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SCHWENCER,A. Licensing Branch 2

SUBJECT: Provides status of evaluation & results re Humphrey Issues 3.1 & 3.3 on steam condensing mode of RHR sys.None of postulated loads from issues compromises plant design.

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APR 0 6 1983

Director of Nuclear Reactor Regulation
Attention: Mr. A. Schwencer, Chief
Licensing Branch No. 2
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

SUSQUEHANNA STEAM ELECTRIC STATION
HUMPHREY ISSUE 3.0
ER 100450 FILE 172 Docket Nos. 50-387
PLA-1584 50-388

REFERENCES: (1) PLA-1542 dated February 28, 1983
(2) PLA-1375 dated November 1, 1982

Dear Mr. Schwencer:

In reference (1), Pennsylvania Power & Light Co. informed you that our final response to Humphrey Issues 3.1 and 3.3 would be submitted by June 1, 1983, as opposed to the original submittal date of March 31, 1983 as stated in Reference (2). In recent discussions with your staff, they informed us that Mississippi Power & Light (MPL) had committed to not using the steam condensing mode (SCM) of the RHR system until they submitted and received approval from the NRC of their final response to all Humphrey concerns related to the SCM operation. Because of MPL's decision to temporarily "lock-out" the SCM, your staff requested that PP&L provide the status of our evaluation and results to date. Per our discussion with your staff, we informed you of the following:

In order to quantify the hydrodynamic loads caused by a postulated steam discharge through the RHR SRV, we believe the steam mass flux time histories need to be determined, since the type of load generated varies with the mass flux time history (i.e.: lateral tip load, air clearing, etc.). To accomplish the first step, Bechtel used the RELAP 5 code to model the thermo-hydraulic characteristic of the SCM. Once the proper valve characteristic, control volumes and plant parameters were inputted, the analysis was performed for two separate failure cases:

- Failed open steam inlet pressure control valves (PCV's)
- Failed open RHR SRV

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For the first case, the RHR SRV cycles open and closed with a cycling frequency depending on the depressurization and re-pressurization rates downstream of the PCV. At lower reactor pressures, the SRV cycling frequency is low enough to consider possible reflood and subsequent water/air clearing loads.

For the latter case, the SRV opens without cycling and the steam mass flux through the SRV depends on the reactor pressure, condensation rate in the RHR heat exchangers (Hx), and other system parameters.

For both cases, we considered a range of reactor pressures and system parameters, as well as condensation in the RHR Hx (normal operation) and no condensation in the Hx (start-up of SCM) to determine the upper and lower bounds of the steam mass fluxes through the RHR SRV. The analysis indicated the steam flux can be as high as 114.4 lbm/ft^2-sec or as low as 0.36 lbm/ft^2-sec.

The results of the above analysis were then used to develop bounding forcing functions to evaluate the plant design.

For the cycling SRV case (failed open PCV's), the RHR SRV discharge line (SRVDL) was evaluated for the reflood and water clearing loads and found acceptable. In addition, the RHR SRVDL supports have been evaluated and found acceptable.

In addition, the subsequent air bubble load causes loads on the adjacent submerged structures and liner plate. These components have been evaluated and found acceptable.

As stated before, our thermo-hydraulic analysis indicated that steam mass fluxes can occur in the medium steam mass flux range or chugging regime. This results in a lateral tip load at the RHR SRVDL exit. We developed and specified a tip load using a statistical approach which paralleled the methodology imposed by the NRC to specify the single vent lateral load on the main vent downcomers. The RHR SRVDL has been evaluated for this load and found acceptable. The RHR SRVDL supports have also been evaluated and found acceptable.

In conclusion, our evaluations to date indicate that none of the postulated loads from Issue 3.1 and 3.3 compromise the SSES plant design.

Very truly yours,

N. W. Curtis
Vice President-Engineering & Construction-Nuclear

paf/ltc219a:sah

cc: R. L. Perch NRC

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