2
COVER SHEET

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Title: STUDY OF SAFETY RELIEF VALVE
OPERATION UNDER ATWS CON-
DITIONS, SUPPLEMENT

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I. INTRODUCTION

A search of literature and industrial data from mid-1975 was commenced by the Energy Technology Engineering Center (ETEC) in October 1978 to update the Nuclear Regulatory Commission (NRC) search made for literature and data prior to that date. The purpose of the ETEC search was to:

- 1) Evaluate the NRC procedure for determining safety relief valve operating characteristics under ATWS conditions (e.g., 3000 psia, 40°F subcooled water).
- 2) Understand methods of determining backpressure and its effect on valve capacity.
- 3) Understand methods of calculating valve reaction forces under dynamic flow conditions.
- 4) Determine with NRC the availability of test facilities which could be used to provide any requisite data not available in literature and industry.

The results of this search were reported in Reference 1 where it was concluded that:

- 1) The isentropic homogeneous equilibrium (IHE) model used by NRC tends to underpredict flow due to metastable (superheated liquid) flow conditions.
- 2) No safety relief valve test facility exists with the simultaneous temperature, pressure, flowrate, and duration capabilities stated by NRC.

After the report was published, NRC requested that the ETEC search be expanded to include safety relief valve flow data for steam-water in the supercritical region. Also, some subcooled flow data and some relief valve test facility capability information, which had been requested but was not available for the original analysis, were now available and should be included in a supplemental report.

This report is a supplement to Reference 1 and is prepared in the same format as the original report for better correlation. It is intended that the supplement can be appended to the original report or exist as a separate report. Only headings with additional data are used.

II. SUMMARY

Shortly after the ETEC report (Reference 1) was published NCR requested ETEC to expand the search to include literature and data on safety relief valve flow under steam-water supercritical fluid conditions. This information as well as subcooled water data and safety relief valve test facilities which had become available since the report and paper analysis was made were to be included in a supplemental report.

Five additional searches were made into the files available to ETEC. Abstracts were printed out and papers obtained for analysis.

An industry and related-organization search was made for data pertaining to the supercritical state. Utilities and design organizations known to have operated or designed supercritical boiler systems were contacted via telephone or by letter.

Papers and reports obtained from these searches were submitted to ETEC Systems Analysis Unit for evaluation.

Safety relief valve flow data was not found for any size valve flowing steam-water in the supercritical state. This was surprising since many facilities have been designed and operated with steam-water in the supercritical state for many years. Considerable literature and data was available with nozzles and other flow devices with various cryogenic fluids in the supercritical state. Experimenters were able to predict flows with fluids other than those tested by using the method of corresponding states with good accuracy and were confident that predictions could be made with steam-water as well.

The additional subcooled data which had been requested for the original report was analyzed and included in the data matrix.

The additional safety relief valve test facility capabilities information supplied by NRC is included in tabular form.

III. CONCLUSIONS

In addition to the eight conclusions presented in Reference 1, the following are drawn from the supplemental study:

- 1) No data is available on safety relief valves of any size with steam-water in the supercritical state.
- 2) Considerable experimentation has been conducted with cryogenic fluids in the supercritical states. Experimenters are confident that the cryogenic data can be used to design models to predict steam-water flows by using the method of corresponding states.
- 3) The additional subcooled data substantiates the tendency of the IHE model to underpredict flow.
- 4) The additional safety relief valve facility capability data substantiates the conclusion that a single facility does not exist with the current capability of simultaneously testing a safety relief valve at the temperature (650°F), pressure (3200 psi), flowrate (575 lb/sec), and duration (equivalent to 8500 gallons) required for the ATWS simulation.

IV. RECOMMENDATIONS

In addition to the original recommendation to further search for safety relief valve facilities, the following recommendations are made:

- 1) NRC pursue with Lewis Research Laboratory (NASA) the development of the NASA working computer program for computing two-phase choked flow of water using a very fine thermophysical property program.
- 2) The Marviken data, referred to but not given by Kroeger (Reference 8), be obtained and evaluated.

V. LITERATURE AND DATA SEARCH

A. COMPUTER SEARCH OF INDEXES

Five additional searches listed in Appendix A were made for literature and data on supercritical fluid flow and 475 abstracts printed out for evaluation.

B. LETTERS AND TELEPHONE CALLS TO INDUSTRIAL AND RELATED ORGANIZATIONS

ETEC made telephone contacts with utilities known to have or currently operating supercritical boilers. ETEC also contacted by telephone design outfits designing supercritical systems and laboratories conducting supercritical fluid flow experiments. NRC sent inquiry letters to supercritical component manufacturers. The contacts and responses are listed in detail in Appendix C.

VI. ANALYSIS OF REPORTS AND PAPERS

A. GENERAL

The search was expanded to include literature or data relating to flow of fluids in the supercritical state. Additional literature or data on subcooled water which became available after the original analysis was completed is included.

B DISCUSSION

1. Flow Capacity and Pressure
- b. Report and Paper Analyses

The additional reports and papers analyzed are:

Hendricks, Simoneau, and Barrows (Reference 2) gives the results of two-phase flow experiments with subcooled liquid oxygen and nitrogen in converging-diverging nozzles. The experiments were undertaken to demonstrate that the corresponding-states normalization ranged from 0.61 to 1.67 times the thermodynamic critical temperature. Pressures ranged from near saturation to twice the thermodynamic critical pressure. From these data it was concluded that the two-phase choked flowrates and pressure ratios of subcooled oxygen and nitrogen can be normalized in a corresponding-states manner.

No mention of a steam or water application was made in this report. Also, no metastable flow was observed except for a few data points which did not follow the data trend.

Custer and Buxton (Reference 3) present test results for saturated water up to 124.4 psig. These data are not in the range required for the ATWS conditions. The actual measured flowrates of two 0.500 in. relief

valves (Crosby and Dresser) were much larger than anticipated by theory. This suggests that metastable flow conditions existed and that the relief valves acted like an orifice instead of a high L/D nozzle or tube.

Hendricks and Simoneau (Reference 4) present data for very large L/D tubes which are not typical of the geometry of a relief valve. Relief valves generally fit the classification of an orifice or small L/D nozzle where non-equilibrium, two-phase flow conditions exist for saturated and subcooled liquids. Thus, this report for cryogenic nitrogen is not applicable to nuclear reactor relief valves under ATWS conditions.

Allen (Reference 5) presents data and analyses which are not applicable to relief valves for pressurized nuclear reactors under ATWS conditions. The data herein is for saturated mixtures of water and steam at pressures less than 150 psig.

Gilbert and Williams (Reference 6) present an analytical paper concerned with the thrust force of a ruptured pipe and contains no new experimental data of the type applicable to safety relief valves for nuclear reactors under ATWS conditions.

Kroeger (Reference 8) gives an analytical paper which compares recent tests at the Marviken Test Reactor in Sweden to the HEM model and the Henry-Fauske thermal non-equilibrium model. Subcooling and pressures are not given in this report; however, the referenced data (Marviken) should be obtained and evaluated. This reference is "The Marviken Full-Scale Critical Flow Tests Interim Report," Reports No. MXC-201 to 203, Joint Reactor Safety Experiments in the Marviken Power Station, Marviken AB Atomenergi, Sweden (1978).

Lee (Reference 9) presents an analytical model which contains no new experimental data of the type applicable to safety relief valves for nuclear reactors under ATWS conditions.

Simoneau, Henry, Hendricks, and Watterson (Reference 10) reports experimental data for the two-phase critical flow of saturated and subcooled liquid nitrogen through a convergent-divergent nozzle. The nozzle consisted of an axisymmetric venturi flowmeter with a diameter of 0.293 in. The nozzle inlet conditions ranged from highly subcooled liquid ($P_0 = 42$ atm., $T_0 = 90K$) to conditions very near the thermodynamic critical point. Stagnation pressures and temperatures compared to the thermodynamic critical point ranged from 0.33 to 1.25 and 0.70 to 1.00, respectively. Test results showed that liquid nitrogen exhibits the same general nozzle behavior as water. No anomaly was observed for flows at near-critical stagnation conditions compared to those for saturated nitrogen away from the critical point. It is noted in this report that this data should not be equated to the discharge of saturated and subcooled liquids through sharp edged orifices. No metastable flow conditions were noted for these tests with nitrogen, probably due to the L/D of the test nozzle. The application of this test data to a relief valve with high pressure (near critical) liquid (saturated and subcooled) is doubtful.

Simoneau (Reference 11) in a paper presents two-phase choked flow data for nitrogen in the super-critical region; however, the narrow slit has a very large L/D and cannot be expected to represent a relief valve. These tests were intended to predict the two-phase leak rate and flow distribution of pressurized cryogenics discharging into space through a long, narrow, rotating seal passage. No application can be found for this data to predict safety valve operation for nuclear reactors under ATWS conditions.

Simoneau (Reference 12) in a report provides data for maximum two-phase flowrates of subcooled cryogenic nitrogen through a sharp-edged orifice. The initial stagnation conditions were always single-phase and the pressure drop on discharge was to a level substantially below saturation pressure. The data were taken along five stagnation isotherms ranging from 0.75 to 1.03 times the thermodynamic critical temperature. Stagnation pressures ranged from near saturation to twice the thermodynamic critical pressure. The experiment yielded the following conclusions:

- 1) Subcooled cryogenic liquids do not choke in a sharp-edged orifice even though the backpressure is well below saturation and two-phase flow clearly exists. It appears that under these test conditions, a metastable liquid jet is maintained through the orifice.
- 2) Flow coefficients for two-phase nitrogen maximum flowrates are not constant because of metastable flow conditions.

This data may be useful qualitatively to predict the flow phenomenon during safety relief valve operation under ATWS conditions, but no direct applicable use can be made because of the differing mediums (water vs nitrogen).

Walters and Boyle (Reference 13) present an analytical model which uses data from other sources (Faletti, Moy, Cruz, Fauske, Klingebiel) to predict low pressure (500 psig), high quality water-steam flow in a duct. No subcooled, super-critical, or low quality applications were covered. This report has no information not previously determined for safety relief valve operation under ATWS conditions.

Fabic (Reference 14) presents no test data or models or the results thereof. This report does not provide useful information for safety relief valve flow under ATWS conditions.

Schrock (Reference 15) gives an analytical discussion of the two-phase critical flow phenomenon relative to the LOCA in a LWR. No equations, test data, or theory is provided. This report is not useful for the safety relief valve flowrate determination under ATWS conditions.

Hendricks, Simoneau, and Ehlers (Reference 16) in a report present choked flow data for liquid nitrogen and compare this data with the equilibrium and non-equilibrium flow theory. The test nozzle was an axisymmetric venturi flowmeter with a somewhat higher L/D than for a safety relief valve nozzle. For this data to be applicable, the corresponding state theory would have to be accepted for nitrogen and water-steam and the nozzle geometry

would somehow have to be rationalized as being similar to a relief valve nozzle.

For nitrogen, the following conclusions were made:

- 1) Proximity of the fluid at the nozzle entrance to the critical point causes a significant influence on the critical flowrate. The flowrate exhibits an inflection and the pressure ratio a sharp peak when stagnation conditions are near the transposed critical point.
- 2) At stagnation temperatures above the thermodynamic critical temperature, the critical flowrates can be predicted quite satisfactorily by an isentropic equilibrium expansion. For subcooled and saturated liquids non-equilibrium effects appear significant.
- 3) Some inconsistent data was found in the subcooled regions.

Hendricks and Simoneau (Reference 17) report on saturated and subcooled nitrogen flow through a 55 L/D tube with a Borda type inlet. This geometry is very different from that of a safety relief valve; hence, the data and conclusions do not apply.

Simoneau (Reference 18) reports on tests where choked flowrates and axial pressure distributions were measured for subcooled nitrogen in a converging-diverging nozzle with a constant area section in the throat. Stagnation pressures ranged from slightly above saturation to twice the thermodynamic critical pressure. Stagnation temperatures ranged from 0.75 to 1.03 times the thermodynamic critical temperature. The test section was a conical, 7° half-angle convergence and a 3.5° half-angle divergence nozzle with a constant area throat with an L/D of 3.2. These data showed that nitrogen above the thermodynamic critical temperature follows the isentropic homogeneous equilibrium flow model but in other areas the data was anomalous. The author states that these data are only directly applicable to nitrogen over the stated range and to other simple cryogens.

The test fluid (nitrogen) and the nozzle geometry of this report are not applicable to steam-water flow through a safety relief valve nozzle under ATWS conditions.

Choe and Weisman (Reference 19) present an analytical study of the relationship between flow patterns and frictional pressure during cocurrent gas-liquid flow in horizontal pipes. No critical flow of water-steam mixtures through orifices or short nozzles is mentioned. This report is not applicable to nuclear relief valves under ATWS conditions.

Simoneau and Hendricks (Reference 20) in a report use data for choked flow of cryogenic liquids from other NASA Lewis work to produce generalized charts for the computation of two-phase choked flow of simple cryogenic liquids. These data support a recommendation for a use of a non-equilibrium theory of Henry and Fauske in the lower temperature regions and homogeneous isentropic equilibrium theory near the critical point and beyond. The charts presented herein are recommended only for cryogenic fluids of simple molecular structure.

This report offers no new experimental data and no data or application for the study of nuclear reactor relief valve operation under ATWS conditions.

Bankoff (Reference 21) presents an analytical report with no data. It involves the mixing of cold water and steam and local condensation rates. No information for the study of safety relief valve operation under ATWS conditions is provided by this reference.

Powell (Reference 22) presents data from tests which were conducted specifically to determine the flow characteristics of safety relief valves for pressurized water reactors. The nozzles tested were of a low L/D type similar to relief valve nozzles. A large series of tests were conducted at temperatures from 400 to 600°F and pressures from 500 to 2500 psia. The test results were evaluated with respect to the theoretical isentropic flow process to provide coefficients of discharge over the range of inlet temperatures and pressures noted. At an inlet pressure of 2500 psia, these coefficients

of discharge (actual flow divided by isentropic flow) varied from slightly less to slightly greater than unity as the inlet temperature was increased from 400 to 600°F (see Figure 1). It was also found that a decrease in inlet pressure (at constant temperature) caused the coefficients of discharge to become increasingly greater than unity; thereby leading to the conclusion that metastable flow was occurring in the nozzle throat and that the degree of metastable flow will be increased by decreasing the inlet subcooling.

The data and conclusions of this report appear directly applicable to the study of safety relief valve operation under ATWS conditions. The only area that this data lacks is the thermodynamic supercritical region. This may not be required since the attached figure shows that above 2500 psia the flow can be expected to be close to that predicted by the isentropic flow process. Cryogenic liquid flow tests in the supercritical region also support this trend.

Ardon and Furness (Reference 23) present an analytical study which compares test data of others with various equilibrium theories. As such, it does not provide new information or data regarding safety relief valve operation under ATWS conditions.

Hammitt, Mikieliewiez, Ernst, Krause, and Rockwell (Reference 24) describe experiments of a program for the study of liquid film stability and downstream liquid droplet formation and flow characteristics. Also included was the development and calibration of electrical conductivity film thickness gages. These experiments and studies are not useful for determining safety relief valve operation for nuclear reactors under ATWS.

Saha (Reference 25) presents an analytical study which includes no significant data not previously found in other literature. It does, however, further substantiate the tendency of the IHE model to underpredict flow.

Stolzenberg (Reference 26) summarizes the information on safety relief valve test facility capability information which became available since the original report was published.

(Back Pressure Less than Saturation Pressure)

$$C_D = \frac{\text{Measured Flow}}{\text{Isentropic Flow}}$$

Figure 1. Convergent Nozzle and Convergent-Divergent Nozzle Coefficient of Discharge vs Inlet Pressure for Various Inlet Temperatures (Figure 15 of Reference 22)

B.1 c. Data Matrix

The Powell data (Reference 22), which was not available to ETEC for the original report, is added to the data matrix as Table 1.

NOZZLE GEOMETRY SIZE AND LENGTH	PRESS. (PSIA)	TEMP. (°F)	QUALITY	SUBCOOLING (°F)	CRITICAL MASS FLUX (LBM / SEC - FT ²)	
					EXPERIMENTAL	0.9 HEM
CONVERGING NOZZLE 0.4375" DIA.	1000	397	-0.00020	148	18800	17500
	1500	↓	-0.00056	199	24200	23000
	2000	↓	-0.00120	239	27800	26500
	2500	↓	-0.00243	271	31700	30100
ROUNDED ENTRANCE WITH 0.4375 RADIUS.	1000	446	-0.00018	99	16800	15300
	1500	↓	-0.00062	150	21700	20700
	2000	↓	-0.00144	190	26000	24800
	2500	↓	-0.00294	222	29800	28500
AND EXIT CHAMBERS.	1000	497	-0.00015	48	15000	11200
	1500	↓	-0.00068	99	20100	17700
	2000	↓	-0.00171	139	24200	22300
	2500	↓	-0.00369	171	27700	26200
	1000	550	-0.00042	27	15700	8800
	1500	↓	-0.00075	46	17700	12100
	2000	↓	-0.00186	86	22100	18400
	2500	↓	-0.00488	118	25700	22600
	1000	592	-0.00090	13	14600	7200
	1500	↓	-0.00125	29	17300	11000
	2000	↓	-0.00201	44	18500	14000
	2500	↓	-0.00606	76	22600	19200

Table 1. Data Matrix

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VII. FACILITY SURVEY

Considerable additional information pertaining to safety relief valve test facility capabilities have become available since the original report was published. These are summarized by Stolzenberg (NRC) and added to this report as Table 2. Data for the ETEC facility were expanded by the authors of this report.

The temperatures, pressures, volumes, and flowrates for the various facilities may be compared with the following values provided by NRC as desired capabilities:

Pressure:	2400 to over 3000 psia
Temperature:	approximately 40°F subcooling (550 to 600°F)
Flowrate:	2.07×10^6 lb/hr
Volume:	approximately 1100 ft ³ of subcooled water

Table 2. Summary of Available High Pressure Steam and Water Facilities
for Valve Testing
(Sheet 1 of 2)

FACILITY	LOCATION	MEDIUM	PRESS.	TEMP.	VOLUME	RATE	UPGRADE POSSIBILITIES	ADDITIONAL CAPABILITIES AND REMARKS
WPEC Nuclear Power Engineering Center	Isogo, Japan	Steam	(175 Bar) 2537 psi	(343°C) 649°F	(20m ³) ₃ 714 ft ³	434t/hr 868,000#/hr	880 Bar 1100 psi	1500t/hr pump capy for testing coeff of hyd loss Accident environment simulation. Steam test for modifing of valves.
KWU	Erlangen, FRG	Steam and/or Water	(175 Bar) 2537 psi	Sat	(1060 Liters)			Testing of pilot valve of press relief valve has been done with steam sat. water and subcooled water. There will be test transition from steam to water
	Karlstien, FRG	Steam and/or Water	(150 Bar) 2175 psi	Sat	(20m ³) ₃ 770 ft ³			Test will be run on pilot and main valve, but main valve will have reduced size nozzle because of flow limits.
HDR	Kahl, FRG		(110 Bar) 1595 psi	(300°C) 572°F				
Wyle Laboratory	Huntsville, AL	Hot Water 2 Phase	2700 psi	540°F		2000 GPM		
		Hot Water 2 Phase	1500 psi	540°F		50 GPM		The vessel is rated @ 2852 PSIG @ 600°F-Higher Pressure and Flows could be obtained with piping & boiler modifications.
		Hot Water 2 Phase	2500 psi	600°F				
		Steam	1650 psi	609°F	1400 ft ³	1,200,000#/hr		
		Steam	1200 psi	567	900 ft ³	1,800,000#/hr		Same 2852 psi tank as above with pot upgrade in press and flow.
		Water	400 psi		6000 gal	8500 GPM		Heating coils would allow 600°F
	Norco, CA	Water	5500 psi	650°F	300 ft ³			These Nos. are based on mod to existing facility to upgrade press rating
		Steam	5500 psi	650°F	300 ft ³			

Table 2. Summary of Available High Pressure Steam and Water Facilities
for Valve Testing
(Sheet 2 of 2)

FACILITY	LOCATION	MEDIUM	PRESS.	TEMP.	VOLUME	RATE	UPGRADE POSSIBILITIES	ADDITIONAL CAPABILITIES AND REMARKS
	France	Water Steam	(120 Bar) 1740 psi (260 Bar) 3770	Sat & Sub-cooled Sat	(6m ³) 1500 gal (7m ³) 250 ft ³	(.2m ³ /sec) 50g/sec 30sec blowdown		Steam is used to push the water through 6" nominal size safety valve
EC	Canoga Park, CA	Steam Water	6000 psi	100°F 650°F	600ft ³	468 lb/sec	The Nos are based upon use of use Air Force Vessels to be installed at ETEC - Will be upgraded to 650°F at slightly reduced pressure	
CD nold Engineering velopment Center	HLA	Steam	760 psi	Sat	30,900 ft ³	5 x 10 ⁶ #/hr		Higher pressure Air (3800 psi) and Nitrogen (5000 psi) available

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APPENDIX A
COMPUTER AND MANUAL INDEX SEARCHES

1. Computer Searches

The computer index search was expanded to include supercritical flow. In the various indexes supercritical flow does not exist as a heading. It is listed under turbulent flow.

Search 1. Terminal LME, Session 0001, File ESR, 4/6/79

<u>Set</u>	<u>Descriptor</u>	<u>Cits</u>	<u>Abstract Printout</u>
1	RB = Michigan	46	-
2	IT = Turbulence	28	-
3	1 * 2		
4	IT = Two-phase flow	2	2
5	1 * 4		
6	IT = Nuclear power plants	43	-
7	LC = MI	42	-
8	1 + 7	54	-
9	IT = Safety	118	-
10	8 * 9	1	-
11	6 * 9	1	1

<u>Set</u>	<u>Descriptor</u>	<u>Cits</u>	<u>Abstract</u> <u>Printout</u>
1	E6-E40	104	-
Expand	MD = Turbulent flow		
2	E7-E12	7	-
Expand	MD = Turbulent Flow/Nusselt Number		
3	1 + 2	109	-
4	LA = Foreign	81962	-
5	3-4	73	73
6	IT = Water	13638	-
7	IT = Fluid Flow	11836	-
8	6*7	717	-
9	3*8	8	-
10	E2-E10	13	-
Expand	MD = Relief Valves		
11	IT = Relief Valves	50	-
12	IT = Valves	1938	-
13	IT = Reactor Cooling Systems	3275	-
14	E5-E8	6	-
Expand	MD = Pressure Gradients	928	-
15	IT = Pressure Gradients	50	-
16	10+11	50	50
17	16*12	65	65
18	13*15		
19	17*18		

Search 3. Terminal LME, Session 0006, File EDB, 4/27/79

<u>Set</u>	<u>Descriptor</u>	<u>Cits</u>	<u>Abstract</u> <u>Printout</u>
1	IT = Critical Flow	116	-
2	IT = Turbulent Flow	930	-
3	It Steam	3925	-
4	MD Steam/Flow Models	1	-
5	LA = Foreign	81962	-
6	1 - 5	80	80
7	2*3	7	7
8	AU = Barrowes, S.C.	2	-
9	IT Peach Bottom-1 Reactor	68	-
10	9*3		
11	9*1		

Search 4. Terminal LME, Session 0007, File NSA, 4/27/79

<u>Set</u>	<u>Descriptor</u>	<u>Cits</u>	<u>Abstract</u> <u>Printout</u>
1	AU = Simoneau, R.J.	1	-
2	AU = Barrows, R.G.	3	-
3	It = Critical Flow	157	157
4	IT = Turbulent Flows	386	-
5	IT = Steam	2583	-
6	4*5	5	5

Search 5. Terminal LME, Session 0017, File EDB, 4/27/79

<u>Set</u>	<u>Descriptor</u>	<u>Cits</u>	<u>Abstract</u> <u>Printout</u>
1	IT = Fluid Flow	11836	-
2	IT = Flowrate	2034	-
3	IT = Steam	3925	-
4	1-3/*	49	-
5	LA = Foreign	81962	-
6	4-5	35	35
7	2*3	114	-
8	IT = Turbines	4818	-
9	7*8	9	-

APPENDIX B

LETTER AND TELEPHONE SEARCH TO INDUSTRIAL AND RELATED ORGANIZATIONS

The following telephone or letter contacts were made:

1. Combustion Engineering Co., Windsor, CT, Ms. June Cicerchia, letter from Gary L. Bennett, NRC.
2. General Electric Company, BWR Products Standards, San Jose, Ca., Dr. W.H. D'Ardenna, Manager, letter from Gary L. Bennett, NRC.
3. Westinghouse Electric Corporation, letter from Gary L. Bennett, NRC.
4. Philadelphia Electric, Philadelphia, Pa., telephone call Hutmacher to N. Gaffin of the Mechanical Engineering Department.
5. LA Dept. of Water and Power, Los Angeles Ca., telephone call Hutmacher to S. Craus.
6. Townsend and Bottom, Detroit Michigan, telephone call Hutmacher to G. Lamb of the Mechanical Engineering Department.
7. Lewis Research Center, Cleveland, Ohio, telephone call Hutmacher to R.J. Simoneau.
8. Southern California Edison, Los Angeles, Ca., telephone call Hutmacher to M. Newhouse of the Mechanical Engineering Department.

Response to Letters or Telephone Calls

Philadelphia Electric (N. Gaffin) - He volunteered the following information:

Philadelphia Electric no longer has supercritical boilers.

He did not know of any literature or data on components flowing under supercritical conditions.

Users sized relief valves for single phase choked flow at the fluid condition of interest to them.

He did not know of any facility capable of testing relief valves of any size at supercritical fluid conditions.

He had sized systems for sonic velocity at pressures to 5900 psig and temperatures to 1230°F. He used the T-S diagram to determine the fluid properties.

No detrimental seat damage was experienced for relief valves popping at 4200 psig and 1050°F.

LA Department of Water and Power (S. Craus) - His observations were:

He does not know of any data, published or unpublished, concerning water fluid flow in the supercritical state for any size valve or flow device.

Systems are designed well above the normal operating pressure so the safety-relief valves can also be set high which avoids the normal reseating and wire drawing problems with safety-relief valves.

They do not size safety relief valves but depend on the A&E contractor to properly design the system. If a reorder of a component were necessary, an identical component would be specified.

Townsend and Bottom (G. Lamb) - The following information was given:

He did not know of any data or literature specifically dealing with water fluid flow in the supercritical state.

He had worked for Combustion Engineering Co. and thought some personnel there could have something of value. He said he would contact them and call back if he found anything.

He indicated that data above 2-1/2 in. pipe size would not be available as the Code limits openings on supercritical boilers to that size.

He suggested contact with several safety relief valve companies we had already contacted.

In a subsequent call, G. Lamb indicated he had some material from Crosby Valve Company on sizing of their relief valves for supercritical conditions. The sizing information included pressures from 3100 to 5000 psig and nozzles from 7/8 to 3.96 in. He forwarded this information for evaluation.

Southern California Edison (M. Newhouse) - The following information was given:

Mr. Newhouse indicated that he did not have or know of any literature or data on flow of steam-water in the supercritical state; however, he suggested talking to Mr. G. Fraser of their Maintenance Department at phone number 572-1637.

Mr. Fraser was not in the office and Mr. D. Schroeder discussed the matter. Mr. Schroeder indicated that they were in maintenance and did not have access to literature or data in the flow regime. He said he would talk to Mr. Fraser and that it would be best to call back. No additional information was obtained in followup calls.

Lewis Research Center (R. J. Simoneau) - The following information was given:

He does not know of any work being or have been done in this flow regime with steam-water fluid.

He has done considerable work in all types of flow geometries with liquid nitrogen in the supercritical state.

He feels confident that the liquid nitrogen work would predict supercritical steam-water single phase flow using the homogeneous equilibrium model. The prediction would be within instrument error (2 to 3%).

He has a program that could predict supercritical steam-water flows: however, he has not had reason and has not run this program for steam-water.

He thought he could provide this information, given about two weeks funded computer time which he would only do if requested from agency to agency (NRC to NASA).

He offered to send an additional six of his reports (two of his reports were available from the literature search) and several more coauthored. These were received as enclosures to a letter.

Westinghouse, References 3, 5, and 22 were supplied to NRC.

Combustion Engineering, Reference 25 was recommended as a source of information.

APPENDIX C
LIST OF REPORTS AND PAPERS SELECTED FOR EVALUATION

Additional reports and papers are:

- 1) Allen, W.F. Jr., "Flow of a Flashing Mixture of Water and Steam through Pipes and Valves," ASME Transactions, 1979 Annual Meeting, page 257-265.
- 2) Ardon, K.H. and Furness, R.A., "A Study of the Critical Flow Models Used in Reactor Blowdown Analysis," Nuclear Engineering and Design, February 1976, page 257-265.
- 3) Bankoff, S.G., "Interfacial Mass and Momentum Transfer in Two-Phase Flow," ANS Transactions, 1979 Annual Meeting, page 811.
- 4) Fabric, S., "Analytical Modeling of Transient Two-Phase Flow," ANS Transactions, 1979, Annual Meeting, page 806.
- 5) Gilbert, J.S., and Williams, K.A., "TRAC-PIA Calculations of Edwards Blowdown Experiment," ANS Transactions, 1979 Annual Meeting, pages 439-440.
- 6) Hammitt, G.G., et al., "Steam Tunnel Program and Liquid Film Thickness Gage Developments at University of Michigan, UMICH 012449-6-T, January 1975, 29 pages.
- 7) Hendricks, R.C., "Normalizing Parameters for the Critical Flow Rate of Simple Fluids Through Nozzles," Lewis Research Center, NASA TMX-71545, May 1974, 11 pages.
- 8) Hendricks, R.C., "Critical Mass Flux Through Short Borda Type Inlets of Various Cross Sections," paper to be presented as BI-77 at the International Congress of Refrigeration, September 1979, 5 pages.

- 9) Hendricks, R.C., "Some Aspects of a Free Jet Phenomena to 105 L/D in a Constant Area Duct," paper to be presented as BI-78 at the International Congress of Refrigeration, September 1979, 9 pages.
- 10) Hendricks, R.C. and Simoneau, R.J., "Application of the Principle of Corresponding States to Two-Phase Choked Flow," Lewis Research Center, NASA TMX-68193, 17 pages.
- 11) Hendricks, R.C. and Simoneau, R.J., "Some Flow Phenomena in a Constant Area Duct with a Borda Type Inlet Including the Critical Region," ASME Paper 78-WA/HT-37, August 1978, 7 pages.
- 12) Hendricks, R.C., and Simoneau, R.J., "Two-Phase Choke Flow in Tubes with Very Large L/D," Lewis Research Center, NASA TM 73726, August 1977, 21 pages.
- 13) Hendricks, R.C., Simoneau, R.J., and Barrows, R.F., "Two-Phase Choked Flow of Subcooled Oxygen and Nitrogen," Lewis Research Center, NASA TN D-8169, February 1976, 40 pages.
- 14) Hendricks, R.C., Simoneau, R.J., and Ehlers, R.C., "Choked Flow of Fluid Nitrogen with Emphasis on the Thermodynamic Critical Region," Lewis Research Center, NASA TMX-68107, August 1972, 27 pages.
- 15) Hendricks, R.C., Simoneau, R.J., and Hsu, Y.Y., "A Visual Study of Radial Inward Choked Flow of Liquid Nitrogen," Advances in Cryogenic Engineering, Vol. 20, Plenum Press, 1975, pages 370-382.
- 16) Hsu, M. and Redmond, J.W., "Subcooled Blowdown Thrust Force Calculations Using RELAP4/REPIPE," ANS Transactions, 1979 Annual Meeting, pages 437-438.
- 17) Kroeger, P.G., "Comparison of Quasi-Steady Blowdown Models with Recent Experiments," ANS Transactions, 1979 Annual Meeting, pages 432-433.

- 18) Lee, W.H., "A Critical Flow Mode for Air-Steam-Water Mixtures in Reactor Containment Systems," ANS Transactions, 1979 Annual Meeting, pages 428-429.
- 19) Saha, Pridip, "A Review of Two-Phase Steam-Water Critical Flow Models with Emphasis on Thermal Equilibrium," Department of Nuclear Energy, Brookhaven National Laboratory, NUREG/CR-0417, September 1978, 59 pages.
- 20) Schrock, V.E., "Two-Phase Critical Flow Phenomenon," ANS Transactions, 1979 Annual Meeting, pages 806-807.
- 21) Simoneau, R.J., "Pressure Distribution in a Converging-Diverging Nozzle During Two-Phase Choked Flow of Subcooled Nitrogen," Lewis Research Center, NASA TMX-71762, December 1975, 11 pages.
- 22) Simoneau, R.J., "Two-Phase Choked Flow of Subcooled Nitrogen Through A slit," Lewis Research Center, NASA TMX-71516, April 1974, 14 pages.
- 23) Simoneau, R.J., "Maximum Two-Phase Flow Rates of Subcooled Nitrogen Through a Sharp-Edged Orifice, Lewis Research Center, NASA TMX-71760, July 1975. 22 pages.
- 24) Simoneau, R.J. and Hendricks, R.C., "Generalized Charts for Computation of Two-Phase Choked Flow of Simple Cryogenic Liquids, Cryogenics, February 1977, pages 73-76.
- 25) Simoneau, R.J., et al., "Two-Phase Critical Discharge of High Pressure Liquid Nitrogen," Lewis Research Center, NASA TMX-67863, September 1971, 9 pages.
- 26) Stolzenberg, M.D., "Safety and Relief Valves," NRC Memo to Gary L. Bennett, May 1979, 49 pages including 43 pages of enclosures.

- 27) Walters, W.P. and Boyle, J., "The Flow of a Liquid Vapor Mixture Through a Duct," USA Ballistic Research Laboratory, BRL-MR-2779, August 1977, 37 pages.