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Dr. Thomas E. Murley
Regional Administrator, Region 1
U.S. Nuclear Regulatory Commission
631 Park Avenue
King of Prussia, PA 19406

SUSQUEHANNA STEAM ELECTRIC STATION
TURBINE BYPASS TRANSIENT
ER 100508 FILE 841
PLA-2229

Docket No. 50-388

Dear Dr. Murley:

As a followup to our meeting with your staff on May 31, 1984, attached is a detailed description and evaluation of the turbine bypass transient which occurred at Susquehanna SES Unit 2 on May 28, 1984. Also attached is the NSAG report on the transient.

Very truly yours,

B D Kenyon

B. D. Kenyon
Vice President-Nuclear Operations

Attachment

cc: R. W. Starostecki - NRC Region 1
R. H. Jacobs - NRC Resident Inspector
R. L. Perch - NRC Bethesda

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Attachment 1
PLA-2229

Turbine Bypass Transient

Following shift turnover on May 27, 1984, Operations personnel began establishing plant conditions to perform the RCIC controller hot functional tune-up test (HF-250-010). The EHC pressure setpoint was reduced from an initial value of approximately 930 PSIG to 920 PSIG at the completion of CRD scram time testing at 00:10 on May 28, 1984. Control rods were then withdrawn to increase reactor power to open the #1 turbine bypass valve to approximately 50% to ensure adequate pressure control during the RCIC hot functional test.

Steam dilution flow to the secondary Steam Jet Air Ejector (SJAE) began to oscillate (+/-150 lb/hr) at approximately 23:50 based on strip chart recorder data. This corresponds to the decrease of the EHC pressure setpoint. At approximately 00:45 the SJAE steam flow increased from 9400 lb/hr to 9650 lb/hr as reactor pressure increased. The flow then decreased to 9400 lb/hr as reactor pressure decreased.

At approximately 01:00 a control rod was selected and withdrawn. This action directly preceded an increasing amplitude oscillation of turbine bypass valve position as the #1 bypass valve re-positioned to maintain pressure setpoint. The oscillation induced reactor pressure and bypass steam flow perturbations which apparently induced offgas system steam dilution flow oscillations. Variations in the offgas system flow rate would have a negligible effect on reactor pressure and would not be expected to influence EHC operation. The offgas system oscillations were further complicated by the fact that two root valves to the offgas main steam pressure reducing controller (PC-20701A) were isolated causing the pressure control valves to be open fully. A review of previous oscillations in the SJAE steam flow showed two oscillations occurred on May 27, 1984 when reactor pressure was adjusted. Had the pressure regulating valves been operable, the oscillations would have been dampened. Offgas isolated on low steam dilution flow at approximately 01:01:03 due to the oscillations in the steam flow caused by the increasing magnitude of the oscillations in the total bypass valve capacity. This generated the first indication to the plant control operator (PCO) that an abnormal plant condition existed. Subsequent engineering simulations of the resulting transient support the supposition that the #2 turbine bypass valve opened at 01:01:15 which pressurized the #2 bypass valve discharge piping and caused narrow range reactor pressure to drop to 896 PSIG. Data in the attached Figure 1 shows an expected reactor pressure drop of 20 pounds based on main steam line surge flow when two bypass valves initially open. This correlates well with the recorded pressure decrease of 19 pounds and recorded steam flow increase. As reactor pressure decreased, vessel level swelled to approximately +53" due to increased void formation and a HPCI high level trip signal was generated. Process computer data indicates the bypass valves went fully closed to recover reactor pressure. Vessel level dropped to 30" and the feedwater system, in automatic vessel level control using the low load controller (LIC-2R602), responded to the level perturbation by increasing feedwater flow. The effect of reactor pressure increasing due to the bypass valve closure provided a net reactivity increase and subsequent small power spike.

The maximum APRM power level recorded in the Shift Supervisor and Startup Test logs during the transient was 10% on APRM 'E'. This APRM had a gain adjustment factor of 1.85 as determined by Startup Test (ST) 12.1 which would indicate an actual power level of 5.4% rated power. This is supported by the fact that a rod block was received, however no reactor scram signals were generated. Based on APRM settings for a rod block at 11% indicated power and scram function at 14% indicated power, and applying the current gain adjustment factors for individual APRM's, the minimum power at which a rod block would occur was 4.6% actual power. The minimum power at which a half scram would occur was 5.9% actual power. This would indicate that the actual power level reached during the transient is bounded by 4.6% to 5.9% rated power. A separate transient power analysis using the actual IRM recorder traces, as adjusted by a Hot Functional test which correlated observed IRM readings to the calibrated APRM readings, indicates the initial power level was 3.2% and that actual power reached during the transient was between 4.7% and 5.0% rated power.

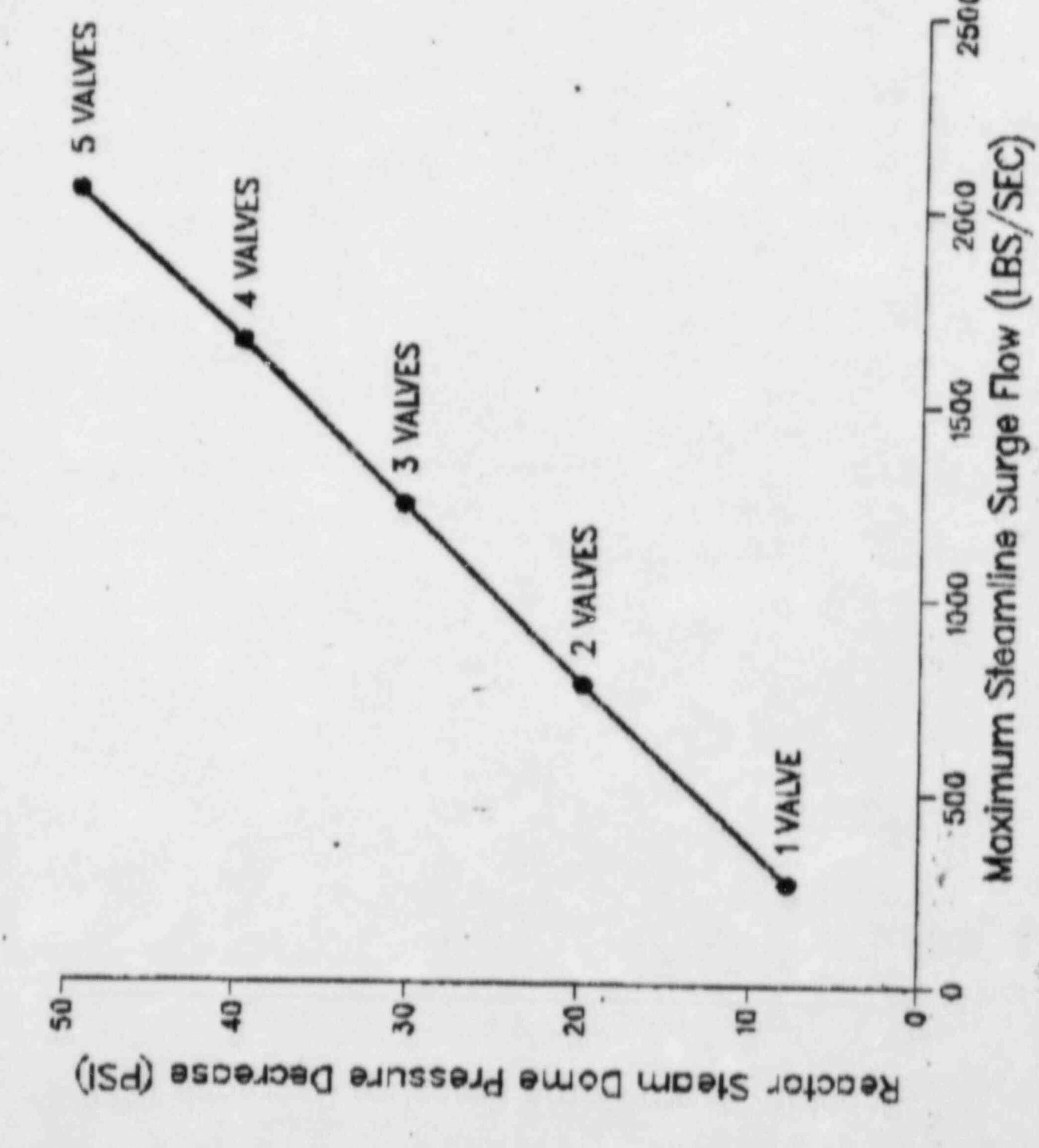
Reactor pressure continued to increase to a maximum value of 925 PSIG at 01:01:48 at which point turbine bypass valves #1 and #2 re-opened to 34% of total bypass valve capacity. Reactor pressure decreased to 911 PSIG and both bypass valves reclosed at 01:01:53 to regulate pressure based on the pressure setpoint. The #1 bypass valve re-opened at 01:02:03 and oscillated three times while attempting to stabilize reactor pressure. When pressure again exceeded the pressure setpoint at 01:02:58 the #2 bypass valve re-opened and a maximum of 28% total bypass valve capacity was recorded by the process computer. Maximum reactor pressure reached prior to the second bypass valve operation was 924 PSIG. Figure 2 graphically depicts the total bypass valve open position and narrow range reactor pressure response during the transient. At 01:03:09 the #2 bypass valve reclosed and the #1 bypass valve stabilized reactor pressure at approximately 918 PSIG. Minor reactor vessel level swings and a power transient, less severe than the initial power spike, also occurred during these subsequent bypass valve operations.

To mitigate the transient, Operations personnel began to insert control rods and placed the feedwater low load controller in manual to minimize further vessel level oscillations. Plant conditions stabilized at approximately 01:04 and the offgas system hydrogen recombiner was returned to service. Shift supervision, believing the offgas system had initiated the transient, then directed the PCO to manually isolate the offgas system to preclude further transient operation. At 01:05:19 the main steam supply (HV-20701A/B) to offgas was isolated. At 01:15 the Unit #2 recombiner was shutdown and the mechanical vacuum pump was placed in service. HP-250-010 and ST 14.2 (RCIC Quick Start to the Vessel) were satisfactorily completed at 04:58.

Subsequent plant staff I&C and General Electric investigation of the EHC pressure control and bypass valve control logic indicated that all control functions were operating normally. Plant staff Mechanical Maintenance investigation of the #1 bypass valve revealed that a chipping hammer was found wedged between the bypass valve seat and the valve disc preventing the #1 bypass valve from fully closing. Since the valve indicated fully closed on two occasions during the transient, it can be reasonably assumed that the hammer became lodged in the valve either during the final stages of the transient or when plant conditions had stabilized following the transient. The presence of the hammer in the vicinity of the #1 bypass valve may have impeded the steam flow rate through the valve, possibly producing the pressure increase which precipitated the initial observed bypass valve oscillation. The chipping hammer that was retrieved was the type typically used during welding procedures. The head of the hammer was 6" wide and tapered to a point with a chisel design at the opposite end. Apparently the bypass valve, upon closing, severed the spring portion from the handle and the spring was lost in the main condenser. Approximately 6" of the handle remained intact. The upper portion of the handle was flattened on both sides due to operation of the bypass valve. Mechanical interference of the bypass valve operation caused by either the hammer or the spring device attached to the handle can only be inferred but not conclusively demonstrated. Disassembly of the #1 bypass valve showed small dents on the disc and the seat of the valve. All the dents on the disc appeared to be concentrated in an area approximately 1 square inch. The dents on the seat corresponded to the same relative position as on the disc. The seat was machined to remove the dents and the disc was replaced due to the difficulties in removing the old disc.

An evaluation of the event by PP&L's Engineering Analysis Group has determined that the event is less limiting than the main turbine trip without bypass transient from <30% power which is discussed in Section 15.2.3 of the FSAR. The turbine trip event results in a greater reduction in steam flow and is initiated from a higher power level. The higher initial power level results in a larger void collapse in the core causing a higher power spike. Section 15.2.3.3.3.3 of the FSAR states that the turbine trip without bypass event results in a high vessel pressure scram. Therefore, the peak power remains below the flow biased simulated thermal power upscale trip setpoint and the MCPR remains well above the GETAB safety limit. Since the initial power was lower, the steam flow reduction and subsequent pressurization was less. The pressurization increase was mitigated by bypass valve operation, and therefore the event that occurred on May 28, 1984 was less severe than a turbine trip without bypass event from low power.

Figure 1


5 VALVES
4 VALVES
3 VALVES
2 VALVES
1 VALVE
Reactor Steam Dome Pressure Decrease (PSI)
0
10
20
30
40
50
Maximum Steamline Surge Flow (LBS/SEC)
0
500
1000
1500
2000
2500

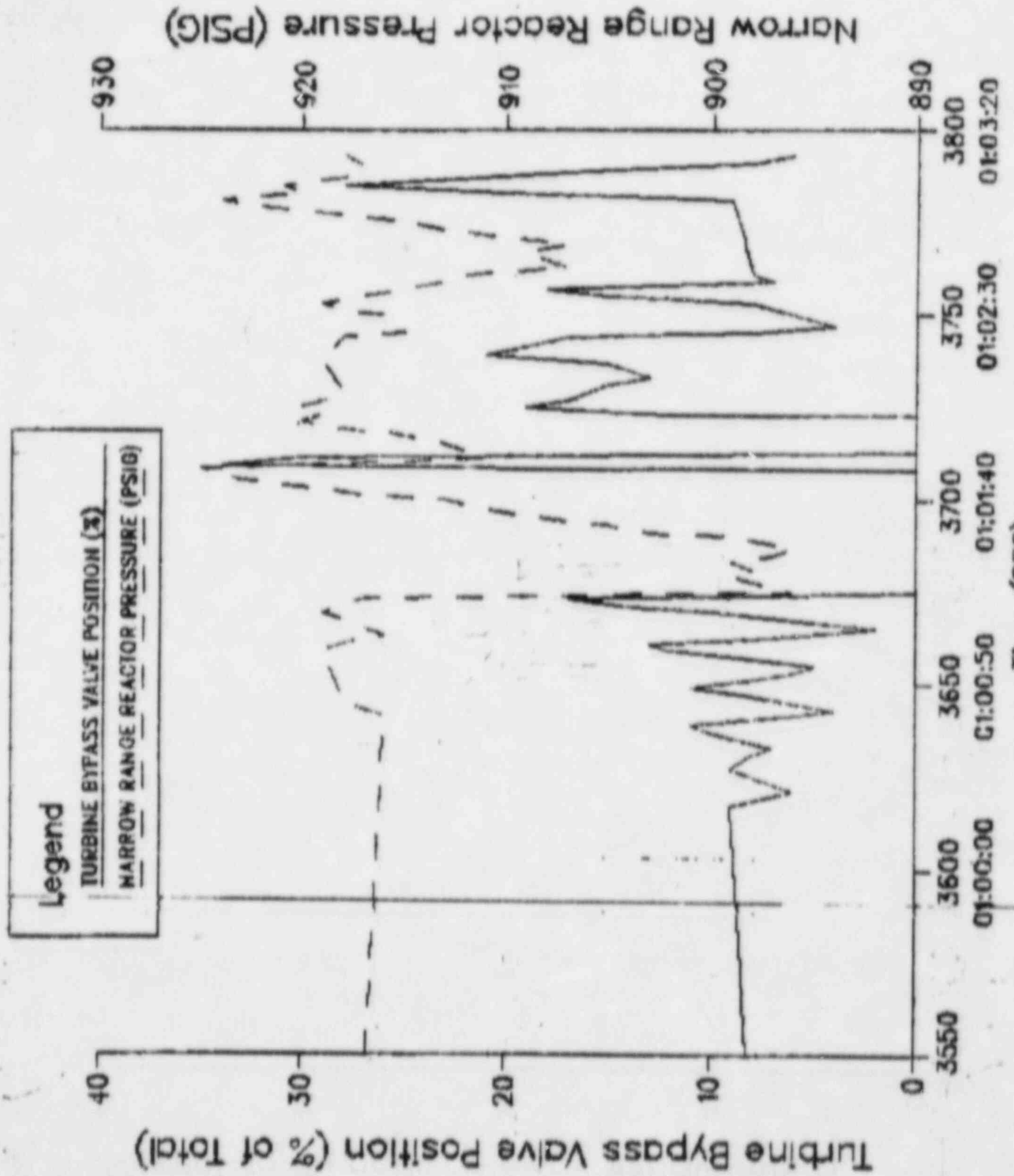

Legend
TURBINE BYPASS VALVE POSITION (%)
NARROW RANGE REACTOR PRESSURE (PSIG)
Narrow Range Reactor Pressure (PSIG)
930
920
910
900
890
Turbine Bypass Valve Position (% of Total)
40
30
20
10
0
3550
3600
01:00:00
3650
01:00:50
3700
01:01:40
3750
01:02:30
3800
01:03:20
Time (SEC)

Figure 2