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JOSEPH A. TIERNAN
VICE PRESIDENT
NUCLEAR ENERGY

January 6, 1988

U. S. Nuclear Regulatory Commission
Washington, DC 20555

ATTENTION: Document Control Desk

SUBJECT: Calvert Cliffs Nuclear Power Plant
Unit Nos. 1 & 2; Docket Nos. 50-317 & 50-318
TMI Action Item II.D.1, Performance Testing of Relief and Safety Valves

REFERENCE: (a) Letter from Mr. S. A. McNeil (NRC), to Mr. J. A. Tiernan (BG&E), dated April 30, 1987, Request for Additional Information

Gentlemen:

On April 30, 1987, you asked for additional information regarding our implementation of TMI Action Item II.D.1, Performance Testing of Relief and Safety Valves (Reference a). The attachments to this letter contain our response.

Should you have further questions regarding this subject, we will be pleased to discuss them with you.

Very truly yours,

JATiernan

JAT/WPM/dlm

Attachments

cc: D. A. Brune, Esquire
J. E. Silberg, Esquire
R. A. Capra, NRC
S. A. McNeil, NRC
W. T. Russell, NRC
T. Foley/D. C. Trimble, NRC

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RESPONSE FOR REQUEST FOR ADDITIONAL INFORMATION
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NRC's questions is restated, followed by BG&E's response. Background information that was included in NRC's more lengthy questions was removed for brevity.

1. The plant block valve is Velan B9-354-B-MS while the test valve was a Velan B10-3054B-13MS. Discuss the differences in the valves due to one being an MS and the other being a 13MS. Discuss what impact these differences may have on valve operability.

   RESPONSE:

   In previous correspondence we inadvertently referred to our block valve as an MS. In fact, they are Velan B9-354B-13MS. We regret the error.

2. Provide the torque produced by the Limitorque SMB-00-5 operators. If the torque is less than 82 ft-lbs (the minimum torque tested by EPRI), provide test data to demonstrate the operators are capable of providing adequate torque to close the block valves.

   RESPONSE:

   The operators will be tested using MOVATS during Unit One's spring 1988 refueling outage. The results will be provided to you upon completion.

3. Provide the maximum expected backpressure and bending moment for the Calvert Cliffs 1 and 2 PORVs.

   RESPONSE:

   Calculated steady-state backpressure assuming two PORVs and 2 SVs open:

Unit 1	Unit 2
833 psia ±10%	653psia ± 10%

   Calculated bending moments:

Valve	Unit 1	Unit 2
ERV-404	1214 ft-lbs	1495 ft-lbs
ERV-402	1460 ft-lbs	2380 ft-lbs

Bending moments include loads due to dead weight, thermal, seismic anchor movement, seismic, PORV discharge, and SV discharge loads.

4. Verify that BG&E has installed the heavier springs recommended by Dresser for PORV operation at less than 100 psig.

RESPONSE:

BG&E has installed heavier springs consistent with Dresser's recommendation.

5. Provide documentation to show the PORV control circuitry has been qualified under 10 CFR 50.49. Alternatively, provide information to demonstrate the control circuitry is qualified under NUREG-0737.

RESPONSE:

Negotiations between the NRC Staff and the Nuclear Utility Group on Equipment Qualification (NUGEQ) resulted in the reformulation of this question. Reprinted below is the newly worded question as taken from a July 8, 1987 request for additional information regarding San Onofre Nuclear Generating Station, Unit 1. Our response is directed toward this question and not the one stated above.

> NUREG-0737, Item II.D.1 requires that the plant-specific control circuitry be qualified for design-basis transients and accidents. The licensee should provide information which demonstrates that the above requirement has been fulfilled. The Nuclear Regulatory Commission staff has agreed that meeting the licensing requirements of 10 CFR 50.49 for this circuitry is satisfactory and that specific testing per NUREG-0737 requirement is not required. Therefore, verify whether the PORV control circuitry has been reviewed and accepted under the requirements of 10 CFR 50.49.
>
> If the PORV circuitry has not been qualified to the requirements of 10 CFR 50.49, provide information to demonstrate that the control circuitry is qualified per the guidance provided in Reg. Guide 1.89, Revision 1, Appendix. E.
>
> As an alternative, the staff has determined that the requirements of NUREG-0737 regarding the qualification of the PORV control circuitry may be satisfied if one or more of the following conditions is met.

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a. The PORVs are not required to perform a safety function to mitigate the effects of any design basis event in the harsh environment, and failure in the harsh environment will not adversely impact safety functions or mislead the operator (PORVs will not experience any spurious actuations and, if emergency operating procedures do not specifically prohibit use of PORVs in accident mitigation, it must be ascertained that PORVs can be closed under harsh environment conditions.).

b. The PORVs are required to perform a safety function to mitigate the effects of a specific event, but are not subjected to a harsh environment as a result of that event.

c. The PORVs perform their function before being exposed to the harsh environment, and the adequacy of the time margin provided is justified; subsequent failure of the PORVs as a result of the harsh environment will not degrade other safety functions or mislead the operator (PORVs will not experience any spurious actuations and, if emergency operating procedures do not specifically prohibit use of PORVs in accident mitigation, it must be ascertained that PORVs can be closed under harsh environment conditions).

d. The safety function can be accomplished by some other designated equipment that has been adequately qualified and satisfies the single-failure criterion.

Our PORV control circuitry has not been qualified to the requirements of 10 CFR 50.49. However, the PORVs are not required to perform a safety function to mitigate the effects of our design basis events.

It is expected that the PORVs will open during any event in which the RCS pressure exceeds the PORV setpoint. Should the PORVs fail to close, operators will be alerted via acoustic flow monitors downstream of the valves. Digital indicators are provided on control panels C06 and C31. Additional indications are provided through the plant annunciator. Operators will then close the PORV block valves located upstream of the PORVs, terminating flow.

All electrical equipment for the acoustic monitors and PORV block valves are qualified in accordance with the requirements of 10 CFR 50.49 as applicable.

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6. Your response to Question 2 of our initial RAI stated the liquid discharge case for the PORV during low temperature overpressure protection was not analyzed because it was not considered a design basis event for the piping analysis. Also, it was stated such transients occur only after the plant is in a safe shutdown mode and, therefore, do not constitute a safety concern. This is not an acceptable response. First, the fact the plant is undergoing a transient indicates the plant is not safely shutdown. Also, the intent of NUREG-0737, Item II.D.1 was to show the overpressure protection system, including the piping, will be able to handle all loads imposed by overpressure transients. It is not acceptable to say, as was implied by BG&E's response, that because the plant is in a safe shutdown mode, damage to the PORV piping would be acceptable. Provide the results of a structural analysis using conditions typical of cold overpressure protection at Calvert Cliffs 1 and 2 for our review. Include a comparison of calculated and allowable stresses for the most highly loaded locations. Alternately, provide data to show loads during a low temperature overpressure transient are bounded by other transients and accidents analyzed in the FSAR.

RESPONSE:

We disagree with your characterization of the intent of NUREG-0737, Item II.D.1 and believe the concern described in your question is outside the scope of this TMI item. The "Position" statement of Item II.D.1 in NUREG-0737 states:

> Pressurized-water reactor and boiling-water reactor licensees and applicants shall conduct testing to qualify the reactor coolant system relief and safety valves under expected operating conditions **for design-basis transients and accidents**. (emphasis added)

Since a liquid discharge event during low temperature operation is not a design basis event at Calvert Cliffs, it need not be considered in response to Item II.D.1.

However, we believe your concern may be a valid one with respect to the review of our low temperature overpressure protection system that resulted in your safety evaluation dated November 17, 1977. As such, we will continue to investigate this concern and will report our findings to you. In a follow-up conversation with your staff, we learned that it may be possible to crimp piping or elbows in the PORV lines, thereby restricting flow during a low temperature overpressure (LTOP) event. In the conversation, the NRC Staff alluded to analyses that have been performed that demonstrate this crimping. It would be most effective if we could apply those analyses to Calvert Cliffs to begin to address your concern. Therefore, please provide us with the details of the analyses you alluded to in our meeting.

Be assured, we will be pursuing the LTOP concern with the NRC staff but it should not be a condition to completing this TMI item.

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7. Provide information on the verification of REPIPE. Provide comparisons of the results for REPIPE calculations and EPRI/CE data to verify this code is an appropriate tool to evaluate piping discharge transients.

RESPONSE:

The REPIPE computer program was used as the force post-processor to RELAP5 for the Calvert Cliffs, Units 1 and 2, Pressurizer Safety and Relief Valve Piping Transient Evaluation. This program was developed by Control Data Corporation, and has been validated by Bechtel Power Corporation for nuclear power plant applications. REPIPE solves the one-dimensional momentum equation for force on a control volume (Ref. 1) using time-dependent thermodynamic parameters output from RELAP5. Individual control volume forces can be assembled to obtain net reaction forces on pipe segments, typically bounded by elbows or large reservoirs (for pipe rupture consideration). The REPIPE program meets all of the requirements of the Bechtel Power Corporation Quality Assurance Manual and Engineering Department Procedures regarding "Standard Computer Programs." A complete set of user and validation documentation, including recommended modeling guidelines, is maintained for the program. The documentation is considered proprietary, but can be made available for audit by the Nuclear Regulatory Commission at any Bechtel office on request. The Validation Manual addresses typical applications of the REPIPE program when used in conjunction with the thermal-hydraulic analysis program RELAP5. Industry standard benchmark problems are used for comparison to program results. The benchmarks include Edward's and O'Brien's experiments (Ref. 2) and Hanson's subcooled blowdown force experiments (Ref. 3) for subcooled liquid conditions, and Moody's analytical/experimental three pipe segment vessel blowdown with saturated steam conditions (Ref. 4).

The Joukowski analytical solution for instantaneous valve closure (Ref. 5) was also evaluated. REPIPE results provide excellent agreement with all these problems, provided that recommended modeling guidelines are followed for both REPIPE and for RELAP5. Additionally, the EPRI/CE test 908 with a Crosby 6M6 safety valve and cold water loop seal has been analyzed with the RELAP5 and REPIPE programs. Details of this analysis and comparison of REPIPE to test results and to other force post-processor methods are presented in the Validation Manual and also in the attached paper (Ref. 6, Attachment B), "Comparison of Analytical and Experimental Results for a PWR Pressurizer Safety Valve Discharge," presented at the Third Multiphase Flow and Heat Transfer Symposium-Workshop, Miami Beach, in April 1983. While the piping geometry with a cold loop seal does not depict the Calvert Cliffs pressurizer safety and relief line piping configuration, this benchmark does confirm the RELAP5/REPIPE methodology and modeling approach. Benchmarks have also been performed for RELAP5/REPIPE against Moody's graphical solution for blowdown force resulting from saturated steam source conditions, with an upstream restriction in the pipe (Ref. 4), again with excellent agreement.

8A. BG&E's October 23, 1985 submittal states the safety valve flow area input to RELAP5 was 1.4 x 10^{-2} ft^2 resulting in a mass flow rate of approximately 297,000 lbm/hr. EPRI reported an area of 1.77×10^{-2} ft^2 and a mass flow rate of 323,000 lbm/hr. Justify use of the smaller flow area.

RESPONSE:

Prior to performing the Calvert Cliffs safety and relief valve piping analysis, a RELAP5/MOD1 model was developed to simulate the EPRI/CE test for the Dresser Model 31739A safety valve with steam inlet conditions. The objective of this model was to establish an appropriate RELAP5 safety valve model to match the test pressure and flow conditions. This model was then used for the Calvert Cliffs piping configuration and inlet pressure. The valve bore area was given as 2.545 in^2 (1.77×10^{-2} ft^2), but the actual valve discharge coefficient was not known. The safety valve was modeled in RELAP5 using the motor valve option, with the single-phase abrupt area change model to account for vena contracta losses at the throat. No external losses (due to the valve geometry) were applied. Use of the actual bore area with no external loss resulted in over-prediction of the test flow rate with steady state inlet pressure of approximately 2600 psia. The effective valve throat area was reduced accordingly to match the test flow rate as closely as possible. The resulting effective area is in agreement with that determined from the compressible, critical flow equation, or that determined using Moody critical mass flux (Ref. 7).

The same valve model was used for the Calvert Cliffs safety and relief valve piping analysis. The resulting mass flow rate of approximately 297,000 lbm/hr was lower than the test flow rate, since the pressurizer pressure used for the valve inlet condition was lower than that of the test.

8B. The time step used in the piping analysis was said to vary from 2.5×10^{-4} to 5.0×10^{-4} seconds. To prevent the shock wave generated by a valve discharge from passing through a volume in one time step the time step should be 1.265×10^{-4} seconds. Justify the time steps used in the RELAP5 analysis.

RESPONSE:

As stated in the initial request for additional information (RAI) response, the RELAP5 time step used in the Calvert Cliffs safety and relief valve piping analysis varies between 2.5×10^{-4} and 5.0×10^{-4} seconds. This information was based solely on RELAP5 major edit summaries of the time step control convergence. The specified minimum time step for all cases was 1×10^{-7} seconds. A small percentage of the total time step advancements used less than 2.5×10^{-4} seconds. The automatic time step control scheme of RELAP5 (Ref. 8) used several criteria for assessment of the adequacy of the selected time step. For the saturated steam conditions appropriate for the Calvert Cliffs analysis, the two significant criteria are the acoustic or mass transport Courant limit (represented simply by $\Delta t = \frac{\Delta x}{a}$, where 'x' is the minimum control volume length and 'a' is the sonic speed) and the density difference between that calculated using the state equations. If either is not satisfied with the maximum input time step (5×10^{-4}), the time step is halved and the evaluation repeated. The process is repeated until the convergence criteria are satisfied or the minimum input time step (1×10^{-7} seconds) is reached. If the criteria are still not satisfied, the time step is flagged as unsuccessful and the run proceeds to the next transient iteration step. All of the RELAP5 runs for the safety and relief valve transient analysis indicated an insignificant number of repeated or unsuccessful advances.

The potential for supersonic shock valve transition was not expected to occur in the small piping upstream of the valves (since the geometry is not conducive to supersonic flow), but was considered for the larger downstream piping. Allowing for the larger diameter and corresponding increased node length, the 2.5×10^{-4} second time step is small enough to account for supersonic shock transition, if predicted to occur. This approach is consistent with Reference 9. Supersonic velocities were predicted by RELAP5 at enlarging tees in the downstream pipe during the transient, but none were observed in upstream junctions. Additionally the input minimum and maximum time steps used are consistent with those used for RELAP5/REPIPE benchmarks (see response to Question #7).

9. The structural analysis was performed using Bechtel's computer program, ME-101. Provide more details on the verification of ME-101, including comparisons of calculated results and EPRI/CE data for our review.

RESPONSE:

The ME-101 Program has been verified against the following standard piping programs.

- ME-632
- EDS SUPERPIPE
- NUPIPE
- TRIPE
- ADINA
- MSC/NASTRAN
- EASE2
- ANSYS
- PIPESD
- Pressure vessel and piping 1972 computer programs verification, The American Society of Mechanical Engineers

The comparison of calculated results of force time history analysis is attached with this response for your information (see Attachment C). The ME-101 verification manual is considered proprietary to Bechtel. However, the manual can be made available for further review by the Nuclear Regulatory Commission at any Bechtel office upon request.

10. More information on the integration time step used in the structural model is needed. Also, provide information on the structural model lumped mass spacing and the damping factors used in the analysis. Justify that all of the desired structural frequencies would be accounted for in the structural response with these modeling techniques.

RESPONSE:

The forcing function generated by RELAP5 and REPIPE was carefully reviewed before selecting the time steps and cut-off frequency. The time steps of 0.001 second was chosen so that all the peaks in the force time history were accounted for as accurately as possible. Further the forcing function was reviewed and all significant contributory modes and their frequency contents were evaluated and it was determined that at a cut-off frequency of 100HZ, all the significant harmonics will be accounted for. The dynamic analysis was carried out to include all harmonic responses up to 100 HZ. The total response of the system was calculated by superposition of the response from all modes up to 100 HZ. The time history analysis was performed using a conservative one percent of critical damping. The lumped masses are carefully located to adequately represent the dynamic properties of the piping system. A lumped mass is located at the beginning and end of every elbow, valve, at the extended valve operator, intersection of every tee or branch connection and at least two mass points between two supports in the same direction. There is a total of approximately 450 ft of pipe per unit, from pressurizer nozzle to the quench tank, represented by 335 lumped mass points.

11. Provide a discussion on whether additional moisture from the pressurizer spray was considered in determining valve inlet conditions and in the analysis done to select the transient producing the maximum loads on the safety valve discharge piping.

RESPONSE:

The operation of pressurizer spray will not increase the discharge piping peak loads because the peak load occurs prior to the time when any wet steam due to entrained spray can reach the safety valve.

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The maximum discharge piping loads occur upon valve opening. This means that the peak force in a given piping segment occurs when the initial pressure surge due to valve opening reaches that segment. The inlet piping for the safety valve will initially contain saturated steam. In order for any postulated wet steam to reach the discharge piping, the initial quantity of saturated steam must pass through the safety valve. For Calvert Cliffs Units 1 and 2, this would take at least 0.015 seconds after the valve initially opens. By the time the postulated wet steam reaches the valve, the valve is fully open and the initial pressure surge has already occurred. This is further substantiated by EPRI safety valve test data for steam-to-water transition tests. In these tests the safety valve actuated on saturated steam followed by a transition of saturated water after the valve opened. The peak loads occurred when the valve initially opened prior to the transition to water. Therefore, the operation of pressurizer spray will not result in discharge piping loads in excess of those values previously presented in the Calvert Cliffs safety valve report.

In addition to the ASME Code report, the EPRI test program demonstrated the structural adequacy of the safety valve during valve actuation transients.

Also, the bending moments predicted to act on the safety valve discharge flange in the Calvert Cliffs piping analysis are less than those measured during the test program. The operability of the safety valves is therefore not impared by the calculated piping loads.

12. Pressurizer nozzle loads during safety valve and PORV discharge were not discussed. Compare the calculated and allowable loads for the pressurizer nozzles.

RESPONSE:

The following table compares the CE allowable loads (Ref. 11) with the calculated loads for all four nozzles. In all but one case the calculated loads are considerably lower than the CE allowable. The "Fy" load for nozzle number two for pressurizer eleven is slightly higher (2.6%) than the allowable. however, the resultant force is considerably less than the resultant allowable and it was considered acceptable. The calculated loads include the loads due to dead weight, thermal, seismic, seismic anchor movement and PORV and safety discharge loads.

Calculated/Allowable Load

	Calculated Loads Pressurizer No. 11		Calculated Loads Pressurizer No. 21		Allowable Loads
	Nozzle No. 1	Nozzle No. 2	Nozzle No. 1	Nozzle No. 2	
FX (lbs)	1010	1170	1490	2120	5472
FY (lbs)	6220	6920	3120	4440	6743
FZ (lbs)	3390	3780	3310	2550	5472
Mx (ft-lbs)	3010	3430	1560	3010	9208
My (ft-lbs)	2120	1610	3340	2210	8859
Mz (ft-lbs)	2560	3470	1560	2200	9208

13. In your response to Question 13 of our initial RAI, it was stated the Class I piping was analyzed using the given load combinations and the USAS B31.7 1969 Code as indicated by a series of referenced Code sections and equations. This information was not specific enough to allow identification of the load combinations and allowable stresses for the Class I piping without considerable effort. Therefore, provide the specific load combinations and code allowables used in the Class I piping stress analysis.

RESPONSE:

The following loading combinations and stress allowable were used for the Nuclear Class I piping.

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Combinations	Plant/System Operating Condition	Load Combination	Allowable Stress
1	Normal	N	1.5 Sm
2	Upset	$N+SOT_U+OBE$	1.5 Sm
3	Emergency	N+SOT	1.5 Sm
4	Faulted	$N+DBE+SOT_E$	3 Sm
5	Faulted	$N+DBE+SOT_U$	3 Sm

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

Sm = Allowable design stress intensity value at operating temperatures

Also, as indicated in response to Question #13 of the initial RAI, all Nuclear Class 1 piping was analyzed using the above loading combinations and USAS B31.7 1969 Code as indicated below.

- o Primary stress intensity limit for each point analyzed in accordance with Code requirements 1-705.1 (Equation 9) using the allowable indicated in above table.

- o 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).

- o Cumulative damage for each point analyzed in accordance with Code requirements 1-705.3.4.

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14. In response to Question 14 of our initial RAI, it was stated the maximum usage factor for the Class I piping was 0.0084 versus an allowable of 1.0. Since the usage factor only compares the number of actual cycles to the number of design cycles, this number does not provide any information on the calculated versus allowable stresses for the Class I piping. Provide a table for the Class I piping comparing the calculated and allowable stresses for the most highly loaded locations.

RESPONSE:

The following table summarizes the five highly loaded points and the stress values are compared against the code allowable for Class I piping.

Data Point	Equation 9 Stresses: Calculated Stress (ksi)	Equation 9 Stresses: Allowable Stress (ksi)	Equation 10 Stresses: Calculated Stress (ksi)	Equation 10 Stresses: Allowable Stress (ksi)
4A	12.77	24.64	32.106	49.275
12	14.70	24.64	40.667	49.275
41	9.53	24.64	36.798	49.275
43	11.80	24.64	32.634	49.275
37	9.53	24.64	39.665	49.275

15. The allowable stresses for the Class I and II piping supports were given as a fraction of the minimum yield stress. Provide the specific reference in the USAS B31.7 1969 Code that defines the allowable support stresses in this way.

RESPONSE.

The piping supports addressed in response to Question #13 of the initial RAI, consist of manufacturer's standard components attached to piping, such as struts and snubbers, as well as structural steel rolled sections used as supplementary steel to transfer loads to the building structure. The evaluation of standard components was based on the manufacturer's published load ratings. The structural rolled sections were evaluated using stress allowables based on the American Institute of Steel Construction (AISC) standard practice as prescribed by Section 1-720.2.4 of the USAS B31.7 1969 Code.

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Section 1.5 of the Specification for the Design Fabrication and Erection of Structural Steel for Buildings, as found in the Eighth Edition of the AISC Steel Construction Manual, defines allowable stresses as a fraction of the minimum yield stress. This methodology is commonly used in the industry.

16. Provide information to show the support modifications required to reduce stresses to within code allowables were completed. If the required modifications have not yet been made, provide a schedule outlining when this work will be complete.

RESPONSE:

The required support modifications were completed.

17. Bechtel's report on the discharge piping system identified the conditions analyzed for the PORV as max pressure, 2538 psia, and pressure ramp rate, 46.0 psi/sec. The safety valve conditions were 2534 psia and 64.4 psi/sec for the maximum pressure and pressure ramp rate, respectively. In BG&E's response to our initial RAI the conditions analyzed were identified as 2434 psia, maximum pressure, and 46.0 psi/sec, pressure ramp rate, for the PORVs. The safety valves were analyzed for 2538 psia and 64.4 psi/sec. Identify the actual conditions analyzed for Calvert Cliffs 1 and 2.

RESPONSE:

The appropriate boundary conditions for the Calvert Cliffs safety and relief valve opening transient are identified in the table below. A small conservatism was incorporated in the RELAP5 safety valve model, so that a maximum pressure of 2538 psia was applied for both PORV and safety valve opening transients. The response to Question 10 of the initial RAI submittal incorrectly reported the maximum PORV pressure used in the analysis.

Transient	Valves	Max. Pressure	Ramp Rate
Loss of Load	PORV	2538 psia	46.0 psi/sec
Loss of AC	Safety	2534 psia (2538 actually used)	64.4 psi/sec

18. The CE inlet conditions report listed the FSAR transients and accidents for each plant which result in a peak pressure greater than the safety valve setpoint. For some plants, this list included the feedwater line break (FWLB), but for other plants the FWLB was not included. Calvert Cliffs 1 and 2 was a plant that did not include the FWLB in its list of transients and accidents that challenge the safety valves. From the CE report it was not clear whether the FWLB was missing because Calvert Cliffs was licensed prior to the issuance of Reg.

Guide 1.70, Rev. 2 and, therefore, the FWLB was not initially analyzed as part of Calvert Cliffs' design basis. Discuss why the FWLB was not listed for Calvert Cliffs. If the FWLB was not listed for the second reason discussed above, it is the staff position that the Calvert Cliffs submittal is incomplete. Item II.D.1 in NUREG-0737 specifically requires that PORVs and safety valves be qualified for fluid conditions resulting from transients and accidents referenced in Reg. Guide 1.70, Rev. 2. Additionally, from the staff review of other plant-specific responses to Item II.D.1, it is clear that the many plants the FWLB accident is the limiting case for providing high pressure liquid to the safety valves, a fluid for which they were not specifically designed originally. This is exactly the type of concern that NUREG-0737, II.D.1, was established to address. In accordance with the requirements of the NUREG, we require that information be provided to demonstrate that the PORVs and safety valves will function as required to assist in safe shutdown of the plant and will not experience any degradation that would inhibit safe plant shutdown if exposed to the FWLB.

RESPONSE:

Feedwater line break was missing from the CE report because, at the time, FWLB was not considered a design basis event and we understood the requirement of Item II.D.1 to address design basis events in each plant's FSAR.

In conversations with NRC Staff subsequent to your question, it was explained that the intent of Item II.D.1 is to address the transients and accidents referenced in Reg. Guide 1.70, Rev. 2, regardless of whether these transients and accidents are in our FSAR. Therefore, we should consider FWLB whether it is a design basis event or not. We accept this position and will respond no later than March 31, 1988, concurrent with the related open item: Is feedline break a design basis event at Calvert Cliffs? We will demonstrate the operability of the PORVs and safety valves at that time, regardless of the answer to that question.