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ARTHUR E. LUNDVALL, JR.  
VICE PRESIDENT  
SUPPLY

October 23, 1985

Director of Nuclear Reactor Regulation  
Attention: Mr. E. J. Butcher, Jr., Chief  
Operating Reactors Branch #3  
Division of Licensing  
U.S. Nuclear Regulatory Commission  
Washington, D.C. 20555

Subject: Calvert Cliffs Nuclear Power Plant  
Units Nos. 1 & 2; Dockets Nos. 50-317 and 50-318  
TMI Action Item II.D.1

References: (A) Letter from E. J. Butcher, Jr. to A. E. Lundvall, Jr., dated  
July 9, 1985, same subject.  
(B) Letter from R. C. L. Olson to E. J. Butcher, Jr., dated  
September 4, 1985, same subject.

Gentlemen:

You requested additional information on performance testing of safety and relief valves in Reference (A). By Reference (B) we stated that we were in the process of answering the questions enclosed in Reference (A) and expected to submit a full response by October 31, 1985. The enclosure to this letter contains our response.

If you have any further questions on this subject, please do not hesitate to contact us.

Very truly yours,

*A. E. Lundvall, Jr.*  
for  
A. E. Lundvall, Jr.  
Vice President - Supply

AEL/WPM/dmk

Enclosure

cc: D. A. Brune, Esq.  
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PDR ADOCK 05000317  
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bcc: Messrs. J. A. Tiernan  
R. F. Ash  
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R. C. L. Olson/ B. S. Montgomery  
M. J. Meirnicki/ S. R. Cowne  
W. P. McCaughey, Jr.

The responses presented herein are to NRC questions (per Reference 1) related to TMI Action NUREG-0737 (II.D.1) as it applies to Calvert Cliffs Nuclear Power Plant (CCNPP) Units 1 and 2.

Response to Question 1:

The plant specific portion of the C.E. prepared report on valve inlet fluid conditions (specifically, table 5-4 of reference 10) correctly states that under loss of A-C conditions, the PORVs at Calvert Cliffs do not operate. This is true since the PORVs do not obtain power from the emergency buses and under loss of A-C no power is available to operate them.

Response to Question 2:

The lifting of the PORVs during low temperature conditions was not considered a design basis event for the piping analysis. This transient occurs only after the plant is in a safe shutdown configuration and, as such, does not constitute a safety concern.

Response to Question 3:

The EPRI tests demonstrated that Dresser PORVs operate satisfactorily, i.e., open and close on demand, under a wide range of inlet fluid conditions. These conditions included steam at high pressure (2400-2500 psia), water at pressure of 2400-2500 psia, and water at low pressure (680-700 psia) (See Reference 8).

Additionally, a series of steam tests with various PORV set pressures were conducted by the manufacturer (See References 7 and 8). According to Dresser, the tested PORV opened and closed without

failure and with no apparent leakage at opening pressures of 40 psig to 1964 psig. Consequently, Dresser conservatively concluded that the minimum operating pressure for Calvert Cliffs Units 1 & 2 PORVs is 75 psig.

The PORVs tested in both test programs were representative of those in service at Calvert Cliffs Units 1 and 2.

Since the inlet fluid conditions in these tests enveloped or approximated the plant-specific inlet fluid conditions presented in Reference 9, it is concluded that the PORVs in Calvert Cliffs Unit 1 and Unit 2 are expected to operate satisfactorily under all anticipated inlet fluid conditions including low pressure steam.

#### Response to Question 4:

The bending moments recorded during the EPRI Safety Valve Test Program represent an as-tested valve loading where both valve operability and structural integrity were demonstrated. With the measured bending moment acting on the safety valve, the valve opened, discharged, and closed in a satisfactory manner. These tests demonstrated that all valve body or component distortion due to this bending moment was small and did not cause binding or interference.

The bending moments reported in Table A were calculated by multiplying the vertical force at the second discharge elbow times the horizontal distance to the valve discharge flange. In Figure 1, the moment equation is expressed as  $M = [WE (32 + 33)Y]L$ . The expression  $WE (32 + 33)Y$  reflects the vertical summation of two load cell readings at the second elbow. The vertical support at this

elbow is shown in Figure 2 and consists of an A-frame with a load cell in each leg of the A-frame. This vertical load is recorded continuously throughout each test.

These bending moments are based on the value of  $WE (32 + 33)Y$  just before the valve opens and just after the valve closes. The loads recorded at the second elbow support at these times consist of dead weight, initial bolt up, and thermal expansion loads. All of these loads are transmitted back to the test valve and develop bending moments at the valve discharge flange. These bending moments act about the horizontal, out of plane Z axis in Figure 1. This direction of bending is as severe or more severe than any other plane of bending so that these allowable moment values are recommended to be considered the as-tested values for all moment directions.

An effort has been made to determine the moments imposed on the test valve while the valve is open and flowing. This effort has included both periods of valve instability and valve steady-state open flow. When the valve is open, hydrodynamic forces are acting on the valve and discharge piping supports. When these forces are resisted at the second elbow vertical support, only a portion of the load is transmitted back to the valve. Sufficient test data are not available to separate out these portions of load, so the second elbow vertical forces are not applicable for the dynamic moment calculation.

The horizontal piping downstream of the second elbow did not influence the load cell readings or bending moment calculations. The initial cold load cell reading was zeroed out in the moment calculation to eliminate any piping dead weight effects. In addition, the horizontal piping was supported by dead weight spring hangers to minimize dead weight effects. The load cell reading prior to valve opening represents tank thermal growth and the load

cell reading after valve closing includes the effects of discharge piping thermal growth.

The measured bending moments which act on the safety valve can be directly compared to analytically calculated moments since both challenge valve operability and structural integrity. Note that the tested load acceptance criterion is not an upper limit load based on failure criteria. This acceptance limit represents the highest measured load during the test program. During the entire test program, the measured bending moments never caused a valve to malfunction. This test record indicates that valve operability is not sensitive to bending moments and that actual bending moments limits are higher than the measured values.

In conclusion, the results of the EPRI Safety Valve Test Program demonstrate that the bending moments on the safety valve flanges due to thermal expansion of the pressurizer and piping and by the discharge loads will not impair valve operability provided that the anticipated loads are less than those measured in the tests. In this case, the maximum calculated bending moments acting on the safety valve discharge flanges are 12,727 in-lbs for Unit 1 and 20,561 in-lbs. for Unit 2. The maximum as-tested bending moment for this type of safety valve was 241,738 in-lbs (See Table A, Test 1011), i.e. more than ten times the plant-specific bending moments, and the valve operability was not impaired.

Table A

## AS-TESTED BENDING MOMENTS

<u>Test</u>	<u>Valve</u>	<u>Opening Moment (in-lbs)</u>	<u>Closing Moment (in-lbs)</u>
302	Dresser (2- $\frac{1}{2}$ x6) 31739A	80,113.	94,250.
304		75,400.	94,250.
306		75,400.	84,825.
308		80,113.	98,963.
310		84,825	103,675.
312		75,400.	98,963.
314		75,400.	94,250.
316		84,825	89,538.
318		84,825.	94,250.
320		84,825.	103,675.
322		80,113.	103,675.
324		75,400.	89,538.
326		98,963.	84,825.
328		65,975.	89,538.
1003		145,625	52,425.
1005		163,100	99,025.
1008		49,513	64,075.
1011		241,738.	75,725.
1012		64,075.	69,900.
1016		186,400.	52,425.
1017		186,400.	145,625.
1018		0.	49,513.
1021		157,275.	52,425.
1025		157,275.	17,475.
1027		131,063.	N/A
1030		87,375.	58,250.
1104		230,913.	56,550.
1107		226,200.	9,425.
1110		169,650.	4,713.
1112		158,340.	18,850
1114		84,825.	N/A

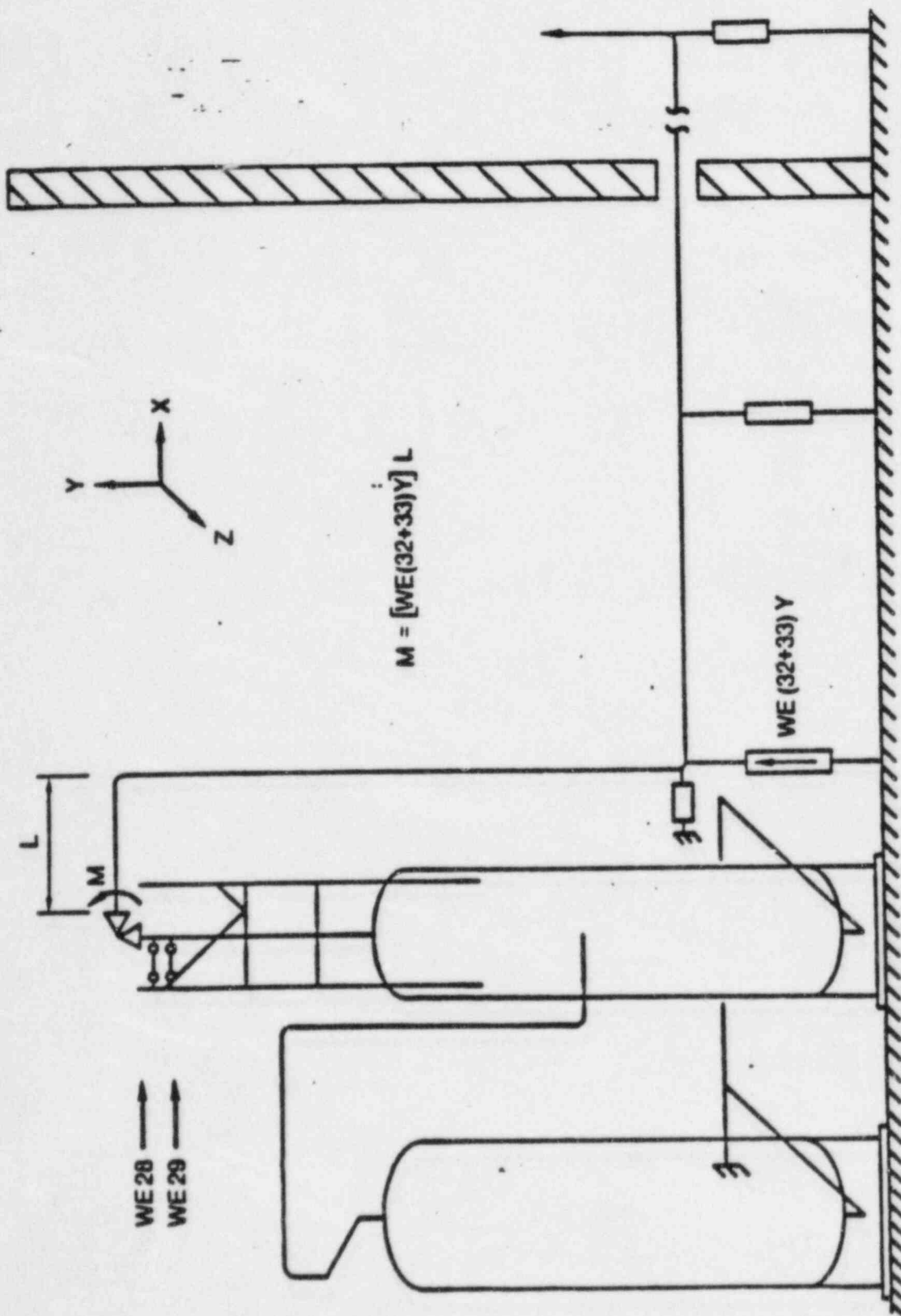


Figure 1. Test Facility Schematic

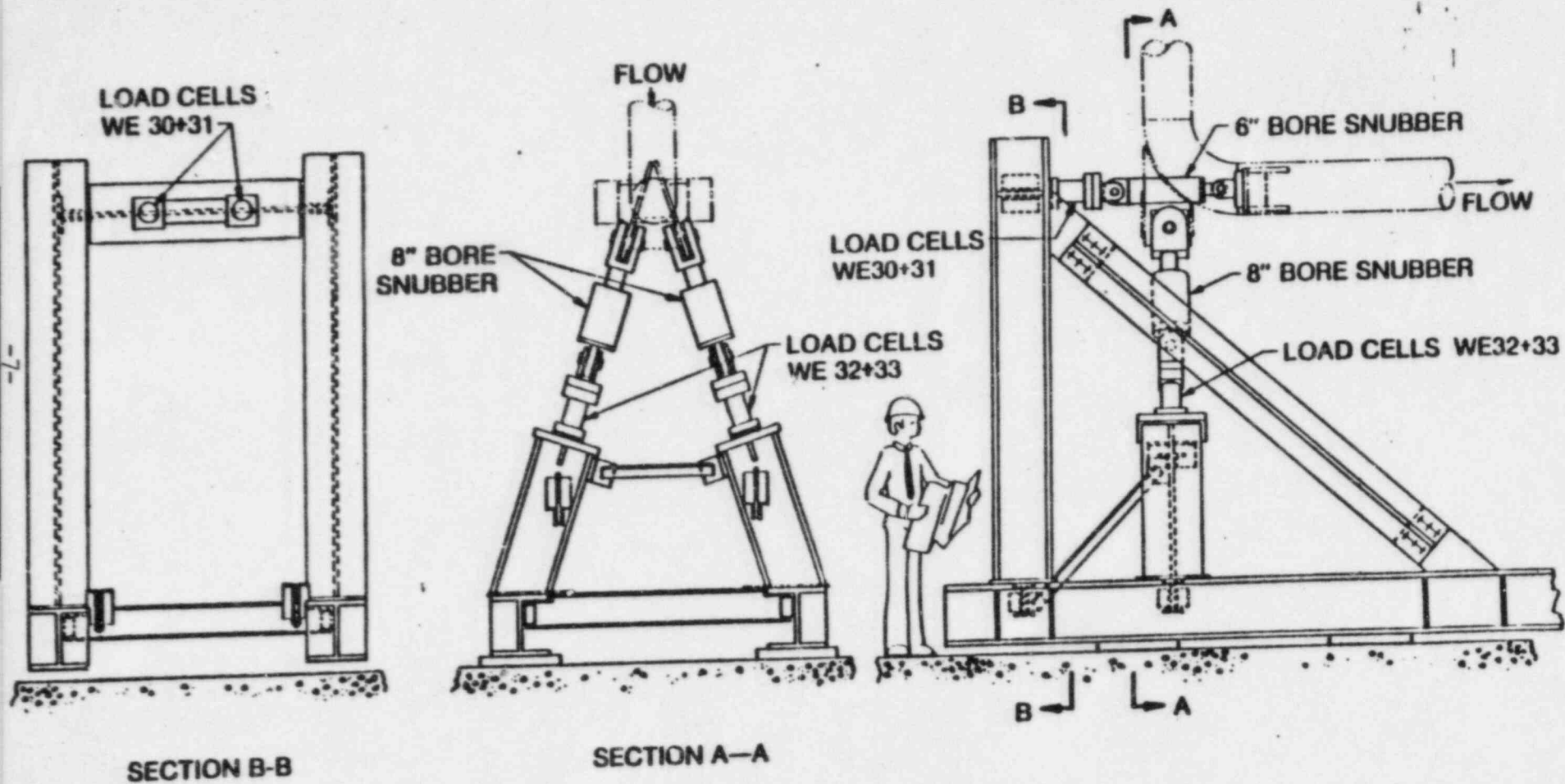


Figure 2. Second Discharge Elbow Support

Response to Question 5:

The bases for stating that the Calvert Cliffs motor operated block valves are capable of functioning as designed are as follows:

1. The valve design represents technology that has been in use in the power production industry for many years. That is, the valve is a simple wedge gate design which is used in fossil as well as nuclear plants in many different applications.
2. The vendor (Velan valves) has assured us that the valve will function at the design maximum differential pressure of 2485 psi.
3. A new environmentally qualified motor operator has been installed on the valve which can provide more than sufficient torque to close the valve and maintain a leak-tight seal at the maximum design differential pressure.
4. Two incidents occurred at Davis Besse Unit No. 1 where the PORV was opened and failed to reseal and where closure of the block valves provided proper isolation -

(a) September 24, 1977

Ref.: Licensee Event Report  
NP-32-77-16 & Supplement  
Docket 50-346  
October 1977 & November 1977

(b) June 9, 1985

Ref.: I.E. Information Notice  
No. 85-50

These events coupled with the fact that the Davis-Besse block valves are Velan valves of similar design and materials of construction as the valves at Calvert Cliffs support the operability of our valve under actual conditions.

5. The EPRI - Marshall Electric Motor operated valve (Block Valve) Test results (See Reference 13) show that a similar design Velan valve with similar materials of construction was cycled 21 times under expected pressure and flow conditions with zero leakage. This valve was cycled against a full flow of 244,790 lbm/hr. which envelopes the Calvert Cliffs design flow for the PORV's which is 153,000 lbm/hr.

#### Response to Question 6:

##### Introduction

It follows from recommendations of the referenced EPRI documents that in a plant-specific evaluation of inlet piping pressure effects on safety valve stability, a comparison should be made between the expected plant-specific inlet piping pressure drop and the inlet piping pressure drop measured in a reference test. It makes a difference, however, what pressure drop to apply as a criterion in the comparison, static or stagnation, for the use of an improper criterion could adversely impact conclusions on stability of plant-specific safety valves.

Based on the methodology and recommendations of References 2, 3, and 4, Combustion Engineering (C-E) has developed a consistent and justified approach to the stability evaluations. This approach, which was used in the Calvert Cliffs Units 1 and 2 safety valve evaluation (See Reference 5), was first employed in the generic evaluation performed for the C-E Owner's Group (See Reference 6).

The approach is based on the use of the stagnation pressure drop as the criterion in the comparison between in-plant and as-tested safety valve/inlet piping configurations. It was demonstrated that the static pressure drop can be used as the criterion only selectively, i.e., when certain conditions are met.

#### Discussion<sup>1)</sup>

As the safety valve opens, the stagnation pressure upstream of the valve drops due to sudden expansion causing an expansion wave moving towards the pressurizer (or the tank, in tests) with the speed of sound. Simultaneously, frictional losses due to the flow in the inlet piping reduce the stagnation pressure even more. The combined acoustic and frictional pressure drop could be large enough to reduce the stagnation pressure down to a level at which forces acting on the valve disc could no longer sustain the valve in an open position. The extent of the stagnation pressure drop, for the same valve, depends mostly on the valve inlet piping geometry: short and straight piping of a large and uniform cross section would minimize the stagnation pressure drop. It should also be noted that due to fast valve opening the pressure drop due to the expansion wave (acoustic wave amplitude) is considerably greater than the frictional pressure drop. Accordingly, the contribution of the acoustic component into the stagnation pressure drop is more significant.

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<sup>1)</sup> The discussion presented herein is partly based on the analysis of Reference 4.

Due to the flow, static pressures in the inlet piping and at the valve inlet decrease to below the reduced stagnation pressure. This reduction in the static pressure is caused by the velocity head of the moving medium. However, unlike the acoustic and frictional pressure drops, the pressure drop due to velocity head has nothing to do with the inlet piping per se. It is just an indicator of the kinetic energy of the flow at a particular point in the inlet piping, be it a pressure transducer location or the valve inlet. Aside from the flow rate and the density of the medium, the pressure drop due to velocity head is only a function of the local inside diameter (I.D.).

Thus, measurements taken simultaneously at PT-12 and PT-105 pressure transducer locations<sup>1)</sup> during some tests indicate different static pressure drops which are mostly attributed to the differences in velocity heads due to different I.D. at these locations. But this indication does not affect the velocity head at the valve disc, which in conjunction with the static pressure provides the necessary force and momentum to keep the valve in an open position.

In other words, the pressure drop due to velocity head, unlike the acoustic and frictional components, does not adversely impact valve stability since it is not a loss, it is just another form of energy which is available at the valve disc.

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1) PT-105 transducer was located in the mating inlet piping end flange at the valve inlet. PT-12 transducer was located in the inlet piping some distance upstream from the valve flange. PT-105 measurements were not available for all valve tests.

Based on the foregoing, one can conclude that it is the stagnation pressure drop (not the static) which is a relative indicator of reduction in forces available at the valve disc and, therefore, should be used as the criterion in the comparison between in-plant and in-test safety valve/inlet piping combinations. The static pressure drop should not generally be used as the criterion since results of such a comparison may not be representative or could even be misleading. This is demonstrated below using the Calvert Cliffs Units 1 and 2 plant-specific evaluation for illustration.

Stagnation vs. Static: Calvert Cliffs Units 1 and 2

1. The procedure followed in the Reference 5 evaluation on safety valve stability is outlined below:
  - (a) Test #1008 was chosen as a reference test for Calvert Cliffs Units 1 and 2 plant-specific evaluation. The test static pressure drop was obtained through the PT-12 transducer.
  - (b) Test-specific data were input into the PIPES computer codes to calibrate the code against the test and establish a difference (if any) between the simulated and test pressure transients.
  - (c) The pressure drop due to velocity head at the PT-12 location was calculated and then subtracted from the test static pressure drop in order to get the test stagnation pressure drop.
  - (d) Plant-specific static pressure drops for Calvert Cliffs Units 1 and 2 inlet piping configurations were calculated using the PIPES code and input data from test simulation.

Then the plant-specific pressure drops due to velocity head were calculated and subtracted from the computer-calculated static pressure drops to get Units 1 and 2 plant-specific stagnation pressure drops. Each plant-specific pressure drop, i.e., static, stagnation, or that due to velocity head, was calculated for the point in the inlet piping corresponding to the PT-12 location in the test configuration.

- (e) The plant-specific and test stagnation pressure drops were compared. Then a conclusion on Calvert Cliffs Units 1 and 2 safety valve stability was drawn.

The PIPES computer code used in the evaluation to simulate the reference test and to calculate the plant-specific static pressure drops employs the method of characteristics to solve the equations of conservation of mass, momentum, and energy. Computations of fluid conditions are made for a large number of discrete points within the piping systems, including the PT-12 location in test simulations or a corresponding point in plant-specific analyses. Static pressure drops, calculated based on the Code output data, include acoustic wave amplitudes and pressure drops due to friction losses and velocity head.

- 2. The results of the evaluation are summarized in the following table.

Safety Valve/Inlet Piping Combination	$\Delta P_{\text{static}}$ , psi	$\Delta P_{\text{stagn}}$ , psi
Test #1008	258	235
Test #1008, simulated	277	255
Calvert Cliffs Unit 1, simulated	279	152
Calvert Cliffs Unit 2, simulated	329	205

3. These results demonstrate that:

- (a) The static pressure drop resulted from the computer simulation of test #1008 (277 psi) overpredicted the measured test static pressure drop (258 psi). It is concluded, therefore, that the Code overpredicts the plant-specific pressure drops. In other words, the real static pressure drops for Calvert Cliffs Units 1 and 2 are expected to be lower than 279 psi and 329 psi, respectively.
- (b) The plant-specific pressure drops due to velocity head (127 psi for Unit 1 and 124 psi for Unit 2) differ drastically from that for test #1008 (23 psi). This difference is mostly attributed to the difference in the inside diameters at the PT-12 location (3.152 in) and the corresponding Unit 1 and Unit 2 plant-specific location (2.125 in). It should be noted that a pressure drop due to velocity head is inversely proportional to the fourth power of inside diameter.
- (c) Both Unit 1 and Unit 2 inlet piping static pressure drops are greater than the test value of 258 psi. Direct comparison of these pressure drops for both plants with the tested value is inappropriate because of the difference in pipe diameters and hence the velocity head components.
- (d) Both Unit 1 and 2 stagnation pressure drops are lower than the test value of 235 psi. Therefore, using the stagnation pressure drop as the criterion results in the conclusion that stable operation is expected for both Units 1 and 2 safety valves.

4. As an exception, the static pressure drop could be used as the criterion but only under the following two conditions:
  - (a) The plant-specific and the tested safety valves are identical, and
  - (b) Inlet piping inside diameters at the locations selected for comparison such as the pressure transducer location in the test and a corresponding plant-specific location are the same.

In such cases, there is no need to subtract the velocity head components out of the static pressure drops since the velocity head is expected to be the same for both plant-specific and test combinations. Consequently, the measured value in the test and the calculated plant-specific static pressure drops can be compared directly.

### Conclusion

Based on the foregoing it is concluded that stable operation is expected for both Calvert Cliffs Unit 1 and Unit 2 safety valves since:

1. A valve similar to both plant safety valves was tested and demonstrated stable operation, and
2. The conservatively calculated plant-specific stagnation pressure drops of 152 psi, for Unit 1 and 205 psi, for Unit 2, are less than the referenced test stagnation pressure drop of 235 psi.

#### Response to Question 7

NUREG-0737, Item II.D.1 requires the Licensee to determine the expected valve operating conditions through the use of analyses of accidents and anticipated operational occurrences. Calvert Cliff's PORVs are non-safety grade equipment designed to limit overpressure transients to below the ASME code safety valve setpoint. The PORVs are not intended to prevent a high pressure reactor trip, but rather, are for use in conjunction with the trip to mitigate pressure transients.

The control circuitry was designed, procured, and installed to meet the expected valve operating conditions. The solenoid valves at Calvert Cliffs are identical to those used in the EPRI Safety and Relief Valve Program. The specific cables, although not part of our Equipment Qualification Program, are identical to those qualified for in-containment use.

#### Response to Questions 8 and 9

As indicated in the response to Question 3, a series of steam tests with various PORV set pressures were conducted by Dresser. These tests, in conjunction with the tests performed during the EPRI test program, demonstrated that Dresser PORVs satisfactorily operate under a wide range of inlet fluid conditions. These conditions include steam at 90 to 2500 psia and water at ~ 700 psia and 2400-2500 psia.

Both EPRI and Dresser tests were performed with Dresser PORVs of the type 2 design which is a modified version of the type 1 design. This upgrade resulted in improved valve seat tightness.

According to Dresser, the Electromatic relief valves, type 31533VX-30, at the Calvert Cliffs Nuclear Plant were modified to the type 2 design. (See Reference 7). Dresser also confirmed that the minimum operating pressure for Calvert Cliffs Units 1 and 2 PORVs is

75 psig. Accordingly, the recommendations made by Dresser in the referenced letter to Metropolitan Edison Co. are considered (by Dresser) to be obsolete. (See Reference 8).

Based on the foregoing, it is concluded that both Calvert Cliffs Units 1 and 2 PORVs are expected to operate satisfactorily under all anticipated plant-specific inlet fluid conditions including those associated with the low temperature overpressure protection mode of operation.

#### Response to Question 10

- a) The thermal hydraulic analysis of the relief valve piping system was performed using Relap 5/MOD 1. The model represented all of the piping from the pressurizer through the relief valves down to the quench tank. The model also included any branch lines of sufficient size to generate any important forces.

The models differed slightly for each unit, but on the average there were 340 volumes with 350 junctions. The volumes varied in size but an attempt was made to keep length/diameter 2.0. The valves were the same for each unit with the flow areas as follows:

Safeties	$1.4 \times 10^{-2} \text{ ft}^2$
PORV's	$7.0 \times 10^{-3} \text{ ft}^2$

The valve flow areas were the only choked flow junctions. The safeties were ramped from closed to full open in 0.012 seconds and the PORV's in 0.13 seconds. These represent the fastest opening times reported in the EPRI test data (See Reference 11 & 12). The fluid condition at valve opening is steam which is reported in the C.E. valve inlet fluid condition report (See Reference 10). The peak pressure and pressurization rates used are:

	<u>Safety Valves</u>	<u>PORV's</u>
Peak Press.	2538 psia	2434 psia
Press. Ramp	64.4 psi/sec.	46.0 psi/sec.

The time step on an average ranges from  $2.5 \times 10^{-4}$  seconds to  $5 \times 10^{-4}$  seconds.

- b) The safety valve models were developed by modelling the EPRI C.E. Test 1000 series (See Reference 11) which represented the inlet conditions at CCNPP. Thus, the safeties responded to inlet piping conditions as opposed to just having a flow rate assigned to it. As the flow rate was based on actual test ASME derating criteria has no significance in the flow calculation. The flow rates developed in the transient calculations are as follows:

Unit 1 - 299,200 lbm/hr  
Unit 2 - 296,700 lbm/hr

The PORV's were assigned flow areas that were half of the safety valve areas.

PORV's (Pressurizer Pressure = 2434 psia)

Unit 1 - 156,200 lbm/hr  
Unit 2 - 155,900 lbm/hr

- c) See response to question 11.
- d) The sketch of the thermal hydraulic model is shown in figures 3 & 4.



# CALVERT CLIFFS NUCLEAR POWER PLANT

## THERMAL HYDRAULIC MODEL - UNIT 1

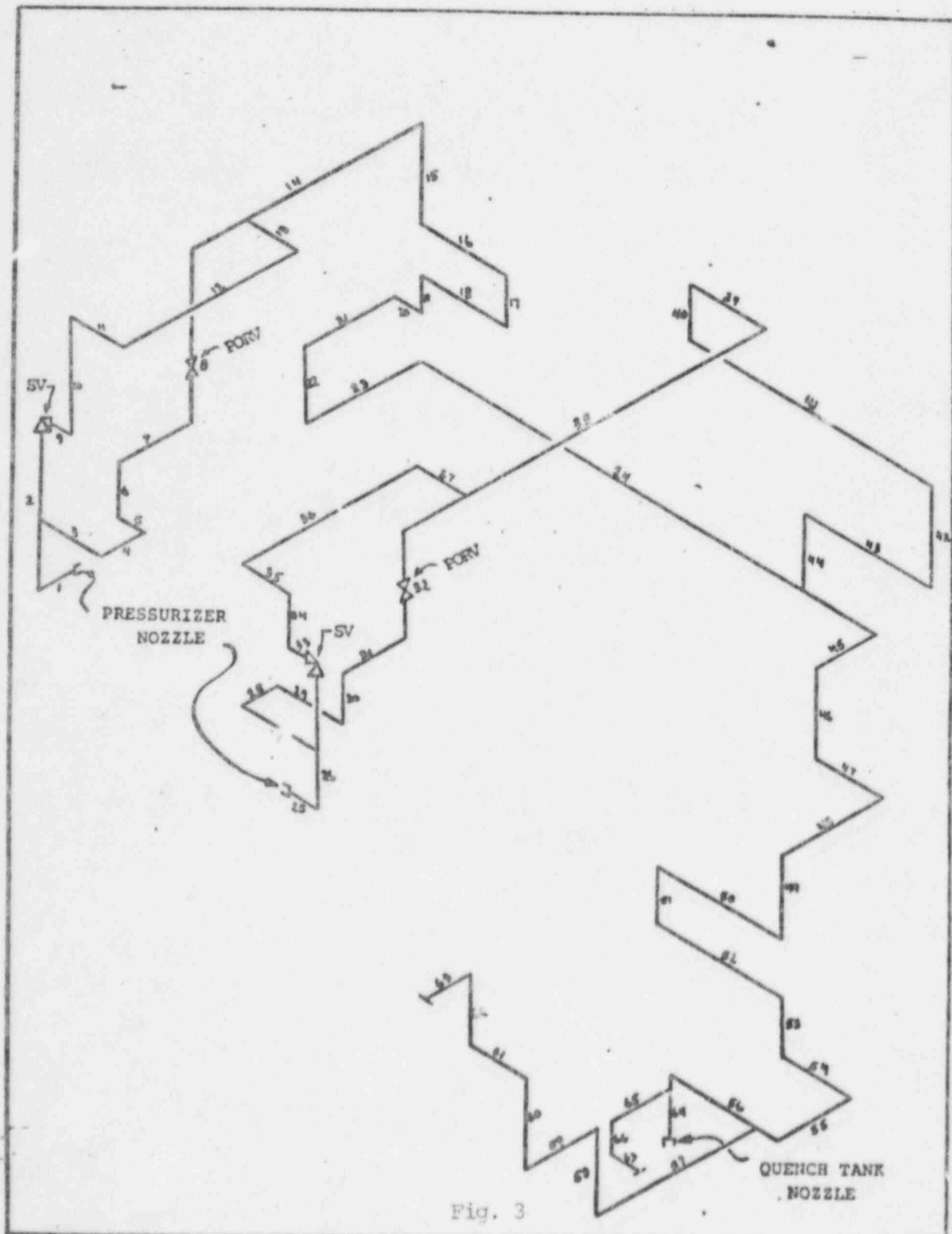


Fig. 3



# CALVERT CLIFFS NUCLEAR POWER PLANT

## THERMAL HYDRAULIC MODEL - UNIT 2

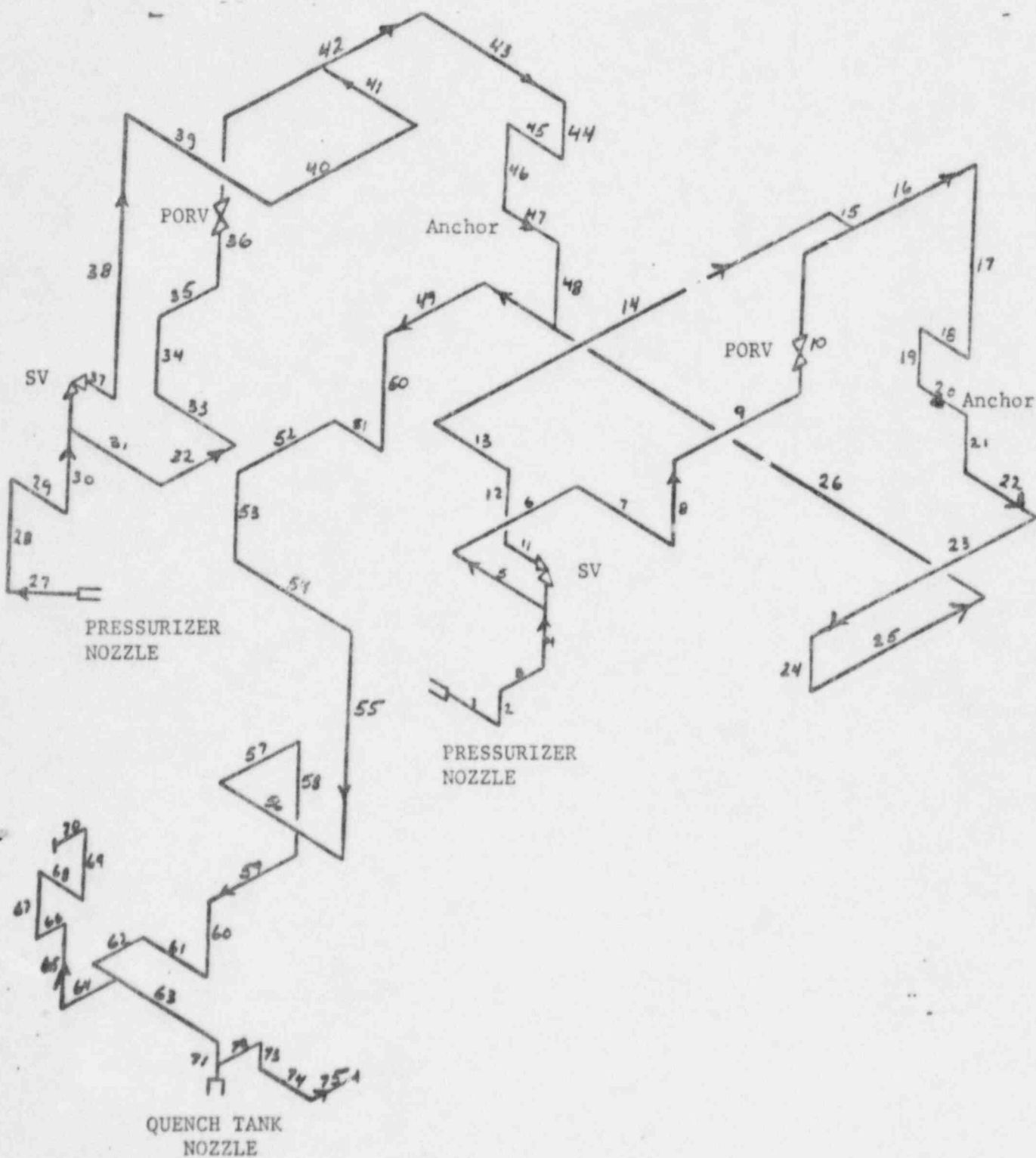


FIG. 4

### Response to Question 11

The opening of the Safeties and PORV's were sequenced according to their set pressures and the relief valve inlet condition report by C.E (See Reference 10). In all of the C.E. analyses, the PORV's were not credited with operation. Therefore, it was assumed that in no case would the safeties and PORV's open together. However, the loss of A-C transient which was analyzed would envelope the case of the PORV's and single safety valve opening. The loss of A-C transient analysis assumed that both safety valves opened and since the mass flow rates from a safety valve is approximately the same as from two PORV's, the opening of two safety valves is equivalent to the opening of two PORV's and a safety valve. Also, the loss of A-C transient analysis is conservative since, according to the peak pressure reported by C.E. (See Reference 10) the second safety would not have opened.

### Loss of A-C (See Reference 10)

Max. Pressure	=	2538 psia
Pressure Ramp Rate	=	64.4 psi/sec.
Valve Flow (avg.)		297,000 lbm/hour per valve

First safety opens at 2500 psia (setpoint = 2485 psig), second safety was modelled to open at 2530 psia (setpoint = 2550 psig).

The PORVs opening was analyzed using the Loss of Load scenario -

Loss of Load (See Reference 10)

Max. Pressure	=	2434 psia
Pressure Ramp Rate	=	46.0 psi/sec.
Valve Flow (avg.)		156,000 lbm/hour

Both valves open at their setpoint of 2400 psia.

Response to Question 12

The ME-101 program has been verified against independent piping program EASE, detailed manual checking and the ASME benchmark problem.

The forcing function was reviewed in detail and it was decided to carry out the analysis to include all modes up to 100HZ. Modal superposition of the response from all modes up to 100HZ would yield a total response of the system.

Response to Question 13

The following loading combinations and stress allowables were used for piping stress analysis and support evaluations:

Combinations	Plant/System Operating Condition	Load Combination	Allowable Stress for Class II Ppg.	Allowable Stress for Supports
1	Normal	N	$S_h$	$0.6 S_y$
2	Upset	$N+SOT_U+OBE$	$1.2 S_h$	$0.6 S_y$
3	Emergency	$N+SOT_E$	$1.8 S_h$	$0.8 S_y$
4	Faulted	$N+DBE+SOT_E$	$S_y$	$0.9 S_y$
5	Faulted	$N+DBE+SOT_U$	$S_y$	$0.9 S_y$

Where

N = Sustained loads during normal plant operation

$SOT_U$  = Relief valve discharge transient

$SOT_E$  = Safety valve discharge transient

OBE = Operating basis earthquake

DBE = Design basis earthquake

$S_y$  = Minimum yield stress

$S_h$  = Stress allowable at operating temperatures

For support design, thermal loads were added to the above combinations. The dynamic responses were combined by SRSS (square root of the sum of the squares).

The Nuclear Class I portion of the piping was analyzed using the above loading combinations and USAS B31.7 1969 Code as indicated below.

1. Primary stress intensity limit for each point analyzed in accordance with Code requirements 1-705.1 (Equation 9).
2. Primary plus secondary stress intensity range for each point analyzed in accordance with Code requirements 1-705.2 (Equation 10) and 1-705.4 (Equation 12 and 13).
3. Cumulative damage for each point analyzed in accordance with Code requirements 1-705.3.4.

#### Response to Question 14

The piping system was analyzed for all load cases as indicated in our response to question 13. The maximum usage factor for the Nuclear Class 1 pipe is 0.0084 versus the allowable of 1. The maximum stresses for the Class 2 pipe is summarized in the following table.

Plant Event	Calculated Stress (PSI)	Allowable Stress (PSI)	<u>Calculated Stress</u> <u>Allowable Stress</u>
Normal	6416	14362	0.438
Upset	14825	17558	0.844
Emergency	19981	26338	0.759
Faulted	22731	30000	0.758

### Response to Question 15

As stated in the submittal, ten supports on Unit 1 were identified as requiring modifications and four supports as requiring field information before a final decision could be made. As a result of the field walkdown, a total of 14 supports required some form of modification on the Unit 1 system. On the Unit 2 system, twelve supports were identified in the submittal as requiring modification, of which one support modification was eliminated after further detailed analysis. Also, it should be noted that three of the supports requiring modification were snubbers which were either deleted or replaced with sway struts.

A linear elastic analysis was used in the evaluation of both the piping system and supporting structures as indicated in the submittal. As previously stated the structural analysis utilized appropriate loads and load combinations (See Response to Question 13) in determining support capacities. It was indicated in the submittal that the analysis of the supports for this piping shows local yielding of some members based on elastic analysis criteria, and that by using a more detailed non-linear analysis criteria, may show that the supports are adequate. Considering the behavior of the entire system, we have concluded based on our judgement and experience, that the system will remain functional during and after relief valve operation coincident with an OBE (the most severe load combination). We therefore have not proceeded with a non-linear analysis and obviously a direct comparison of calculated stress levels to elastic allowables is not meaningful. In addition not all modifications were due to criteria related to member stresses but to other items routinely considered in good pipe support design. The following tables B and C give a comparison of critical components for the modified supports to the appropriate criteria when applicable.

Table - B

Unit No. 1

<u>Support</u>	<u>Critical Component</u>	<u>Repaired Value</u>	<u>Allowable</u>
SK-19012	Excessive Gap in Support	-	-
SK-25008	Concrete Expansion Anchor (interaction)	0.78	1.0
SK-33649	Flexural Stress	6.6ksi	20.6ksi
SK-17500	Weld Stress	4.16ksi	13.8ksi
SK-17505	Clearance for Thermal Growth	-	-
SK-17521	Flexural Stress (common support structure)	6.4ksi	20.6ksi
SK-17522	Flexural Stress (common support structure)	6.4ksi	20.6ksi
SK-19022	Clamp Stress - Clamp replaced w/load rated U-Bolt	-	-
SK-19022	Concrete Expansion Anchor Interaction	0.92	1.0
SK-19031	Weld Capacity (Stress) See SK-19026	8.0ksi	13.8ksi
SK-19026	Weld Capacity (Stress) See SK-19031	8.0ksi	13.8ksi
SK-20900	Weld Capacity (Stress)	12.6ksi	13.8ksi
SK-40324	Concrete Expansion Anchor (Interaction)	0.46	1.0
SK-19027	Weld Capacity (Stress)	Very Small	13.8ksi

Table - C

Unit No. 2

<u>Support</u>	<u>Critical Component</u>	<u>Repair Value</u>	<u>Allowable</u>
SK-2-12720	Concrete Expansion Anchor (Factor of Safety)	8.3	4.0
SK-2-12720	Excessive Support Deflection (Inch)	1/16	1/16
SK-2-12720	Plate Flexural Stress	Very Small	27ksi
SK-2-12722	Double Acting Support Req.	-	-
SK-2-12723	Double Acting Support Req.	-	-
SK-2-17508	Excessive Gap in Support	-	-
SK-2-17508	Flexural Stress	6.43ksi	20.6ksi
SK-2-19025	Clamp Stress - Clamp replaced w/load rated	-	-
SK-2-19031	Concrete Expansion Anchor (Interaction)	0.85	1.0
SK-36704	Double Acting Support Req.	-	-
SK-36704	Concrete Expansion Anchor (interaction)	0.77	1.0
SK-36671	Weld Stress	Very Small	13.8ksi
SK-36711	Deletion of Snubber	-	-
SK-36747	Replace Snubber w/sway strut	-	-
SK-36748	Replace Snubber w/sway strut	-	-

Responses to NRC Questions  
on Calvert Cliffs Units 1 and 2 PORVs and Safety Valves

References:

1. (NRC) Request for Additional Information, TMI Action NUREG-0737 (II.D.I.) for Calvert Cliffs, Units 1 and 2, Docket NO.: 50-317 and 50-318, June 1985.
2. EPRI PWR Safety and Relief Valve Test Program Test Condition Justification Report, EPRI NP-2460-LD, Interim Report, June 1982.
3. EPRI PWR Safety and Relief Valve Test Program Guide for Application of Valve Test Report Results to Plant-Specific Evaluations, Interim Report, March 1982.
4. A. Singh, On the Stability of a Coupled Safety Valve-Piping System, EPRI Transmittal, July 2, 1982. (Also in "Testing and Analysis of Safety/Relief Valve Performance", ASME, 1983).
5. BG&E Calvert Cliffs Nuclear Power Plant Units 1 and 2 Pressurizer Safety Valve Operability Report, CEN-248 (B), Rev. 2, January 1984.
6. Summary Report on the Operability of Pressurizer Safety Valves in C-E Designed Plant, CEN-227, December 1982.
7. Calvert Cliffs Units 1 and 2 PORVs, Dresser (R. S. Huffman) Letter to C-E (R. J. Quinn), August 12, 1985. (Attached)

8. Calvert Cliffs Units 1 and 2 PORV's Dresser (R. S. Huffman) Telex to C-E (R. J. Quinn), August 29, 1985.  
(Attached)
9. BG&E Report on the Operability of Pressurizer Power Operated Relief Valves in Calvert Cliffs Nuclear Power Plant, Units 1 & 2, CEN-246(B), April 1983.
10. EPRI Report - Valve Inlet Fluid Conditions for Pressurizer Safety and Relief Valve in Combustion Engineering - Designed Plants, EPRI NP-2318-LD, Interim Report April 1982.
11. EPRI/C-E PWR Safety Valve Test Report Volume 3 of 10 Test Results for Dresser Safety Valve Model 31739A, EPRI Research Project V102-2 Interim Report. July 1982.
12. EPRI/Marshall Power Operated Relief Valve Test Data Report, Draft Copy, August 19, 1981.
13. EPRI PWR Safety and Relief Valve Test Program Status Report, March 1981.

**DRESSER  
INDUSTRIES**



INDUSTRIAL VALVE OPERATIONS □ BOX 1430 □ ALEXANDRIA, LOUISIANA 71301  
TEL: 318/640-2250 □ TWX: 510-976-5733 TELEX: 58-6423 CABLE: DIVID

August 12, 1985

Combustion Engineering, Inc.  
P. O. Box 500  
Windsor, Connecticut 06095-0500

Attention: Mr. R. J. Quinn  
Plant Components

Subject: Baltimore Gas and Electric Company  
Calvert Cliffs Units 1 and 2  
Power Operated Relief Valves  
CE PO 9903304 and 9903305

Gentlemen:

We have been able to verify that the Electromatic relief valves, type 31533VX-30, at the Calvert Cliff Nuclear Plant were modified to the -2 design.

Original equipment drawing CP-1356 was revised (R #EV555, dated 6-17-85) to reflect this upgrading. A copy is attached.

Replacement valve (serial no. BT06815) was of the -2 design (drawing 3NC-069).

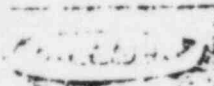
The purpose of -2 design modification was to improve seat tightness.

Regarding minimum operating pressure, see report SV-203A attached. Based on this test and others, we can conservatively conclude that the minimum operating pressure is 75 psig.

Sincerely,

R. S. Huffman  
Sr. Product Engineer

RSH/sc  
Attachment

**DRESSER  
INDUSTRIES**INDUSTRIAL VALVE OPERATIONS □ BOX 1430 □ ALEXANDRIA, LOUISIANA 71301  
TEL. 318/640-2250 □ TWX: 510 976-5733 TELEX: 58-6423 CABLE: DIVIDReport SV-203A  
June 30, 1983Test Data: 6-29-83  
Valve Serial No.: BN04377

Valve heat-up started at 9:05 a.m.; first pop occurred at 9:44 a.m. Trouble was experienced when loading valve with the by-pass until the dead weight of the lever was removed from the pilot stem. The pilot would blow and the main disc would chatter. The valve immediately loaded when the dead weight was removed from the stem.

<u>Run</u>	<u>Time</u>	<u>Remarks</u>
1	9:44 a.m.	Popped at 1805 psig, electrically
2	9:45 a.m.	Popped at 1805 psig, electrically
3	9:46 a.m.	Popped at 1805 psig, electrically
4	9:50 a.m.	Checked seat leakage at 2010 psig. No leakage detected.
5	10:02 a.m.	Blew down to 500 psig, held for 5 minutes and checked for seat leakage. No leakage detected.
6	10:06 a.m.	Blew down to 400 psig, held for 2 minutes and checked for seat leakage. No leakage detected.
7	10:09 a.m.	Blew down to 300 psig, held for 2 minutes and checked for seat leakage. No leakage detected.
8	10:13 a.m.	Blew down to 200 psig, held for 2 minutes and checked for seat leakage. No leakage detected.
9	10:16 a.m.	Blew down to 100 psig, held for 2 minutes and checked for seat leakage. No leakage detected.
10	10:19 a.m.	Blew down to 75 psig, held for 2 minutes and checked for leakage. No leakage detected.

-Continued-

Report SV203A  
June 30, 1983

Page 2 of 3

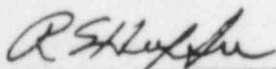
11	10:22	Blew down to 50 psig. Slight leakage on main with no rod (fogging, no droplets).
12	10:24	Blew down to 25 psig. Both seats blowing.
13	10:29	Blew down to 0 psig. Waited 5 minutes. Loaded valve to 25 psig. Both seats blowing. Increased pressure to 50 psig. No apparent leakage. Decreased pressure to 25 psig. Both valves opened. Removed dead weight of lever from pilot stem - both valves closed. This was repeated at least 5 times. In all cases, by raising and lowering lever, the valves would close and open respectively. It was determined that the point at which the dead weight of the lever would open the pilot valve was 33 psig.
20	10:44	Increased pressure 500 psig. Both valves leaking--pilot was visual; main valve by cold rod.
21	10:46	Increased pressure to 1000 psig. Visual leakage disappeared on the pilot at approximately 900 psig. Main valve tight. Pilot leaking by cold rod.
22	10:52	Increased pressure to 2000 psig. Pilot leaking by cold rod at base outlet. With 6" vent nipple installed, no leakage was detected.
20	10:57 10:58 10:58	Popped at 1964 psig, electrically. Popped at 1964 psig, electrically. Popped at 1964 psig, electrically.
21	11:07 11:07 11:07	Popped at 500 psig, electrically. Popped at 507 psig, electrically. Popped at 506 psig, electrically.
22	11:16 11:17 11:18 11:18 11:19	Popped at 51 psig, electrically. Popped at 52 psig, electrically. Popped at 56 psig, electrically. Popped at 40 psig, electrically. Popped at 40 psig, electrically.

Report SV203A  
June 30, 1983

Page 3 of 3

23	11:24	Checked for seat leakage at 50 psig. Slight on pilot only.
24	12:33	Popped at 1906 psig, manually.
	12:34	Popped at 1850 psig, manually.
	12:35	Popped at 1912 psig, manually.
25	12:37	Checked for leakage at 2000 psig. No leakage detected.
	12:38	Test terminated.

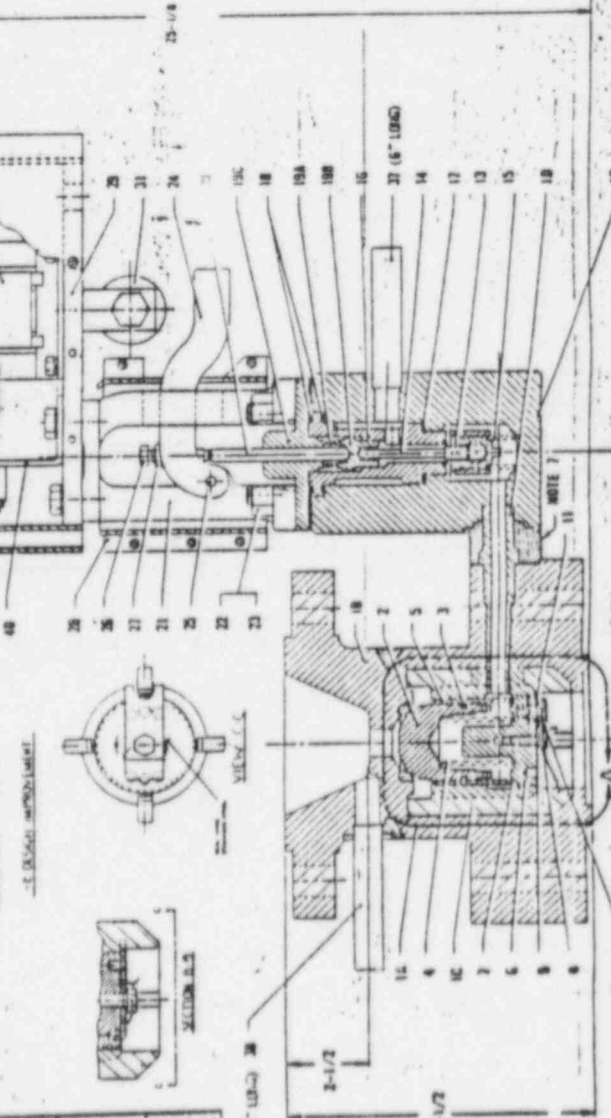
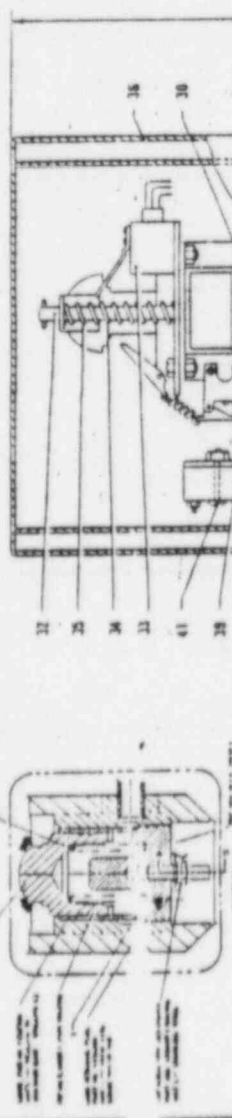
NOTE: For Runs 1-3 and 20-24, the valve opened and closed without failure and with no apparent leakage after closing.



R. S. Huffman  
Sr. Product Engineer

6/30/83

Date



NOTES: 1. ULTRASONIC INSPECTED PER ENG. INST. SP-32.  
 2. LIQUID PENETRANT INSPECTED PER ENG. INST. SP-32-42.  
 3. MAIN BASE ASSEMBLY WELD IS RADIOGRAPHED PER ENG. INST. NG-7.  
 4. ALL WELDS AND SEALING SURFACES ARE LIQUID PENETRANT INSPECTED PER ENG. INST. SP-32-42.  
 5. GENERAL ELECTRIC PART NO. CE-949-213CAF203 MODIFIED WITH CLASS M INSULATION AND COILS WITH ADDITIONAL EPOXY (GELCO) TO RESIST MOISTURE.  
 6. CAPABLE OF CONTINUOUS 200° F OPERATION.  
 7. MAIN VALVE ASSEMBLY TO PILOT VALVE WELD RADIOGRAPHED PER NG-7 REV 4.

ASME SECTION III 3-1/2" - 3153398

1. BRUN WILL CHECK  
 2. ENG. INST. SP58-178  
 3. 4.  
 4. 5.  
 5. 6.  
 6. 7.

WILM DRESSER ENG. INST. OS17B

MODEL 2-1/2" AMSI ST. B. SMALL TONGUE (EXCEPT 3-1/8 FLANGE THICKNESS)

11-65 SW - 20-EC-45 - 91/1  
1/16 59-33-02-45 59-33

RECEIVED PER C.E. SLIP #1 P. 0. 9903304 &amp; 9903305

TLX I RECV Line 2  
CONNECTED 29-Aug-85 14:29 29

COMBEN WSOR BB

DRESSER ALXA

TO: COMBUSTION ENGINEERING, INC.

ATTENTION: R. J. QUINN

FROM: DRESSER INDUSTRIES, INC.  
ALEXANDRIA, LOUISIANA

DATE: AUGUST 29. 1985

RE: 31533VX AT CALVERT CLIFFS

CLARIFICATION TO OUR LETTER OF 8-12-85 REGARDING THE 31533VX AT  
CALVERT CLIFFS:

- 1) TEST MEDIA FOR LOW PRESSURE TESTING (REPORT SV-203A) WAS  
SATURATED STEAM.
- 2) THE VALVE TESTED, S/N BN04377, IS IDENTICAL TO THE CALVERT  
CLIFFS VALVE.
- 3) IT IS ACCEPTABLE TO OPERATE THE 31533VX BETWEEN 75-1000 PSIG ON  
SATURATED STEAM.

REGARDS,

ROLLAND S. HUFFMAN  
SR. PRODUCT ENGINEER

DRESSER ALXA

COMBEN WSOR BB

DISCONNECTED 29-Aug-85 14:31 49