significant although less favorable or desirable way by improving the reliability of the break isolation arrangements.