Steve Scott for PDR

~~Bennett~~

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

MAY 18 1979

MEMORANDUM FOR: Gary L. Bennett, Chief
Research Support Branch

FROM: M. D. Stolzenberg
Research Support Branch

SUBJECT: SAFETY AND RELIEF VALVES

At the request of D. F. Ross, NRR, this memorandum and enclosures, summarize information available regarding relief and safety valve flow discharge (water and two-phase), operation, and testing. This memorandum completes EDO-TMI Action #18, as amended by the May 7, 1979 telephone request from D. F. Ross to you.

RELIEF/SAFETY VALVE FLOW

To support NRR's evaluation[1] of the flow through the safety valves on the B&W and GE plants under ATWS conditions RES contracted with ETEC (Energy Technology Engineering Center) to conduct a literature search to determine if there were any additional data generated in the interim between the 1975 report on ATWS[2] and 1978.

The final report, Study of Safety Relief Valve Operation Under ATWS Conditions (NUREG/CR-068) was completed in January 1979 and published in March 1979, thereby completing the first phase of the research requested in Reference 1.

1 Memorandum from E. G. Case to S. Levine, Subject: Request for Confirmatory Research Related to the Behavior of the Pressurizer Safety/Relief Valves During Subcooled Discharges (RR-NRR-78-10), dated May 10, 1978.

2 Anon., "Status on Anticipated Transients Without Scram for Westinghouse Reactors," Appendix D, Division of Systems Safety, U.S. Nuclear Regulatory Commission, December 9, 1975.

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In summary, the ETEC report states:

"Flow capacity data was not found in the literature or industry survey for safety relief valves of any size operating under ATWS conditions. Considerable data was found on nozzles and other devices at pressures to 2500 psia with various degrees of subcooling. This data included in Table 1 (attached to this memo)... illustrates the tendency of the IHE model* to underpredict (flow is greater than predicted) flow. The underprediction is due to metastable (superheated liquid) flow conditions. Certain conditions may, however, cause the IHE model to over-predict: dissolved gases which may prevent metastable flow conditions, subcooling in excess of 100°F where discontinuity of the sonic velocity occurs at the saturated liquid line, and the nozzle geometry where increased diameter reduces the critical mass flux."

This conclusion is further substantiated in a separate BNL report[3] where it is stated for pipe breakflow: "... both the homogeneous-equilibrium model (HEM) and the Moody model underpredict the critical flow rate data considerably."

ETEC is currently working on a supplemental report which will include consideration of supercritical behavior. Contact has been made with staff members at Westinghouse, Combustion Engineering and General Electric. Westinghouse supplied the following documents which have been informally transmitted to NRR:

1. "Flow of a Flashing Mixture of Water and Steam Through Pipes and Valves," by W. F. Allen, Jr., Transactions of the ASME (April, 1951).

2. "Flow of Subcooled Water Through Nozzles," by A. W. Powell, WAPD-PT(V)-90 (April 12, 1961).

3. "Capacity Rating of Nozzle Type Safety Valves on Saturated Hot Water," by F. A. Custer and O. E. Buxton (January 2, 1971).

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* IHE = Isentropic homogeneous equilibrium model (ASME Code, Section III).

3 Pradip Saha, A Review of Two-Phase Steam-Water Critical Flow Models with Emphasis on Thermal Nonequilibrium, USNRC Report NUREG/CR-0417 (September 1978).

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It is my understanding that C-E has informed you that the ETEC report did not include Reference 3 (probably due to an overlap in publication) and a U.K. report "A Study of the Critical Flow Models Used in Reactor Blowdown Analysis." It is also my understanding that C-E may have a valve testing capability soon.

Contact has also been made with representatives of Philadelphia Electric which operates a fossil-fueled super-critical power plant (Eddystone). Unfortunately, the safety valves were not tested for flow rate in the plant.

Discussions with W. S. Farmer pointed out that Pratt and Whitney planned super-critical nozzle tests but these were never performed.

RELIEF/SAFETY VALVE FAILURE

A recent NRR report[4] cites over 100 examples of failure of pressure relief and safety relief valves, and indicates that "operating experience with spring loaded safety valves has been essentially failure free." The significance here is that the pressure relief and safety relief valves are of the pilot operated type while the safety valves are spring loaded and self-activated.

The Nuclear Safety Information Center will soon publish a bibliographic report on valve behavior.

FOREIGN RESEARCH

The following sections summarize programs in France, Germany (FRG) and Japan on relief and safety valves with comments.

Federal Republic of Germany

There is a research program on valves being conducted by KWU (Kraftwerk Union, Erlangen, FRG) called "Investigation on the Operational Reliability of the Pressurizer Safety Valves and Relief Valves During Blowdown of Hot Pressurized Water."

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4 U.S. Nuclear Regulatory Commission, Technical Report on Operating Experience with BWR Pressure Relief Systems, NUREG-0462 (July 1978).

The intent of this research program is to demonstrate experimentally and analytically, the capability of the relief valve to meet ATWS conditions. This research effort will involve tests with saturated steam, hot pressurized water and the transition from saturated steam to hot pressurized water.

Testing will take place in two facilities, one in Karlstein and one in Erlangen. The testing will be limited by pressure and flow capabilities of the facilities. These tests are to investigate:

1. performance of pilot and primary (main) valve,
2. interplay between pilot and primary valve,
3. effect of piping layout and operation.

The pilot valve has had some testing completed at Erlangen using saturated steam and hot pressurized water. These tests were conducted with the pilot valve connected to a primary valve but without any fluid connection to the primary valve. The valve performed as expected with saturated steam, but did not open as smoothly and completely with hot pressurized water. In addition the pressure gradient used (10 bars/sec) is higher than the required gradient (4 bars/sec). Additional tests will be run at the lower gradient.

The tests at Karlstein will include a blowdown through the primary valve but at about 1/10 the design flow rate.

The remaining tests are expected to be run in the latter part of 1979.

> Comment: These tests are similar to what may be required for an NRC ATWS safety valve test but they don't go far enough; that is, the pressure is considerably below what we would require (3200 psi). There has been no attempt so far in the FRG research program to measure flow rates. Moreover, the FRG researchers are testing pilot-operated valves while the valves of interest to us are of the self-activated spring type. However, the valve being tested appears to be very similar to the relief valve (RC-R-2) that did not reseat at TMI-2.
>
> Additional details on the FRG research will be sent shortly by W. B. Corwin, an ORNL staff member stationed in FRG. (See enclosed letter for more information on the current status). We can expect to have limited access to data because these tests appear to be privately funded and may not be covered by our research exchange agreement with the FRG Ministry of Research and Technology.

France

Researchers with the French EdF intend to test a 6-inch self-activated spring-loaded safety valve. The testing is to be conducted with hot pressurized water pressurized by steam in a facility near Paris.

The initial steady-state tests were conducted with the valve mechanically held open. The dynamic tests, with the valve opened by system pressure, are to be conducted from September-December 1979. (The French CEA said they sent NRC information on valve tests but the documents seem to have gotten lost within NRC).

> Comment: These tests could contribute considerably to the data we are looking for to confirm the method of analysis of the flow through a safety valve under ATWS conditions. The pressures being used are lower than our ATWS conditions and the flows expected may not match, but the general conditions, size and type of valve appear to be very close to what we are looking for. Since we have no exchange agreement with EdF we must await completion of EPRI negotiations with EdF.

Japan

The Japanese have a multi-purpose facility called the Nuclear Power Engineering Test Center for testing BWR and PWR nuclear plant components. The testing for valves is categorized as follows:

1. accident environment simulation tests,
2. operation and leakage tests,
3. safety and relief valve tests.

The program on valves which is called "Reliability Demonstration Tests for Nuclear Plant Valve" includes testing of safety and relief valves to confirm the functioning of the valves, service life, leakage and valve coefficients. Testing has begun and will continue through 1981, but the valves scheduled for the test all appear to be for BWRs.

> Comment: This program is very broad and covers many facets of safety and relief valves. With the exception of pressure the facility appears to have the capability to provide data which may be applicable to our ATWS problem. There is no indication, as yet, that they are planning that type of test. It is not clear that our exchange agreement with Japan covers this type of testing.

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Enclosed for your use is a summary table of the world-wide test facilities known to me. At present there are no U.S. facilities capable of performing ATWS-type tests; although both the ETEC and Wyle facilities could be modified for such testing. A summary of the ETEC capabilities is enclosed.

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Enclosures: 4, as stated

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Enclosure 1

Table 1. Data Matrix (Sheet 1 of 4)

NOZZLE GEOMETRY SIZE AND LENGTH	PRESSURE (PSIA)	TEMPERATURE (F)	QUALITY	SUBCOOLING (F)	CRITICAL MASS FLUX (lbm/Sec-Ft^2)	
					EXPERIMENTAL	.9 HEM
Convergent-Divergent Nozzle, Schrock No. 3	450	450	NA	6.3	7536	4748
Throat Diameter 0.396 Cm (.156 in)	700	450	NA	63.1	13941	5426
Exit Diameter 2.56 Cm	900	450	NA	82	16956	5765
Convergent 40 DEG-Half Angle	1050	550	NA	0.6	7536	5087
Divergent 12 DEG-Half Angle	1100	550	NA	6.3	11304	5426
	1200	550	NA	17.2	13188	5595
Orifice - Collins	64	297	NA	0	3081	616
(Yarnell Data)	192	378	NA	0	5049	1478
	321	424	NA	0	6675	2234
	144	350	NA	5	4279	1232
	192	350	NA	28	7274	4621
	64	300	NA	3	3081	616
	192	300	NA	78	8558	7317
	96	250	NA	75	6504	5777
	192	250	NA	128	8900	8627
Sozzi & Sutherland Nozzle 1	610	489	.00280	0	6200	3420
Inlet Diameter 1.75 in.	660	490	.00050	0	8420	3690
Well rounded inlet	680	486	-.00050	14	11500	5850
Throat diameter 0.5 in.	700	474	-.00100	29	12100	8640
Diffuser Section - 4.5 in.	725	507	.00380	0	6007	4140
long with a 3 deg. half angle	780	515	.00050	0	9210	4410
	825	522	.00060	0	9080	4590
	850	527	.00460	0	6050	4410
	890	449	.00380	82	16850	14850
	900	532	.00050	0	9780	4590
	955	539	.00680	0	6900	4500
	960	465	-.00410	75	15500	14400

NA - NOT AVAILABLE —

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Table 1. Data Matrix (Sheet 2 of 4)

NOZZLE GEOMETRY SIZE AND LENGTH	PRESSURE (PSIA)	TEMPERATURE (F)	QUALITY	SUBCOOLING (F)	CRITICAL MASS FLUX ($lbm/Sec\text{-}Ft^2$)	
					EXPERIMENTAL	.9 HEM
Sozzi & Sutherland Nozzle 2 Inlet Diameter 1.75. in. Well Rounded Inlet Throat Diameter 0.5 in. Throat Length						
0.0 in.	950	517	-.00130	20	15530	8550
0.5 in.	949	540	.00400	0	7750	5040
1.5 in.	955	540	.00340	0	6541	4950
2.5 in.	952	540	.00320	0	6230	4680
4.5 in.	948	540	.00310	0	5750	4500
7.5 in.	949	540	.00300	0	5488	4500
9.0 in.	950	539	.00050	0	6450	4680
12.5 in.	951	540	.00300	0	4875	4500
20.0 in.	952	522	-.00100	18	7320	5940
25.0 in.	951	540	.00250	0	4675	4590
70.0 in.	925	535	-.00006	< 1.0	3940	4500
Sozzi & Sutherland Nozzle 3 0.5 in. Diameter Sharp-Edged Orifice Length downstream of Orifice Plate:						
0.0 in.	950	540	.00450	0	8248	4770
7.5 in.	949	540	.00280	0	5050	4725
12.5 in.	952	522	-.00100	16	5850	6300
20.0 in.	951	495	-.00250	43	8400	9900
25.0 in.	951	505	-.00200	33	7950	9000

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Table 1. Data Matrix (Sheet 3 of 4)

NOZZLE GEOMETRY SIZE AND LENGTH	PRESSURE (PSIA)	TEMPERATURE (F)	QUALITY	SUBCOOLING (F)	CRITICAL MASS FLUX (lbm/Sec-Ft^2)	
					EXPERIMENTAL	.9 HEM
Sozzi & Sutherland Nozzle 4 Well-Rounded Inlet Throat Diameter .75 in. Abrupt Expansion Length 0.0 in.	954	534	-.00030	6	10500	5310
Nozzle 5 Convergent-Divergent Throat Diameter 2.125 in. Convergent 11 DEG-Half Angle Length ~28 in. Divergent 7.5 DEG-Half Angle Length ~15 in.	955	540	.00610	0	5690	4500
Nozzle 6 Same as Nozzle 5 except Throat Diameter 3.0 in.	925	540	.00825	0	5080	4320
Nozzle 7 2.5 in. long well-rounded Inlet Throat Diameter 1.108 in. 6.5 in. long 7 DEG-Half Angle Diffuser	988	542	0	0	8330	4500
Moody (Fauske Data)	1000	545	0	0	5000	4950
Throad Diameter .25 in. Throat Length >4.0 in.	2000	636	0	0	8200	7650
(Allemann Data)	1250	572	0	0	6000	5760
Throat Diameter 6.80 in. Throat Length >4.0 in.	1600	605	0	0	5800	6750

312 081

Table 1. Data Matrix (Sheet 4 of 4)

NOZZLE GEOMETRY SIZE AND LENGTH	PRESSURE (PSIA)	TEMPERATURE (F)	QUALITY	SUBCOOLING (F)	CRITICAL MASS FLUX ($lbm/Sec\text{-}Ft^2$)	
					EXPERIMENTAL	.9 HEM
NRC STATUS REPORT (Powell Data) Well Rounded Inlet Throat Diameter .438 in. Throat Length .656 in.	1200 1800 2500	450 450 450	NA NA NA	117 171 218	19110 24000 30080	17550 22500 28800
(Zaloudek Data) 20° Conical Reducer Throat Diameter .505 in. Throat Length 10 in.	1000 1400 1800	485 485 535	NA NA NA	60 102 86	14338 17000 17640	13950 18000 17640
Safety Valves @ ATWS Condition Throat Diameter 1 in. to 2 in.	2400 - 3000	560-695	0	0 to 100	None Available	9000-1[illegible]000

NA - NOT AVAILABLE

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