VIRGINIA ELECTRIC AND POWER COMPANY
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February 28, 1986

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
Attn: Mr. Lester S. Rubenstein, Director
PWR Project Directorate No. 2
Division of PWR Licensing - A
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Serial No. 86-094
E&C/TLG/psj:2201N
Docket Nos. 50-280
50-281
License Nos. DPR-32
DPR-37

Gentlemen:

VIRGINIA ELECTRIC AND POWER COMPANY
SURRY POWER STATION UNITS NOS. 1 AND 2
ADDITIONAL INFORMATION RELATED TO
NUREG-0737, ITEM II.D.1
PERFORMANCE TESTING OF RELIEF AND SAFETY VALVES

In our letter dated October 31, 1985, Serial No. 85-615A, we provided a partial submittal in response to your additional information request dated August 13, 1985. The remaining information, not provided at that time, is included in the attachment.

Very truly yours,

W. L. Stewart

Attachment

cc: Dr. J. Nelson Grace
Regional Administrator
NRC Region II

Mr. D. J. Burke
NRC Resident Inspector
Surry Power Station

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Attachment 1
(Serial No. 86-094)

VIRGINIA ELECTRIC AND POWER COMPANY
SURRY POWER STATION UNIT NOS. 1 AND 2
ADDITIONAL INFORMATION RELATED TO
NUREG-0737, ITEM II.D.1
PERFORMANCE TESTING OF RELIEF AND SAFETY VALVES

QUESTION

1. The licensee contended that the feedline break accident was not part of the Surry licensing basis, since the plant was licensed prior to issuance of Regulatory Guide 1.70, Revision 1. Therefore, the feedline break accident was not addressed in the plant specific submittal for Surry (Reference 2).

   However, there is a possibility of an extended period of water discharge through the safety and relief valves following a feedline break, and NUREG-0737, Item II.D.1 requires that all plants consider this accident in determining the limiting valve inlet fluid condition. Therefore, provide the flow parameters (such as the maximum pressure, the maximum pressurization rate, the range of liquid temperature and the maximum surge rate into the pressurizer) which define the fluid conditions at the valve inlet.

RESPONSE

1. The feedline break accident for Surry has been analyzed using the RETRAN System Transient Analysis computer code. The analysis utilized the Surry two-loop model documented in VEPCO Topical Report, VEP-FRD-41A, "Vepco Reactor Transient Analyses Using the RETRAN Computer Code", May 1985. Two cases were analyzed:

   - Case I / Main Feedline Break w/o Power Operated Relief Valves and w/o Safety Injection

- Case II / Main Feedline Break w/ Power Operated Relief Valves and w/ Safety Injection

Case I was used to define the limiting inlet fluid conditions for the safety valves since it did not take credit for operation of the PORV's. Both cases were considered in defining the limiting conditions for the PORV's. The EPRI test results were used to define conservative critical flow models for the safety and power operated relief valves. The opening and closing characteristics observed in the tests were also factored into the definition of conservative valve characteristics for the analysis.

Following are some of the key assumptions used in both analyses that demonstrate the conservative basis of the evaluation of the valve inlet fluid conditions:

- The feedline break was artificially located at the bottom of the steam generator to maximize the rate of liquid removal from the generator. This minimizes the primary system cooldown during the period when the steam generator is blowing down.

- The decay heat is based on 120% of the 1971 ANS standard.

- No credit is taken for reactor trips on high pressurizer pressure or lo-lo level in the faulted steam generator.

- Main feedwater to all steam generators is assumed to cease completely at the time of the leak.

- The level in the intact steam generator is assumed to be 10% above nominal and the level in the faulted steam generator is assumed to be 10% below nominal. This causes the faulted steam generator to dry out earlier and the scram to occur later in the transient.

- The reactor coolant pumps are assumed to operate throughout the transient.

The aforementioned assumptions provide a substantial margin of conservatism in the analyses. More realistic input, such as the proper break location or the use of the 1975 ANS decay heat standard would result in much less severe conditions at the PORV's and safety valves.

The specific results of the feedline break analysis in response to NRC Request No. 1 are as follows:

Case I / MFLB w/o PORV's and w/o Safety Injection

- Maximum pressure = 2575 psia
- Maximum pressurization rate = 74 psi/sec
- Range of liquid relief temperatures = 624 to 626 degF
- Maximum surge rate (during liquid relief) = 374,400 lb/hr (1146 GPM)

Case II / MFLB w/PORV's and w/Safety Injection

- Maximum pressure = (2512 psia steam relief)
  (2353 psia water relief)

- Range of liquid relief temperatures = 587 to 621 degF

- Maximum surge rate (during liquid relief) = 241,200 lb/hr
  (710 GPM)

The test conditions for the safety valves and the PORV's are summarized in EPRI NP-2628-LD, "EPRI PWR Safety and Relief Valve Test Program," September 1982.

Of the safety valves which were tested, the CROSBY HP-BP-86-6M6 was chosen to be representative of the Surry safety valves. These valves are the same configuration as the Surry 6K26 valves, have the same inlet and outlet size, but have a smaller orifice area. The 6M6 valve was exposed to the following test conditions:

- Maximum pressure = 2271 to 2760 psia
- Maximum pressurization rate = 1.5 to 375. psi/sec
- Range of liquid relief temperatures = 644 to 650 degF

- Maximum steady liquid flow = 650,000 to 710,000 lb/hr
  (2115 to 2344 GPM)
  Adjusted for orifice area difference
  454,000 to 496,000 lb/hr
  (1477 to 1637 GPM)

The PORV's used in Surry are the Copes-Vulcan D-100-160 with 17-4PH cage and plug. A total of 10 tests were performed to evaluate the hydraulic performance of the valves. A number of other tests were performed to study the reliability of actuation of the valves. The hydraulic performance tests covered the following ranges:

- Pressure = 2135 to 2477 psia (steam relief)
  585 to 2430 psia (water relief)
- Range of liquid relief temperatures = 112 to 649 degF
- Range of steam flow rates = 220,000 to 265,700 lb/hr
- Range of liquid flow rates = 399,600 to 997,200 lb/hr
- Range of liquid flow rates = 547,200 to 997,200 lb/hr
  (for those tests performed (1796 to 2394 GPM)
  near the pressure setpoint
  for the Surry PORV's)

The pressure and pressurization rates experienced by the safety valves in the conservative feedline break analysis are bounded by the EPRI test data. The range of liquid relief temperatures are also bounded by the data. The PORV pressure and temperature ranges during liquid relief are bounded by the EPRI tests.

The flow rates through the safety valves in the EPRI test program, when adjusted to account for the different orifice areas, are well in excess of the results of the Case I analysis. In Case II the maximum surge rate occurs as the PORV cycles open and closed during the liquid relief period. The maximum predicted liquid relief flow rate for Case II is 241,200 lb/hr or approximately 710 GPM. The Case II liquid relief flow rates are lower than the Case I flow rates because of the cooling effect of the safety injection fluid.

EPRI tests that were conducted with liquid relief at pressures near the setpoint of the Surry PORV's relieved flow rates from 547,200 lb/hr to 997,200 lb/hr or 1796 to 2394 GPM. These flow rates are well in excess of the flow rates predicted in the analysis: Case I, 374,000 lb/hr (1146 GPM), or Case II, 241,200 lb/hr (710 GPM). The test results for the Copes-Vulcan PORV's demonstrate that a single PORV can relieve liquid flow rates well in excess of the liquid release rates in either of the cases analyzed.

In summary, Virginia Electric and Power Company has performed an extremely conservative analysis of the feedline break accident for Surry and demonstrated that both the safety and relief valves experience transient pressure and fluid conditions much less severe than those to which the valves in the EPRI test program were exposed.

QUESTION

3. On Page 6 of Reference 3, the licensee stated that the safety valve could not be challenged by the high pressure injection transient, because the charging pump did not have sufficient pressure head to reach the safety valve inlets. However, Reference 3 did not address the flow condition at the PORV inlets. Confirm whether fluid discharge is expected through the PORV's in the high pressure injection event so that a proper evaluation of this case can be made.

RESPONSE

3. The Surry safety injection system can deliver significant flow (on the order of 300 gpm) at the PORV setpoint of 2335 psig. Therefore it can reasonably be concluded that some steam discharge from the PORV's can be expected in the event of an inadvertent safety injection.

In order to ascertain whether liquid discharge might occur upon an inadvertent high pressure injection, two cases were examined. The first case was an inadvertent injection at hot shutdown conditions. The second case assumed an injection without direct reactor trip at hot full power, beginning of cycle with a high initial boron concentration in the RCS. The first case was analyzed by hand, while the second involved a RETRAN analysis using the same model as for the feedline break analysis discussed above.

Results for Hot Zero Power

With the plant at hot zero power operating conditions, a pressurizer steam space of slightly under 1000 cubic feet is available. Following the injection, pressurizer pressure will increase from the nominal operating value due to compression of the steam space. Increased pressure reduces the injection flow rate. A conservative estimate of the minimum pressurizer fill time was made assuming the pressurizer spray system functions to hold the pressure at or just above the nominal setpoint. Under these conditions the calculations show that the operator has in excess of 18 minutes to identify and terminate the event. Among the clear indications available to the operator, which would indicate that safety injection is functioning, and may be terminated are as follows:

- The safety injection annunciator and flow indication
- RCS pressure increasing
- Pressurizer level increasing; high level and level deviation alarms
- RCS temperature stable or slightly decreasing
- Steam pressure stable
- Adequate RCS subcooling margin.

It is therefore concluded that there is adequate time available for the operator to diagnose and terminate the event prior to the onset of liquid discharge from the PORV's.

Results for Hot Full Power

In the event of a spurious safety injection at power, the most likely immediate response will be a reactor trip as a result of receipt of a safety injection signal. Following the trip the transient will proceed as discussed previously for hot zero power. For purposes of this analysis, we have also postulated actuation of the safety injection system without a direct reactor trip.

For this latter case, an analysis was performed using the RETRAN computer code. Key assumptions of the analysis were:

1) Maximum safety injection flow, associated with the operation of two high head injection pumps. The variation of injection flow rate with pressure is included in the calculation.

2) No boron in the safety injection piping between the Refueling Water Storage Tank and the reactor coolant system. This delays the negative reactivity input associated with safety injection and reduces the predicted pressurizer fill time.

3) A bounding, high value of initial boron concentration associated with beginning of core life. Maximizing initial boron minimizes the negative reactivity effects of the soluble boron in the injection fluid, reduces the cooldown effects and minimizes the pressurizer fill time.

4) Automatic operation of the control rods. Outward motion of the control rods will act to counter the negative reactivity insertion of the safety injection until the control bank is fully withdrawn from the core. This assumption maximizes the pressurizer water volume and minimizes the pressurizer fill time.

The analysis results showed an initial increase in RCS water volume and RCS pressure in response to the injection. Power is held stable by outward motion of the control rods. At about 6 minutes into the transient, the rods are fully withdrawn, and power, temperature and pressure begin to decrease in response to the continuous increase in boron concentration. This trend continues for an additional 2.5 minutes until the reactor trips on low pressurizer pressure.

The reduced pressure results from the fact that the reactor coolant shrink due to cooldown more than offsets the mass addition associated with the injection. Following the reactor and turbine trip, pressurizer level begins to increase again. About 9 additional minutes are available to the operator after the trip to take action to terminate the event prior to filling the pressurizer.

Table 3 shows the PORV inlet conditions realized during this transient. Liquid relief conditions are included, although based on the discussions above, significant periods of water relief for this event are considered unlikely. Also shown are the EPRI test parameters for the Copes-Vulcan D-100-160 valve with 17-4PH cage and plug, which is most representative of the Surry PORV's. As can be seen, the predicted conditions for the extended high pressure injection event fall well within the test conditions.

TABLE 3
SURRY VALVE INLET CONDITIONS SUMMARY
HIGH PRESSURE INJECTION EVENT

PARAMETER	RELIEVING STEAM	RELIEVING LIQUID
Peak pressure, psia	2353	2358
Pressurization rate at PORV opening time, psi/sec	5.0	8.8
Range of fluid temperatures, °F	659-682	558-581
Maximum surge line flow when relieving liquid, lb/hr	-	446,400 (1190 GPM)

EPRI TEST DATA*

Pressure Range, psia	2135-2477 (steam) 585-2430 (water)
Liquid relief temperatures, F	112-649
Steam flow rates, lb/hr	220,000-265,700
Liquid flow rates, lb/hr (all liquid relief tests)	399,600-997,200
Liquid flow rates, lb/hr (for those tests performed near the pressure setpoint for the Surry PORV's)	547,200-997,200 (1796-2394 GPM)

* The manner in which the EPRI PORV testing was done precludes the definition of a meaningful pressurization rate for PORV's.

Question

4. The magnitude of the maximum bending moment induced at the inlet flanges of the safety valves and PORV's during valve discharge were not provided in the Surry transmittal. Give the predicted values of the maximum moments for these valves so that the valve operability can be evaluated.

Response

4. In our letter dated December 29, 1983, Serial No. 721, we stated that North Anna Unit 2 piping evaluations were to be completed first and the plant-specific piping calculations for the remaining three units would use information and guidance from the North Anna Unit 2 evaluation.

   In keeping with this approach, the Surry thermal-hydraulic analysis was performed using only the safety valve discharge. Evaluation of the North Anna Unit 2 and Unit 1 analytical results concluded that the safety valve discharge created the limiting loading force on the piping and support system. The power operated relief valve (PORV) discharge was therefore not analyzed. The calculated moments at the safety valve and power operated relief valve inlets due to the safety valve discharge loading conditions were provided in our submittal dated October 31, 1985, Serial No. 85-615A. Further explanation on the approach is available in the response for question 6a.

Question

6. The thermal-hydraulic analysis of the safety/PORV inlet and discharge piping was discussed on pages 15 and 16 of Reference 3. In the discussions only information concerning safety valve discharge conditions was provided. The PORV discharge conditions were not addressed.

   a. Provide a discussion of the thermal-hydraulic analysis of the PORV discharge conditions and give the input parameters used in the analysis.

Response

6a. No discussion of the thermal-hydraulic analysis of the PORV discharge conditions was provided because the Surry thermal-hydraulic analysis for the pressurizer safety and relief valves (PSARV) piping system was performed using only the safety valve discharge. This approach was established based upon the North Anna Unit 2 and 1 analytical results which concluded the maximum loading force on the piping and support system was created by safety valve discharge, and was the limiting condition.

In keeping with the approach as indicated in our December 29, 1983 letter, the North Anna Unit 2 analysis was completed and evaluated first. Then North Anna Unit 1 was addressed, followed by the Surry analysis for Units 1 and 2.

Surry Units 1 and 2 were reviewed and evaluated on their own merits, as well as, on a comparative basis with the analytical findings and conclusions for North Anna Unit 2 and Unit 1.

It was determined that the safety valve discharge analysis was limiting because of the following:

A) The safety valves have designed loop seals with the inlet and discharge piping being the same diameter, while the relief valve inlet piping is smaller in diameter than the discharge (4" vs 6"). This increase in the discharge piping diameter allows for expansion thus reducing the loading forces. The relief valves do not have designed loop seals. Due to the piping configurations some condensate may collect. However with the smaller inlet piping diameter the volume of liquid will be minimal when compared with that in the safety valve loop seals. This results in smaller loading forces on the piping system.

B) The relief valve opening times are slower than those of the safety valves. This results in a slower release by the relief valves than by the safety valves thus reducing the instantaneous loadings on the piping and support system. This is also supported by EPRI in their report (EPRI NP-2479-LD) entitled, "Application of RELAP5/MOD 1 for Calculation of Safety and Relief Valve Discharge Piping Hydrodynamic Loads", which states that, typically, air operated relief valves are low-capacity valves which open on the order of hundreds of milliseconds whereas spring-loaded safety valves are high capacity, fast-acting devices opening on the order of tens of milliseconds.

C) The flow capacity of the relief valves is some 25% lower than the flow capacity of the safety valves (210,000 lb/hr vs. 290,000 lb/hr). This minimizes the volume of condensate that could possibly pass through the PORV's at the instant it opens, thus reducing the loading to the discharge piping.

D) The North Anna pressurizer safety and relief piping analysis and evaluations addressed both the safety and relief valves. After review of the analysis and evaluations, engineering judgement concluded, from the thermal-hydraulic analysis, the worst applicable loading for the PSARV discharge piping and support system resulted from the three safety valves opening simultaneously. Forcing functions were then generated assuming this condition. The results of the analysis showed the operability and structural integrity of the system was ensured contingent upon the elevated safety valve loop seal and verification of the support adequacy.

The North Anna Unit 2 analysis used a minimum analytically calculated steam flow for each PORV of greater than 255,000 lb/hr; which was a flow of 123% of the rated 210,000 lb/hr, and greater than those used in the EPRI tests (228,600 and 230,400 lb/hr). The opening time used for the PORV's was 1.00 second, while the EPRI tests averaged 2.77 seconds with a minimum opening time of 1.64 seconds. Opening was also assumed to be 100% linear. As can be seen, all values used for the North Anna Unit 2 analysis were very conservative to ensure adequate margin to account for uncertainties and tolerances.

The results of the North Anna Unit 2 analysis demonstrated that the maximum loading force in any particular segment of the piping system was created by the simultaneous opening of the safety valves and not the opening of the PORV's.

The Surry Units 1 & 2 PORV's were reviewed and comparatively evaluated against North Anna Unit 2. Comparisons of the flow rates, valve opening times and piping configurations were made to confirm that safety valve discharge is the limiting condition for Surry.

E) The North Anna Unit 1 analysis further supported the North Anna Unit 2 conclusions in that it concluded all safety and relief valve discharge events were bounded by the three safety valve loop seal clearing event.

Question

7. The discussion of the piping stress analysis on pages 17 through 20 of Reference 3 did not include the stress evaluations for the safety/PORV piping and supports and other pertinent data. The following information is required in order to complete the evaluation of the piping stress analysis.

   a. Give numerical values of the cutoff frequency and total analysis time for safety valve and PORV discharge used in the NUPIPE computer analyses.

Response

7a. As noted in the response to 6a, the Surry thermal-hydraulic analysis for the pressurizer safety and relief valve piping system was performed using only the safety valve discharge. The cut-off frequency and mode and the total integration time for the NUPIPE computer analysis provided in our submittal dated October 31, 1985, Serial No. 85-615A are the applicable values and are as follows:

	Unit 1	Unit 2
Cut-off frequency (HZ)	502	543
Cut-off mode	196	181
Total integration time (seconds)	0.9	0.9

Both the inlet and discharge piping of the safety as well as relief valve piping from the pressurizer nozzle to the pressurizer relief tank nozzle were included in the analysis for the safety valve discharge loading.