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EVALUATION OF THE OCTOBER 28, 1983
SCRAM EVENT AT THE
DUANE ARNOLD ENERGY CENTER

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1.0 INTRODUCTION

This report presents an independent evaluation performed by General Electric (GE) of the transient event which occurred on October 28, 1983 in the Duane Arnold Energy Center (DAEC). The event was a Group I isolation, i.e., closure of all main steam isolation valves (MSIVs), which occurred while the plant was operating at approximately 44% of rated power. The evaluation was performed to determine the possibility of discharging two-phase flow through the safety relief valves (SRVs), and, if necessary, the potential impact of the two-phase flow on the SRVs and their discharge piping. The evaluation was also performed to determine whether there was a possibility for the SRVs to open with the steamlines completely filled with subcooled water at high reactor pressure. The evaluation included a computer simulation of the transient event based on data recorded during the event and provided by Iowa Electric Power and Light Company.

2.0 SUMMARY AND CONCLUSIONS

An evaluation of the October 28, 1983 scram event at the Duane Arnold Energy Center (DAEC) was performed with the GE blowdown code (SAFE). It incorporated realistic assumptions, such as ANS 1979 nominal decay heat and available plant data and provided a realistic simulation of the scram event. The significant results of the analysis are as follows:

(1) The reactor water level was below the main steamlines prior to the SRV opening. Therefore, the SRVs did not open with the steamlines filled with subcooled water.

(2) As a result of water level swell during depressurization with the SRVs, two-phase fluid was discharged through the SRVs for a duration of approximately 25 seconds. Although discharging two-phase fluid during SRV blowdown is an unlikely event, it has always been recognized that a potential for this type of event exists. However, the event is not expected to cause damage to the SRVs or their piping and supports.

The calculated load for discharging two-phase flow during the event is less than that for a normal valve opening. Therefore, while the scram event most likely resulted in discharging two-phase fluid through their SRVs, the loads on the SRVs, their piping and supports, were within their design basis and the SRVs operated (opened and closed) in a proper fashion.

3.0 EVENT DESCRIPTION

On October 28, 1983, the Duane Arnold Energy Center (DAEC) experienced a Group I isolation transient, i.e., closure of all main steam isolation valves (MSIVs), while operating at approximately 44% of rated power. The event occurred as a result of a high steam tunnel temperature which was caused by a combination of ventilation problems and a minor steam packing leak in the steam tunnel.

The Group I isolation resulted in an automatic reactor scram and a reactor pressure rise. The reactor pressure increased from approximately 950 psig to a peak value of approximately 1000 psig. This initial pressure rise caused a void collapse and a drop of the indicated water level. The void collapse resulted in an increase in feedwater flow. One of the two electric driven feedwater pumps was manually tripped at approximately 1 minute into the event and the second was automatically tripped at the high water level trip (Level 8) of 211 inches above the top of active fuel (TAF) at approximately 2 minutes into the event. At approximately 7 minutes, the Reactor Water Cleanup (RWCU) system was re-established to reduce reactor water inventory. The reactor inventory was released at a rate of approximately 140 gpm. The scram was manually reset at approximately 21 minutes to decrease the Control Rod Drive (CRD) flow from 100 gpm to 40 gpm. The CRD pump was manually tripped at approximately 35 minutes. These actions were intended to lower the water level to allow RCIC to be established for pressure and water level control.

At approximately 40 minutes into the event, the reactor pressure had reached approximately 1050 psig and the water level remained at nearly 250 inches above TAF. One of the low-low set (LLS) SRVs was manually actuated. This caused the other LLS valve to actuate automatically via the LLS logic. The automatically actuated valve closed at its LLS setpoint, approximately 900 psig. The manually actuated valve was held open until reactor pressure reached 750 psig. The void collapse following the blowdown resulted in automatic HPCI and RCIC initiation on low-low water level (Level 2, 119.5 inches above TAF). HPCI initiation was terminated after 45 seconds and RCIC was then used to control reactor pressure and water level at approximately 800 psig and 180 inches above TAF respectively. Additional information on the event is provided in Reference 2. The chronological sequence of the event is summarized in Table 3-1.

TABLE 3-1

SEQUENCE OF OCTOBER 28, 1983 SCRAM EVENT

Reference Time (sec)	Event and Plant Conditions
0(142245 Hr)	Power=708.16 MWt; P=965 psia; Water Level=194 in above TAF; MSIVs began to close; Scram initiated; CRD Flow 100 gpm.
4	Level 3 Trip; Water Level $\leq$170 in above TAF(42.88 ft)*
31	Level 3 Clear; $P_{max} \approx$ 1015 psi.
55	Water Level $\approx$197.5 in above TAF (45.1 ft).*
57	Manual Trip 1 Feedwater Pump (RFP 1P-1A).
88	Level 8 Trip; Water Level $\geq$ 211 in above TAF(46.29 ft)*
107	Second Feedwater Pump Trip on Level 8 (RFP 1P-1B).
113	Last Level 8 Trip Signal; P $\approx$880 psia.
420	RWCU Flow Established, Flow $\approx$140 gpm; P increasing.
1259	Scram Reset, CRD $\approx$40 gpm; P increasing.
2101	Trip CRD Pump (1P-209A).
2128	Reactor High Pressure trip (P$\geq$1050 psia).
2229	Reactor High Pressure trip (P$\geq$1050 psia).
2324	Open SRV 4401; P 1058 psia,
2325	Reactor High Pressure Clear (P$\leq$1050 psia); Open SRV 4407.
2381	SRV 4407 close; P$\approx$915 psia.
2419	Level 8 Clear (Water Level$\leq$ 211 in above TAF).
2433	Level 8 Clear (Water Level$\leq$ 211 in above TAF).
2462	SRV 4401 close; P 765 psia.
2466	Level 3 Trip (Water Level $\leq$170 in above TAF).
2480	Level 2 Trip (Water Level$\leq$119.5 in above TAF).
2512	Level 2 Trip Clear (Water Level$\geq$119.5 in above TAF).

* Reference vessel bottom.

4.0 SCRAM EVENT EVALUATION

4.1 TRANSIENT ANALYSIS INPUT AND ASSUMPTIONS

A transient analysis using actual plant data and best-estimate inputs was performed to simulate the observed behavior of the DAEC during the event. The system analysis was performed using the GE SAFE model which predicts long-term pressure and thermodyanmic behavior of the coolant in the reactor vessel. Actual plant data were utilized where possible. Initial water level, core thermal power, recirculation flow, steam flow and feedwater flow were all based on recorded plant data (Table 4-1).

Other inputs to the SAFE model were adjusted to replicate the recorded system behavior. The trip sequences for the MSIV, feedwater, RWCU, CRD, recirculation, and SRV systems were input to match the times recorded by the process computer during the event. The closing time of the MSIVs was based on the average value obtained from the plant surveillance tests. The exact feedwater flow was estimated from recorded data and was adjusted in the code to match the water level data. The feedwater temperature was assumed to decrease to the condensate temperature of 80°F after the MSIV closure had caused the loss of all feedwater heaters. The RWCU flow rate was assumed constant at 140 gpm. The CRD flow rate was 100 gpm with the condensate as its source. The CRD pump was tripped consistent with the plant data. The recirculation pump flow was assumed constant until the Level 2 trip after the SRV blowdown. The actuation of two SRVs at 2324 seconds was input to the model in agreement with the plant data.

Realistic estimates of the decay heat curve and sensible heat transfer from reactor internals were input to the SAFE model. The decay heat curve was based on the 1979 ANS standard assuming infinite bundle exposure.

4.2 ANALYSIS RESULTS

The realistic system analysis using the SAFE code is in close agreement with the plant response recorded during the event. A comparison of the calculated system pressure to the actual plant response is shown in Figure 4-1. The water level response is shown in Figures 4-2 through 4-4.

An examination of the plant data and the transient analysis shows that this transient is dominated by the substantial subcooled feedwater flow and recirculation flow during the first two minutes of the transient. After the MSIV closure, the reactor pressure increases from 965 psia to 1050 psia. The pressure increase causes void collapse in the core. The high rate of recirculation flow transfers a large amount of subcooled water from the downcomer region to the core, further reducing the core void fraction and steam generation rate. This contributes to the decrease in the downcomer water level and the system pressure. The calculated downcomer water level reached a minimum value of 165 inches above TAF. The minimum pressure was approximately 885 psia. Both calculated values are in close agreement with the plant data.

The pressure decrease is terminated when the feedwater flow is terminated since subcooled feedwater flow is the primary contributor responsible for the pressure decrease. After the feedwater flow was terminated, the reactor pressure remained at approximately 885 psia for about 4 minutes before the pressure starts to climb gradually. The reactor pressure does not increase immediately because the decay heat merely reduces the high subcooling in the system. The pressure climb stops when the SRV is manually actuated at 2324 seconds. The calculated reactor pressure is generally in close agreement with the plant data (Figure 4-1).

Figure 4-2 shows the calculated indicated water level for the first 3 minutes of the transient event. The calculation shows the initial water level drop due to the void collapse and the recirculation pump flow. After the initial drop, the water level increases as the feedwater flow continues. In the actual event, the feedwater flow was terminated shortly after the second feedwater pump was tripped at 107 seconds. Since the coastdown time of the feedwater pumps is not known (estimated to be less than 10 seconds), a conservative value of 30 seconds is used in the analysis. Thus, the water level continues to increase in the analysis after the feedwater pump is tripped at Level 8.

Figure 4-3 shows the calculated and indicated water level for the entire transient. After the feedwater flow is terminated and the RWCU flow is established, the indicated water level begins to decrease gradually. This result is expected because the CRD flow rate is less than the RWCU flow rate. The indicated water level stays more than 220 inches above TAF from 100 seconds to after the SRV actuations. This is in close agreement with the plant data which shows that the narrow range water level instrument (maximum at 220 inch above TAF) was off-scale shortly after the event until the SRVs were actuated. The comparisons shown in Figures 4-2 and 4-3 demonstrate that the calculated indicated water level is in close agreement with the plant data prior to SRV actuation.

Figure 4-4 shows the calculated two-phase water level. This is the actual water level expected during the event. The calculation shows that the water level is below the bottom of the steamline before the

SRV actuations. After the SRV actuations, the water level swells to the steamline elevation as a result of the rapid depressurization. This results in a discharge of two-phase flow into the steamline for approximately 25 seconds. The minimum void fraction of the two-phase flow through the SRVs has been estimated to be between 50 and 80%, depending on whether the two-phase flow spilling into the steamlines and flowing through the SRV is assumed to be separated flow or homogeneous flow.

Figure 4-5 shows the calculated total SRV flow rate is almost doubled when the SRVs begin to discharge two-phase fluid. Therefore, both Figures 4-4 and 4-5 show that the SRVs open initially in a steam environment. When the subsequent depressurization causes the water level to swell, two-phase flow is discharged. Although this is unusual, the potential for this condition has been recognized as indicated in Reference 3. The calculation in Section 4.3 and observations at the plant show that this two-phase flow discharge does not affect the SRVs operability or the integrity of their discharge piping and supports.

The calculated duration of the SRV blowdown is approximately 142 seconds which is in close agreement with the plant data. In the SAFE analysis, only the SRV opening and closing setpionts are modeled. This shows that SAFE is able to provide an accurate simulation of pressure and water inventory during the SRV blowdown. After the blowdown, the code predicts a void collapse inside the reactor. However, the calculated void collapse does not result in an immediate Level 2 trip. The Level 2 trip that actually occurred in the plant was actuated from the wide range level instruments. The output of these instruments can be biased by the flow velocity near the instrument taps. The flow dynamics near the instrument taps as well as the flow dynamics within the instrumentation system including the sensing lines, are not modeled in SAFE. Therefore, the SAFE code does not predict the Level 2 trip. Other than the immediate low water level trip after the SRV blowdown, the SAFE calculation provides a reasonably accurate simulation of the event.

4.3 EVALUATION OF PIPING FORCES

The piping force resulting from the discharge of two phase fluid through the SRVs can be determined by the following equation (Reference 4).

$$\frac{F}{P_\infty A} = K \left(\frac{\rho_{mix}}{\rho_\infty} - \frac{\rho_s}{\rho_\infty} \right) \left(\frac{V_s}{C_\infty} \right)^2 \qquad (1)$$

Where
- F = Force
- P = Pressure
- A = Discharge Pipe Area
- K = Gas Ratio of Specific Heats
- ρ = Density
- V = Velocity
- C = Sonic Speed
- s = Shock State
- ∞ = Atmospheric State

Applying the conditions of the two-phase mixture obtained from the transient analysis (Section 4.2) in the above equation, the force (F) is determined to be approximately 17,000 lbf.

For comparison purpose, the shock force due to the sudden opening of the SRV in a steam environment is evaluated based on the method given in Reference 4. The shock force for a sudden valve opening is approximately 23,000 lbf. Thus, the piping force resulting from the two-phase fluid discharge experienced during the transient event is less than the shock force from a sudden valve opening. Therefore, the potential loads that might have resulted from two-phase fluid discharge are less than the normal opening loads for SRV actuation in a steam environment.

TABLE 4-1

INITIAL CONDITIONS

Reactor Power	=	708.6 MWt
Reactor Pressure	=	965 psia
Water Level	=	44.82 ft (192 in above TAF)
Steam Flow	=	768 lbm/sec
Feedwater Flow	=	768 lbm/sec
Core Flow	=	7280 lbm/sec

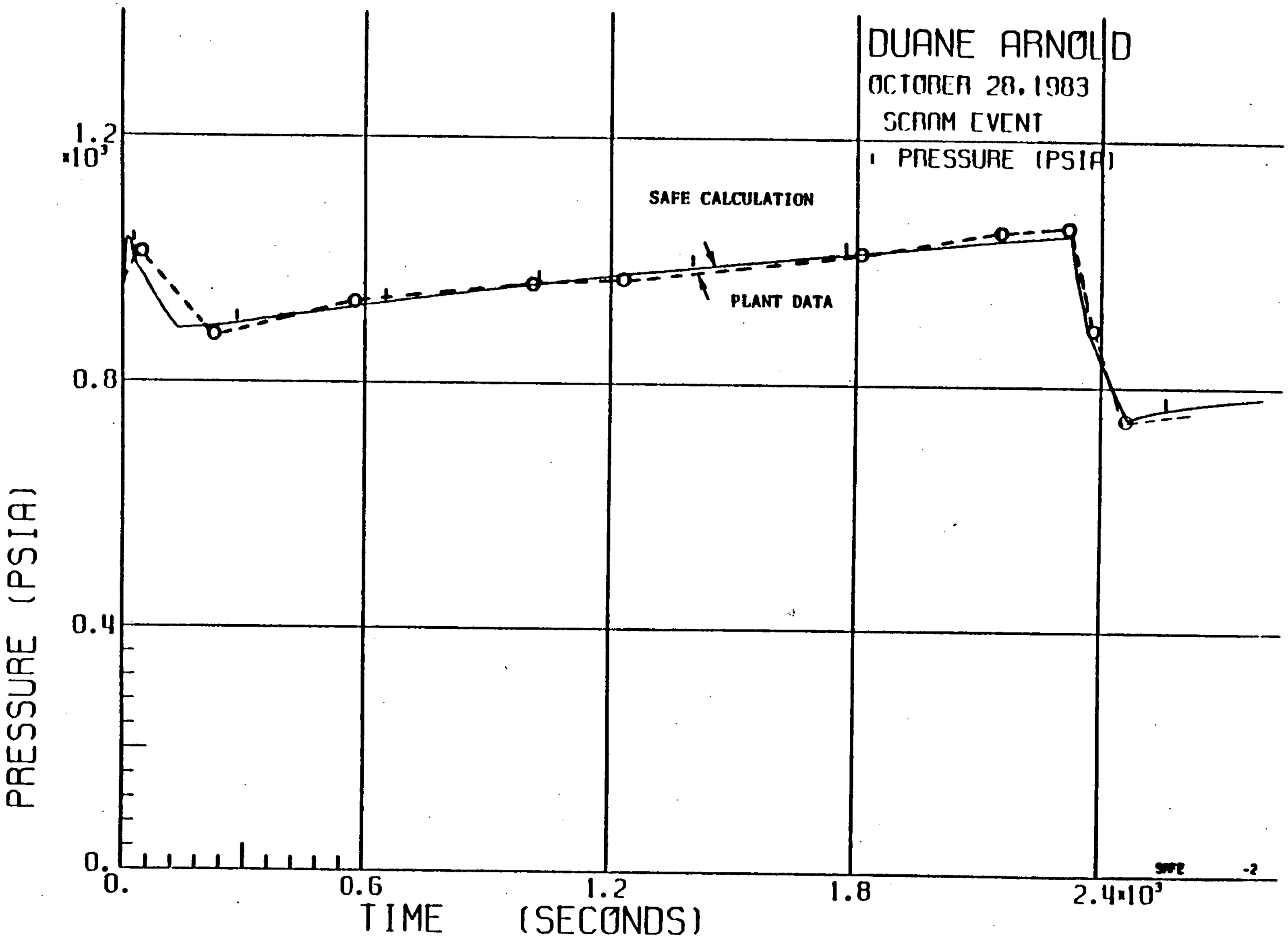
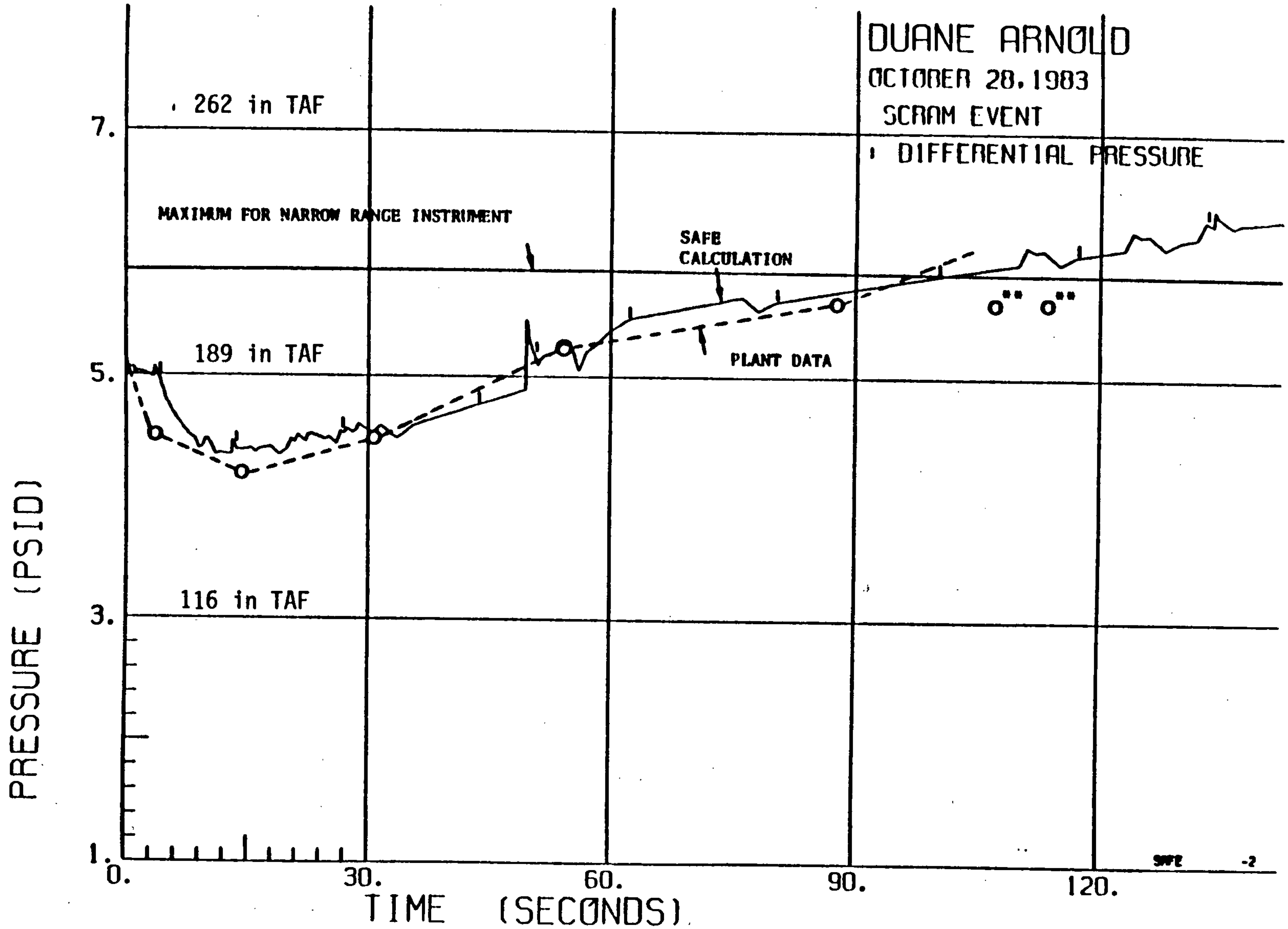
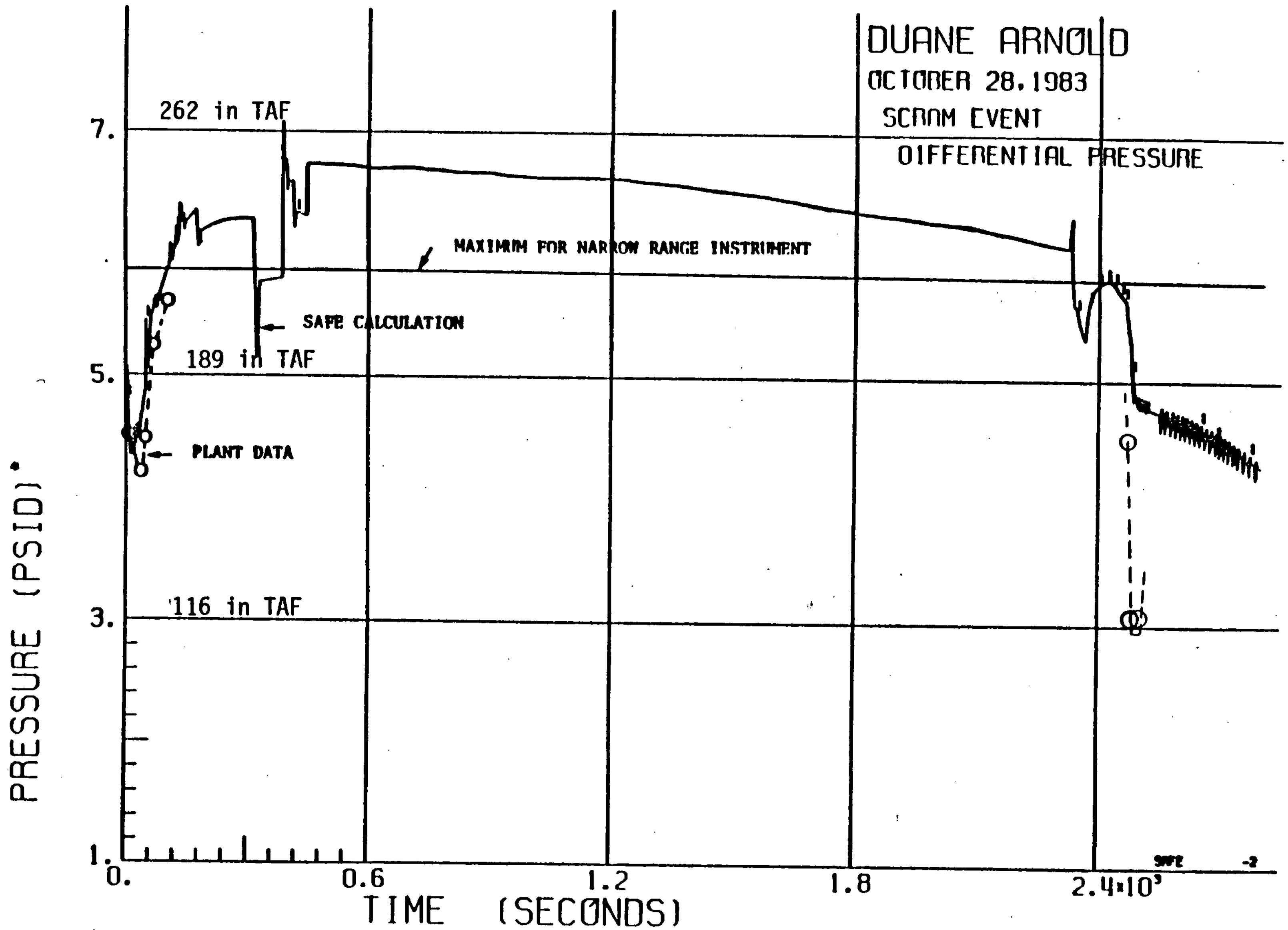


Figure 4-1 DAEC Reactor Pressure Response Comparison



** Other Level 8 indications

Figure 4-2 DAEC Indicated Water Level Response Comparison (First 3 minutes)



* 5 psid = 189 in TAF = 44.43 ft.

Figure 4-3 DAEC Indicated Water Level Response Comparison

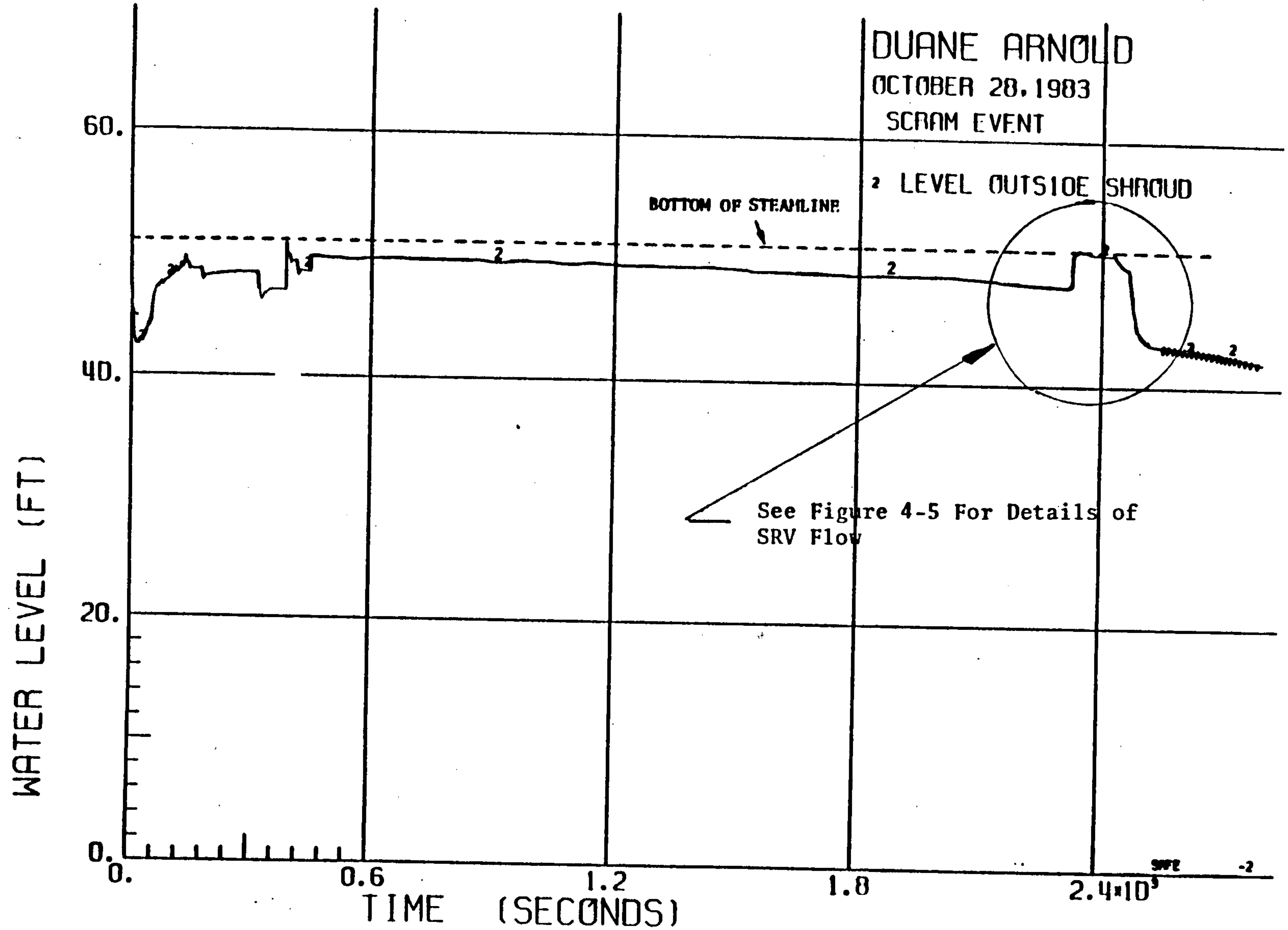


Figure 4-4 DAEC Predicted Water Level Response

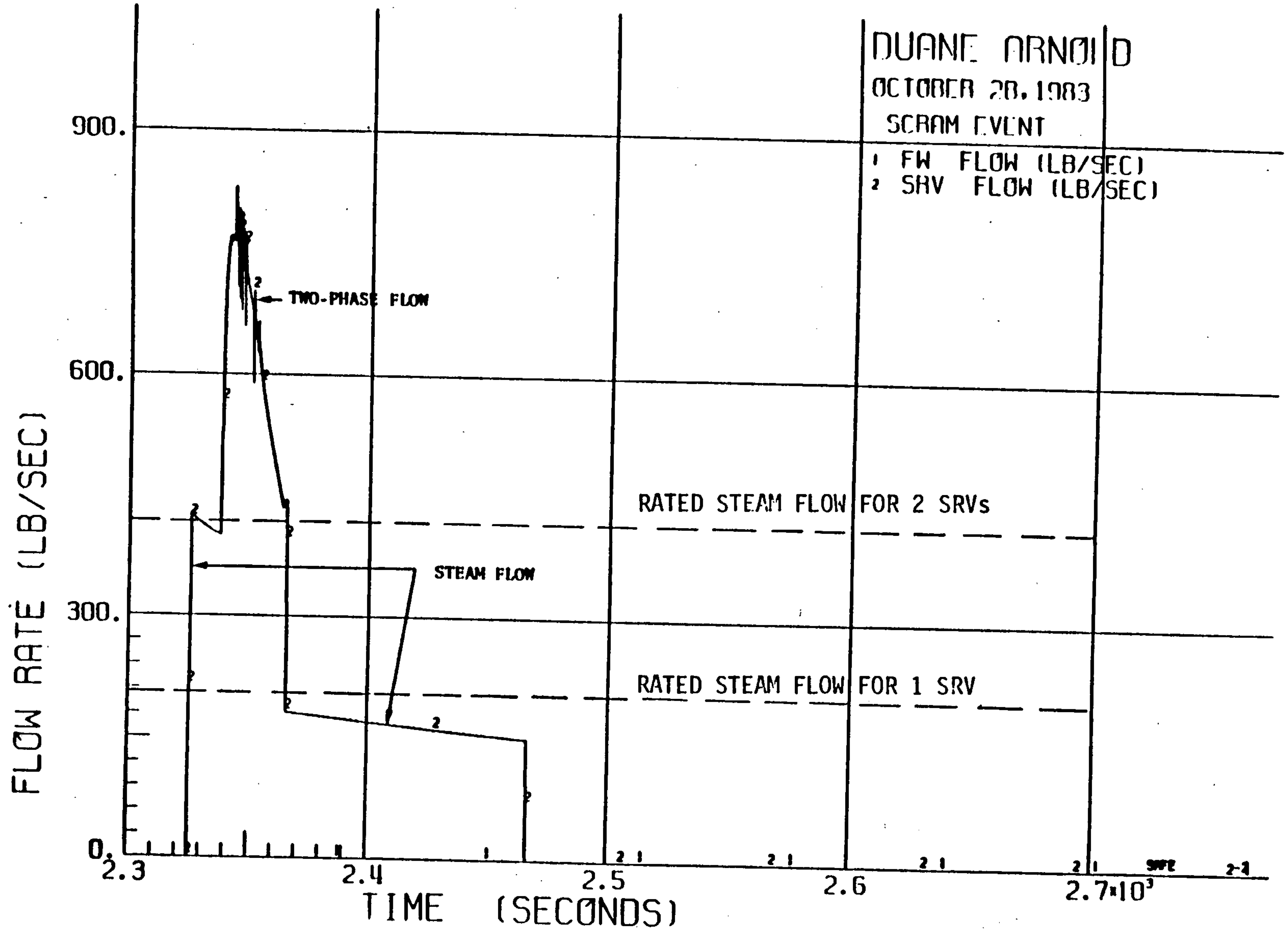


Figure 4-5 DAEC Total SRV Flow Rate

5.0 REFERENCES

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2. "Duane Arnold Energy Center Docket No. 50-331, Plant License No. DPR-49. NRC Request for Additional Information Relative to 10/28/83 MSIV Closure Event", Letter, R.W. McGaughy to H. Denton (NRC), Iowa Electric Light and Power Company, NG-83-4077, November 21, 1983.

3. "NUREG-0578 Requirements 2.1.2 - Performance Testing of BWR and PWR Relief and Safety Valves", Letter D.B. Waters (BWR Owners Group) to R.H. Vollmer (NRC), September 17, 1980.

4. "Unsteady Piping Forces Caused by Hot Water Discharge from Suddenly Opened Safety/Relief Valves", F.J. Moody (GE), Nuclear Engineering and Design, Volume 72, Pages 213-224, 1982.