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SUBJECT: Discusses threat to operability of seven inboard containment valves from postulated high energy line break jet impingement per NRC request. Probability of accident not increased, based on safety of existing plant design.

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SEP 17 1986

Director of Nuclear Reactor Regulation
Attention: Ms. E. Adensam, Project Director
BWR Project Directorate No. 3
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington DC 20555

SUSQUEHANNA STEAM ELECTRIC STATION
JET IMPINGEMENT ON CONTAINMENT
ISOLATION VALVES
FILE R41-2
PLA-2702

Docket Nos. 50-387
50-388

Dear Ms. Adensam:

Pennsylvania Power & Light Company is forwarding this information to you at the request of our Region I senior resident inspector.

A threat to the operability of seven inboard containment isolation valves from a postulated high energy line break jet impingement has been identified on each of the Susquehanna SES Units. The breaks involved are at the terminal end welds of five pipes which join the reactor pressure vessel. These lines are in two categories, one 28 inch reactor recirculation suction line and four 12 inch reactor recirculation jet pump supply lines. The breaks are at the pipe-to-safe-end welds which are located in the inside diameter of the sacrificial shield door. These doors deflect the break jets and tend to focus them on the containment isolation valves. This is an apparent violation of General Design Criteria 4 (GDC4) and FSAR commitments.

The initial discovery was made during a routine review of the reactor recirculation calculations for postulated pipe breaks and affected the "A" and "D" inboard MSIV's. Nonconformance Reports (NCR) were immediately written and the investigation was expanded to include piping with similar modeling geometry. The remaining inboard MSIV's and three other inboard containment isolation valves were also identified as potentially threatened valves. NCR's have also been written for these valves.

A more detailed analysis was pursued for the postulated reactor recirculation suction line break without success. It was then decided to pursue a leak - before - break (LBB) analyses to show that the postulated breaks are not credible. This decision was based on the fact that a precedent has been established for this approach and it is considered to be a more realistic mechanism of failure. This approach would cover all the postulated pipe breaks in question and eliminate the need for modifications. The leak-before-break analyses are presently in progress.

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PP&L is continuing to operate both units at Susquehanna in accordance with our procedures. A safety evaluation has been performed justifying continued operation. The evaluation identified the assumed double-ended-guillotine-break (DEGB) as a non-mechanistic assumption since large nuclear piping has no history of this type of failure. In addition, the quality control used during construction, the Pre-Service Inspections (PSI) performed, the In-Service Inspections (ISI) performed and planned, the existing leak detection systems, the good water chemistry controls, the Induced Heating Stress Improvement (IHSI) performed on both units with attendant reinspection, the corrosion resistant cladding on the jet pump supply lines, and the relatively short service life of both units all raise the confidence level that an actual DEGB would not occur at Susquehanna. PP&L has calculated the probability of a DEGB failure along with a random failure of one of the corresponding outboard containment isolation valves to be less than 1.2×10^{-6} events per year.

In the event that a DEGB were to occur at one of the identified welds and assuming the impacted valves would fail to close, the outboard isolation valve could reasonably be expected to provide containment isolation. If an unlikely random failure of the outboard isolation valve were to occur simultaneously it would be reasonable to assume that the affected system's integrity would be maintained and serve as an effective containment barrier. Additionally, in all cases except for the MSIV's, if containment were lost, the leakage would be into the reactor building and subject to Standby Gas Treatment System processing. In the case of the MSIV's, a Design Basis Accident (DBA) using the MSIV's and main steam lines as the release pathway would yield Peak Cladding Temperatures of less than 2200°F. This would not result in significant fuel damage or the release of radioisotopes of significant concern.

The probability of an accident occurring has not changed as a result of the described discovery. In fact, upon completing the LBB analysis, PP&L expects to conclusively show that the probability of a DEGB at the welds in question will be virtually nonexistent. A large margin of safety is inherent in the existing plant design, the consequences of an accident as described above do not exceed those previously reported in the FSAR, and the margin of safety as defined in the basis of the technical specification has not been reduced.

If you have any questions please contact us.

Very truly yours,

H. W. Keiser
Vice President - Nuclear Operations

cc: M. J. Campagnone NRC
L. R. Plisco NRC