NOV 2 2 1985

MEMORANDUM FOR: Hugh L. Thompson, Jr., Director, Division of Licensing

FROM: Robert M. Bernero, Director, Division of Systems Integration

SUBJECT: SAFETY IMPLICATIONS OF PWR MAIN STEAM SAFETY VALVES FLOW DEFICIENCY

Purpose and Background

Per the Operating Reactors Events Meeting 85-20, held on October 21, 1985, the Division of Systems Integration was assigned to evaluate the safety implications of the Main Steam Safety Valves' (MSSVs') flow deficiency. The purpose of this memorandum is to address this issue.

During the last part of 1984 through mid 1985, Wyle Laboratories conducted several full flow tests on two MSSVs manufactured by Crosby Valve and Gage Company. These two valves were chosen as a representative sample for the MSSVs that are to be installed on the Seabrook plant main steam lines. The purpose of the Wyle tests was to determine the adequacy of different discharge piping configurations. Test measurements indicated that the MSSVs (with the manufacturer's recommended guide ring settings), have a flow capacity of about 50% of their design values at the design pressure. The guide ring settings determine the force being exerted on the valve disc, thereby affecting the degree of valve lift and subsequently the discharge flow capacity of the valve. During the Wyle tests the guide ring settings of both valves were substantially adjusted in order to achieve the full flow capacities. Subsequent to the completion of the tests, the Public Service Company of New Hampshire, the owner of the Seabrook plant, concluded that in order to ensure full flow capacity of the plant's MSSVs they all should be adjusted downward by 130 notches from the manufacturer's settings.

While the Seabrook experience shows a deficiency in the capacity adjustment of the Crosby spring loaded safety valves, it strongly suggests a similar deficiency in similar valves made by other manufacturers, since they all work on the same basic concept.

Safety Implications

Full flow testing of MSSVs is not normally performed by either reactor owners or valve manufacturers, nor is such testing an ASME requirement for capacity certification. Such certification is obtained through extrapolation from tests on much smaller valves at low pressures.

Based on the Seabrook experience and with the lack of sufficient data, it may be assumed that a number of deficient safety valves are installed in some operating plants and/or planned to be installed in plants yet to operate.

CONTACT: S. Diab, RSB, x29440

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It may also be assumed that the capacity of some of these deficient valves may be as low as 50% (as for the Seabrook plant) of the design flow, or even lower. With this potential deficiency, the design basis of the affected plants cannot be met. The design basis of every pressurized water reactor (PWR) requires that overpressure protection for the primary and secondary sides of the plant be provided so that the pressure does not rise above 110% of the design value during anticipated operational occurrences and postulated accidents. PWR vendors perform safety valve sizing analyses such that these valves have sufficient capacity to mitigate the most severe overpressurization event with adequate margin. Generic transient analyses, as opposed to sizing analyses, performed by Westinghouse for the Westinghouse PWRs show that, for the worst overpressurization event (loss of load without condenser bypass) the MSSVs' peak relieving capacity required is about 80% of the nominal valve flow. It should also be noted that any flow degradation through the MSSVs increases the potential for the actuation of the primary safety valve actuation and increases the relieving load on these valves as well. This is because any relief through the MSSVs will remove heat from the reactor coolant, which would otherwise accumulate, expand, and overpressurize the primary system. General Design Criterion 15 of 10 CFR 50, Appendix A, requires that the reactor coolant system and associated auxiliary systems be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during normal operation and anticipated operational occurrences.

There is insufficient test data available to the staff to show whether the valve flow would increase at high upstream pressure conditions and/or if the valve disc travel will be any higher. Therefore, it is not clear whether overpressurizations of this nature are self limiting.

With less than the required relieving capacity, the secondary side is expected to be overpressurized during postulated events, thus increasing the potential for steam side leaks or breaks. While the overpressurization described above would occur following a loss of load event, which is an anticipated operational occurrence, the consequences of that event may lead to a design basis event with its associated severe consequences. While plants are designed to acceptably accommodate their design basis events, the probability of occurrence of those events is sufficiently low. However, if a design basis event were to occur with a substantially higher likelihood or the plant were to be pressurized over its design pressure limits with higher frequency, then the plant cannot meet its design basis requirements.

Similar problems of inadequate ring settings of primary safety valves were discovered during the EPRI test program conducted in response to NUREG-0737, item II.D.1. The problem was identified and confined to valves manufactured by Dresser Industries, Inc. Users of primary side Dresser valves made submittals to justify continued operation of their facilities until the valve ring settings on their plants are readjusted consistent with the EPRI test findings. The staff evaluated and approved those submittals. Verification of the adequacy of all PWR primary safety valve ring settings is being pursued by the staff under multi-plant action MPA F-14.

Based on the limited information available about the adequacy of the secondary side safety valves for overpressurization mitigation, the staff has a reason to doubt that all the safety valves currently in use or planned for use would serve their intended safety function if called upon. Therefore, we suggest that the Division of Licensing send a request for additional information (RAI), per 10 CFR 50.54(f), to PWR plant owners. This RAI would request plant owners, in light of the Seabrook experience, to study the Seabrook experience and justify to the staff that their respective plants continue to have sufficient overpressure protection and are within their safety analyses. The owners' justification may rely on any combination of: (a) relevant experience from which valve performance can be verified; (b) safety analyses assuming inadequate overpressure protection; or (c) representative valve testing.

Original Signed By:
Robert W. Bernero

Robert W. Bernero, Director
Division of Systems Integration

Enclosure:
Suggested 10 CFR 50.54(f) Letter

cc: Edward Jordan, IE
R. Baer, IE
Mark Caruso, DL

DISTRIBUTION:
Docket File
DSI:AD:RS rdg.
DSI:D
RSB rdg.
RSB s/f MSSV Safety
[redacted]
LMarsh
SDiab rdg.
SDiab:js
Doc Name: Diab Rush, js

OFC	:DSI:RSB	:DSI:RSB	:DSI:MEB	:DSI:RSB	:DSI:AD:RS	:DSI:D
NAME	:SDiab:gd	:LMarsh	:FCherny	:BSheron	:RHouston	:RBernero
DATE	:11/21/85	:11/21/85	:11/21/85	:11/21/85	:11/22/85	:11/22/85

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ENCLOSURE

PROPOSED 10 CFR 50.54(f) LETTER

Safety Implications of Main Steam Safety

Valve Flow Deficiency

During the last part of 1984 through mid 1985, Wyle Laboratories conducted several full flow tests on two MSSVs manufactured by Crosby Valve and Gage Company. These two valves were chosen as a representative sample for the MSSVs that are to be installed on the Seabrook plant main steam lines. The purpose of the Wyle tests was to determine the adequacy of different discharge piping configurations. Test measurements indicated that the MSSVs (with the manufacturer's recommended guide ring settings) have a flow capacity of about 50% of their design values at the design pressure. The guide ring settings determine the force being exerted on the valve disc, thereby affecting the degree of valve lift and subsequently the discharge flow capacity of the valve. During the Wyle tests the guide ring settings of both valves were substantially adjusted in order to achieve the full flow capacities. Subsequent to the completion of the tests, the Public Service Company of New Hampshire, the owner of the Seabrook plant, concluded that in order to ensure full flow capacity of the plant's MSSVs they all should be adjusted downward by 130 notches from the manufacturer's settings.

While the Seabrook experience shows a deficiency in the capacity adjustment of the Crosby spring loaded safety valves, it strongly suggests a similar deficiency in similar valves made by other manufacturers, since they all work on the same basic concept.

Based on the Seabrook experience and with the lack of sufficient data, it may be assumed that a number of deficient safety valves are installed in some operating plants and/or planned to be installed in plants yet to operate. It may also be assumed that the capacity of some of these deficient valves may be as low as 50% of the design flow (as for the Seabrook plant), or even lower. With this potential deficiency, the design basis of the affected plants cannot be met. The design basis of every pressurized water reactor (PWR) requires that overpressure protection for the primary and secondary sides of the plant be provided so that the pressure does not rise above 110% of the design value during postulated events. PWR vendors perform safety valve sizing analyses such that these valves have sufficient capacity to mitigate the most severe overpressurization event with sufficient margin. Generic transient analyses, as opposed to sizing analyses, performed by Westinghouse for the Westinghouse PWRs show that, for the worst overpressurization event (loss of load without condenser bypass) the MSSVs' peak relieving capacity required is about 80% of the nominal valve flow. It should also be noted that any flow degradation through the MSSVs increases the potential for the actuation of the primary safety valves and increases the relieving load on these valves as well. This is because any relief through the MSSVs will remove heat from the reactor coolant, which would otherwise accumulate, expand, and overpressurize the primary system. General Design Criterion 15 of 10 CFR 50, Appendix A, requires that the reactor coolant system and associated auxiliary systems be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during normal operation and anticipated operational occurrences.

There is insufficient test data available to the staff to show whether the valve flow would increase at high upstream pressure conditions and/or if the valve disc travel will be any higher. Therefore, it is not clear whether overpressurizations of this nature are self limiting.

With less than the required relieving capacity, the secondary side is expected to be overpressurized, thus increasing the potential for steam side leaks or breaks. While the overpressurization described above would occur following a loss of load event, which is an anticipated operational occurrence, the consequences of that event may lead to a design basis event with its associated severe consequences. While plants are designed to acceptably accommodate their design basis events, the probability of occurrence of those events is sufficiently low. However, if a design basis event were to occur with a substantially higher likelihood or the plant were to be pressurized over its design pressure limits with higher frequency, then the plant cannot meet its design basis requirements.

Similar problems of inadequate ring settings of primary safety valves were discovered during the EPRI test program conducted in response to NUREG-0737, item II.D.1. The problem was identified and confined to valves manufactured by Dresser Industries, Inc. Users of primary side Dresser valves made submittals to justify continued operation of their facilities until the valve ring settings on their plants are readjusted consistent with the EPRI test findings. The staff evaluated and approved those submittals. Verification of the adequacy of all PWR primary safety valve ring settings is being pursued by the staff under multi-plant action MPA F-14.

Based on the limited information available about the adequacy of the secondary side safety valves for overpressurization mitigation, the staff has a reason to doubt that all the safety valves currently in use or planned for use will serve their intended safety function if called upon. Therefore, per 10 CFR 50.54(f), the staff requests that you, as a PWR owner, and in light of the Seabrook experience, justify that your plant continues to have sufficient overpressure protection, your facility continues to be in compliance with GDC 15, and the plant is within its safety analyses. Your justification may rely on any combination of: (a) relevant experience from which valve performance can be verified; (b) safety analyses assuming inadequate overpressure protection; or (c) representative valve testing.

Hugh L. Thompson, Director
Division of Licensing

from [redacted] mtg. minutes — ASME section III Subgroup on Pressure Relief.

ITEM NUMBER

This item was placed on the February 1984 agenda for section III and was withdrawn following adverse comments. Frank Cherny came up with new language that may be acceptable to Sec III Subcommittee. This item will be transmitted to Section III for action. Action is limited to NB-7000. Section III approved this item without negative vote. Floyd Moschini phone the Chairman and raised objections to the definitions. Frank Cherny will check with him..

NP-1-85 Review of NX-7000 Paragraphs For Clearity and Uniformity

The SGPR continued with the review of the NX-7000 paragraphs to clearify intent and maintain uniformity to the extent possible, for all classes. The view will be continued at the next meeting.

NP-1-86 Production Testing of Class 2 Main Steam Safety Valves-NC-7512.1

Frank Cherny will report on this item.

Next Meeting

The next meeting will be held on March 12 & 13, 1986 at the United Engineering Center, 6th Floor Board Room at 10:00 a.m. This will be a two day meeting in order to complete item NP-1-85. There will be no Agenda and this will be the only notice of the meeting.

Frank Catudal

Frank W. Catudal

Chairman, SGPR

B/67

NOV 2 2 1985

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FROM: Robert M. Bernero, Director, Division of Systems Integration

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Robert W. Bernero

Robert W. Bernero, Director
Division of Systems Integration

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Suggested 10 CFR 50.54(f) Letter

cc: Edward Jordan, IE
R. Baer, IE
Mark Caruso, DL

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OFC	:DSI:RSB	:DSI:RSB	:DSI:MEB	:DSI:RSB	:DSI:AD:RS	:DSI:D
NAME	:SDiab:gd	:LMarsh	:FCherny	:BSheron	:RHouston	:RBernero
DATE	:11/21/85	:11/21/85	:11/21/85	:11/21/85	:11/22/85	:11/22/85

ENCLOSURE

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Hugh L. Thompson, Director
Division of Licensing

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

[redacted]

Docket Nos. 50-317
and 50-318

MEMORANDUM FOR: Ashok C. Thadani, Director
PWR Project Directorate #8
Division of PWR Licensing-B

FROM: D. H. Jaffe, Project Manager
PWR Project Directorate #8
Division of PWR Licensing-B

SUBJECT: SUMMARY OF MEETING WITH BALTIMORE GAS & ELECTRIC COMPANY (BG&E) CONCERNING OPERABILITY OF STEAM LINE SAFETY VALVES

On November 26, 1985, representatives of BG&E and the NRC staff met in Room 2242 of the Air Rights Building in Bethesda, Maryland. Enclosure 1 contains the list of attendees. The purpose of the meeting was to discuss the operability of the Calvert Cliffs Unit 2 Main Steam Safety Valves (MSSVs).

The NRC staff indicated concern regarding the as-found setpoints for the Unit 2 MSSVs. The setpoints violated existing Technical Specification (TS) requirements and would have violated the new, more liberal (TS) requirements to be issued as part of a Unit 2, Cycle 7, license amendment.

BG&E responded to staff concerns regarding the Unit 2 MSSVs in a presentation summarized in Enclosure 2. Although several minor MSSV problems seemed to exist, no single or cumulative cause for the setpoint problem could be identified. Evidence presented by BG&E seemed to point to a problem associated with setpoint measurement techniques. In light of this finding, BG&E committed to the following corrective actions related to MSSV setpoint verification:

- Procedural Enhancements
- Set valves at 530°F vice 500°F
- Provide QC coverage while verifying setpoints
- Independently reverify setpoints of 4 valves 12 hours after initial setting.
- Verify setpoints of 4 valves during first outage after 4 month operation.

BG&E also presented its conclusions regarding the Unit 1 MSSV setpoints. This material is contained in Enclosure 3.

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Following the BG&E presentation and discussions among the NRC staff, it was concluded by the NRC staff that based upon information presented by BG&E: (1) safety analyses performed by BG&E, assuming as-found MSSV setpoints, showed no violation of safety limits or the criteria of 10 CFR 50.46, (2) improvements proposed by BG&E would likely improve MSSV setpoint measurement, and (3) the Unit 1 MSSVs appeared to be showing as-expected MSSV setpoint behavior. Based upon the above, it was concluded that no safety problems associated with MSSVs could be identified which would prevent the return of Unit 2 to power operation.

D. H. Jaffe, Project Manager
PWR Project Directorate #8
Division of PWR Licensing-B

Enclosures:
As stated

cc w/enclosures:
See next page

Mr. A. E. Lundvall, Jr.
Baltimore Gas & Electric Company

Calvert Cliffs Nuclear Power Plant

cc:
Mr. William T. Bowen, President
Calvert County Board of
Commissioners
Prince Frederick, Maryland 20768

D. A. Brune, Esq.
General Counsel
Baltimore Gas and Electric Company
P. O. Box 1475
Baltimore, Maryland 21203

George F. Trowbridge, Esq.
Shaw, Pittman, Potts and Trowbridge
1800 M Street, NW
Washington, DC 20036

Mr. R. C. L. Olson, Principal Engineer
Nuclear Licensing Analysis Unit
Baltimore Gas and Electric Company
Room 720 - G&E Building
P. O. Box 1475
Baltimore, Maryland 21203

Resident Inspector
c/o U.S.Nuclear Regulatory Commission
P. O. Box 437
Lusby, Maryland 20657

Mr. Leon B. Russell
Plant Superintendent
Calvert Cliffs Nuclear Power Plant
Maryland Routes 2 and 4
Lusby, Maryland 20657

Bechtel Power Corporation
ATTN: Mr. D. E. Stewart
Calvert Cliffs Project Engineer
15740 Shady Grove Road
Gaithersburg, Maryland 20760

Mr. R. M. Douglass, Manager
Quality Assurance Department
Baltimore Gas and Electric Company
Fort Smallwood Road Complex
P. O. Box 1475
Baltimore, Maryland 21203

Regional Administrator, Region I
U.S. Nuclear Regulatory Commission
Office of Executive Director
for Operations
631 Park Avenue
King of Prussia, Pennyslvania 19406

Mr. Charles B. Brinkman
Manager - Washington Nuclear Operations
Combustion Engineering, Inc.
7910 Woodmont Avenue
Bethesda, Maryland 20814

Mr. J. A. Tiernan, Manager
Nuclear Power Department
Calvert Cliffs Nuclear Power Plant
Maryland Routes 2 and 4
Lusby, Maryland 20657

Mr. R. E. Denton, General Supervisor
Training and Technical Services
Calvert Cliffs Nuclear Power Plant
Maryland Routes 2 and 4
Lusby, Maryland 20657

Combustion Engineering, Inc.
ATTN: Mr. R. R. Mills, Manager
Engineering Services
P. O. Box 500
Windsor, Connecticut 06095

Department of Natural Resources
Energy Administration, Power Plant
Siting Program
ATTN: Mr. T. Magette
Tawes State Office Building
Annapolis, Maryland 21204

Enclosure 1
List of Attendees

NRC

D. Jaffe
T. Foley
F. C. Cherny
A. Thadani
M. Caruso
G. Hammer
M. S. Wegner
R. Perfetti

BG&E

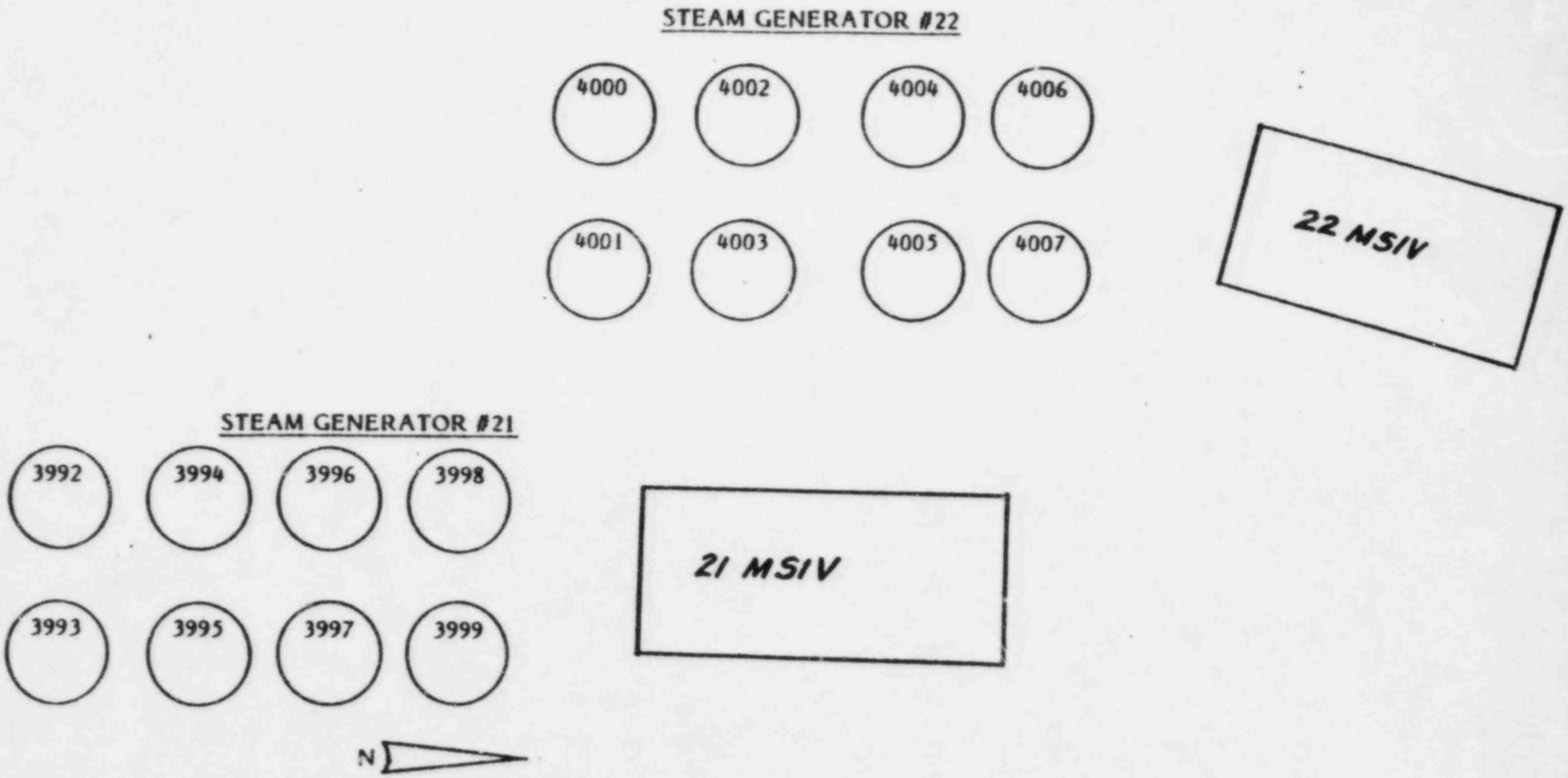
L. B. Russel
J. F. Williams
J. A. Mihalcik
J. T. Carroll
R. R. Allen

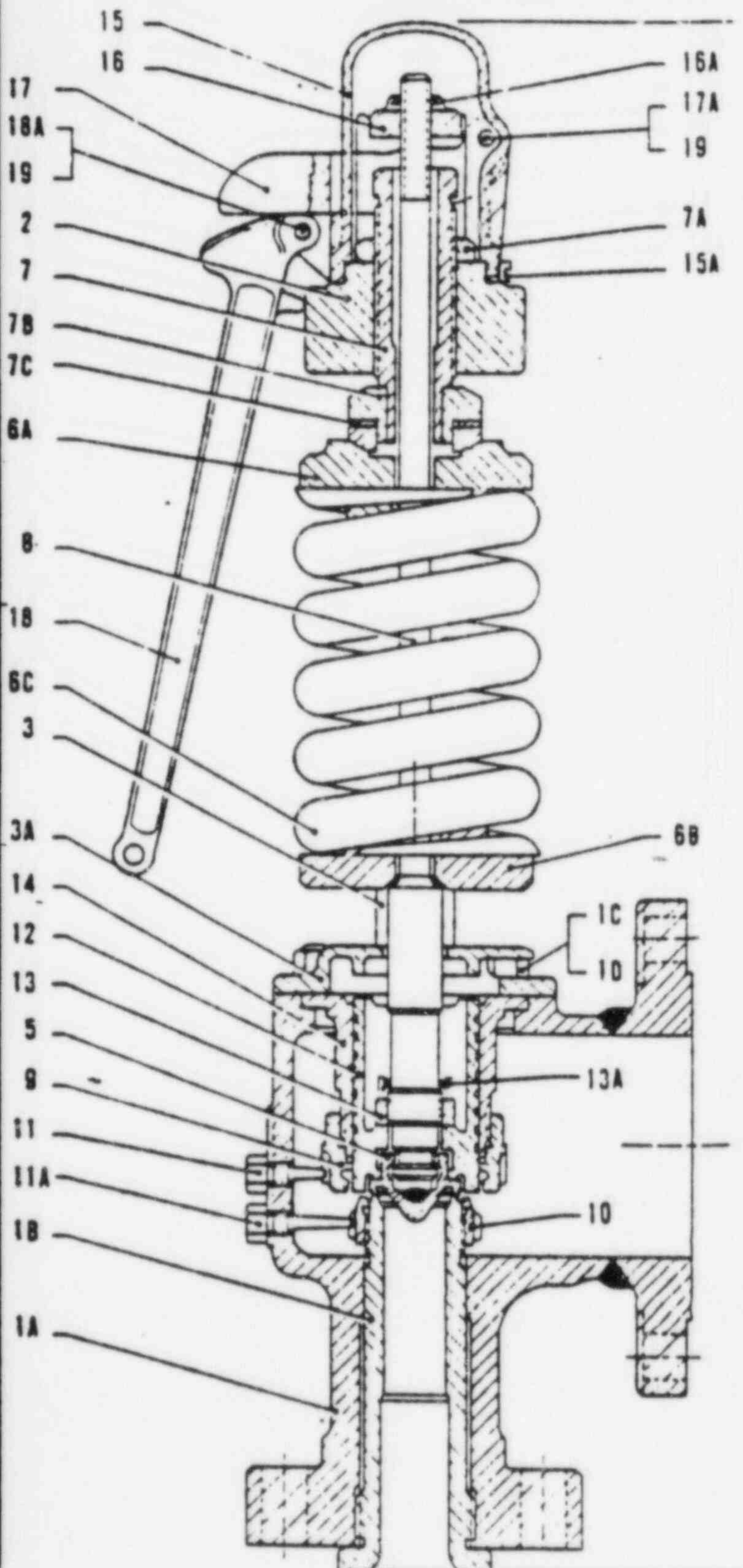
ENCLOSURE 2

AGENDA
U-2

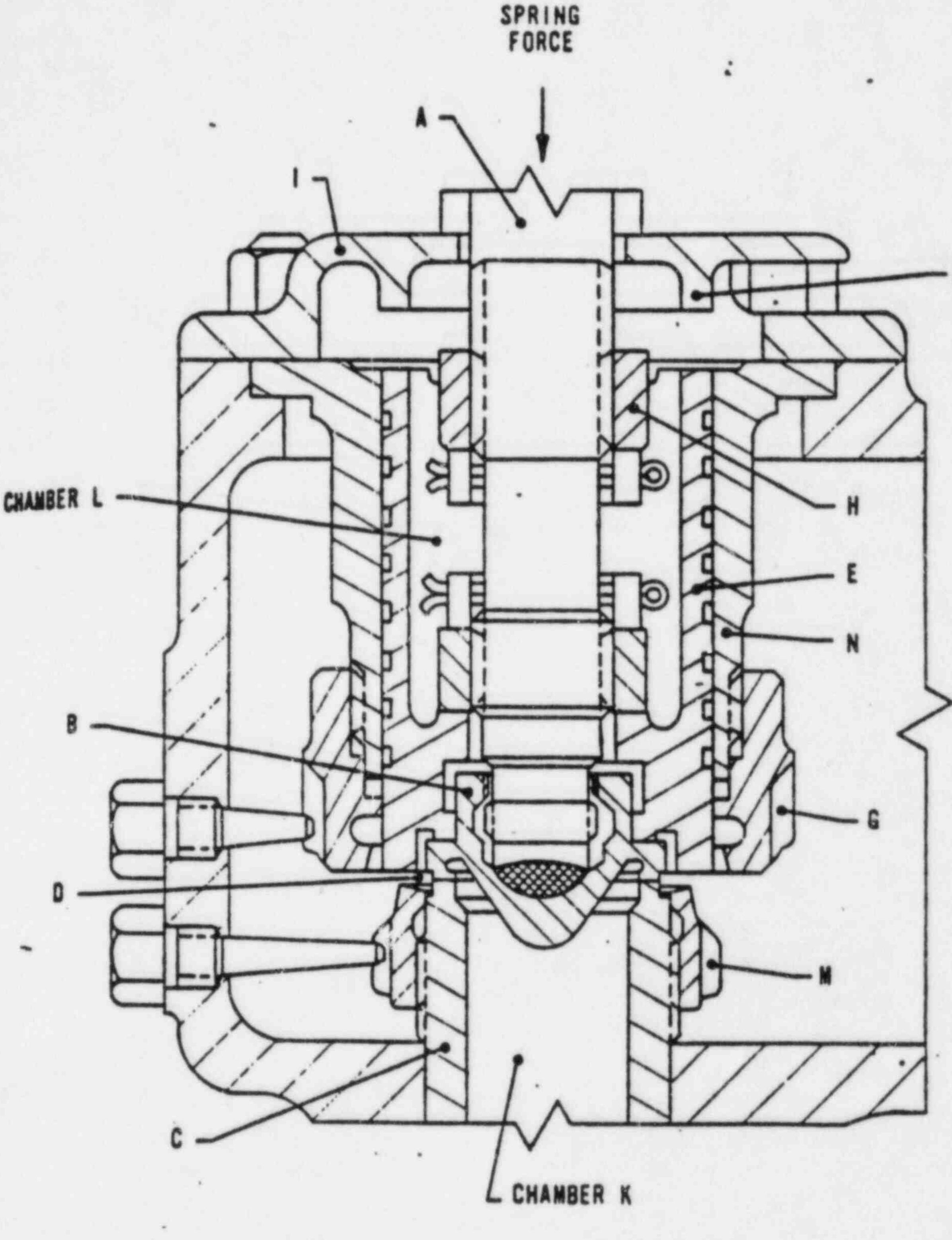
- Describe Safety Valves
- Describe As-Found Results
- Results of Safety Analysis
- Outline Test Program
- Test Program Results
- Conclusions
- Future Actions

u2agenda

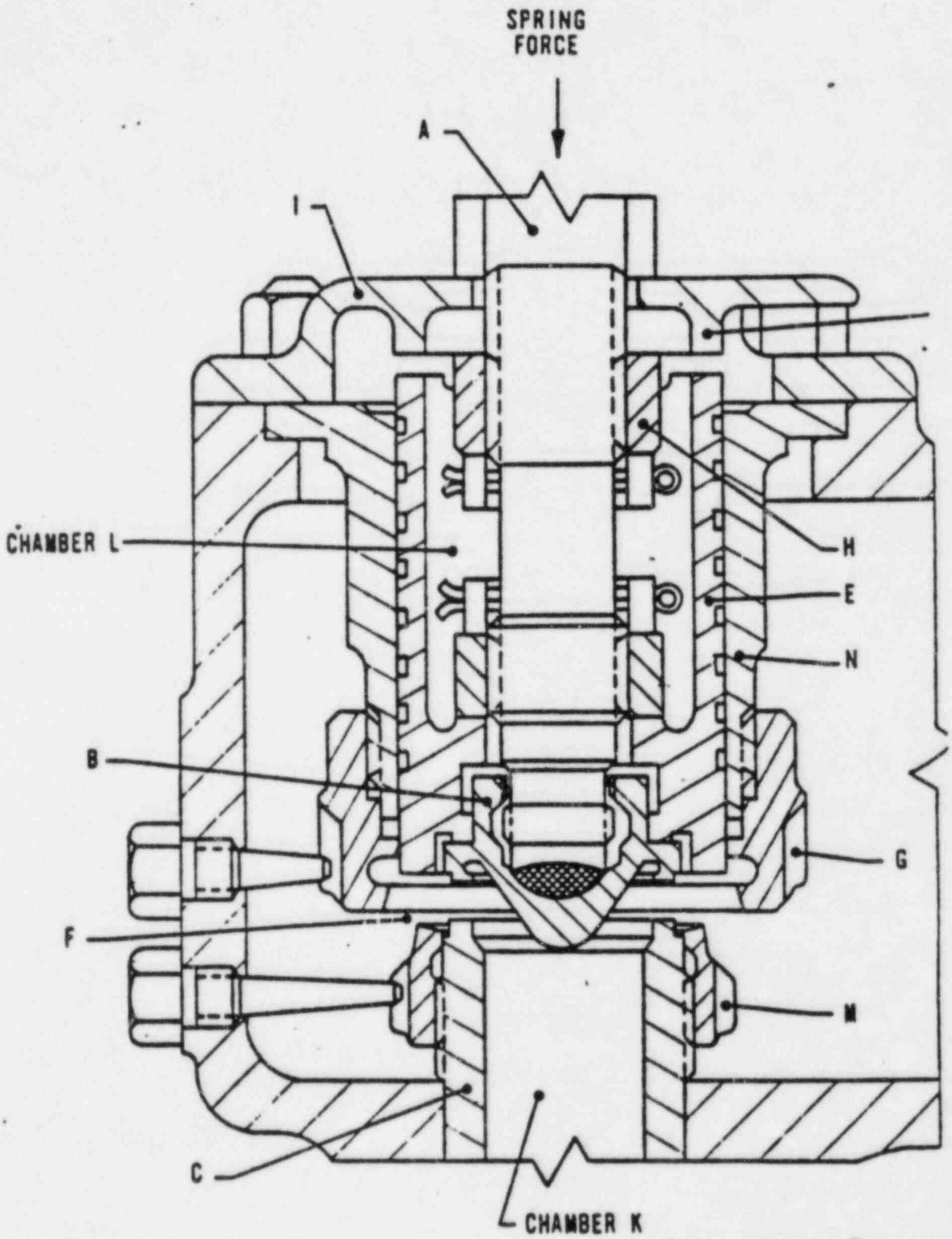

STEAM GENERATOR #22
4000
4002
4004
4006
4001
4003
4005
4007
22 MSIV
STEAM GENERATOR #21
3992
3994
3996
3998
3993
3995
3997
3999
21 MSIV
N



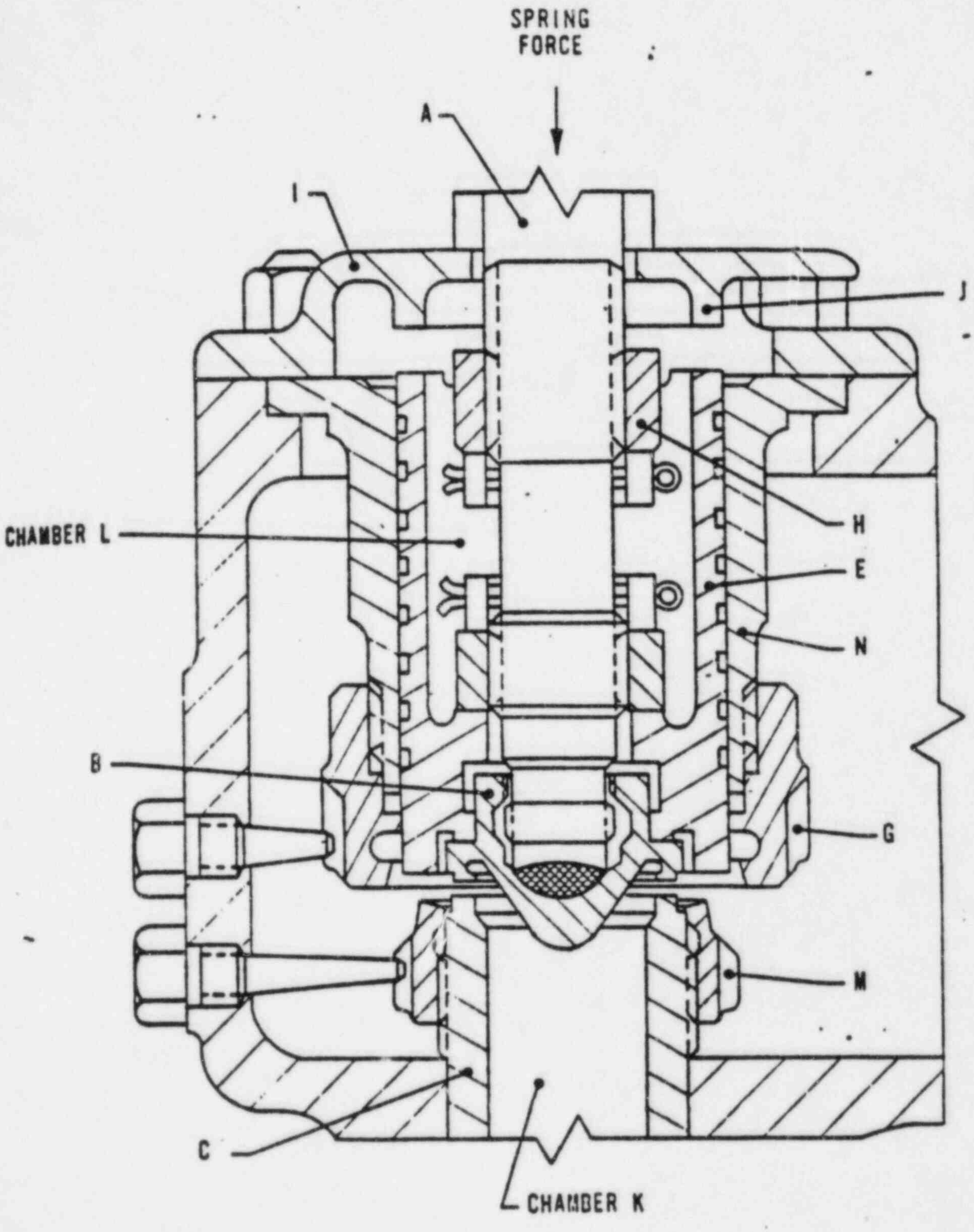
REF NO	QTY	NOMENCLATURE
1		BASE ASSEMBLY
1A	1	BASE
1B	1	NOZZLE
1C	10	BASE STUD
1D	10	BASE STUD NUT
2	1	YOKE
3	2	YOKE ROD
3A	1	COVER PLATE
4	4	YOKE ROD NUT (NOT SHOWN)
5	1	DISC
6		SPRING ASSEMBLY
6A	1	TOP SPRING WASHER
6B	1	BOTTOM SPRING WASHER
6C	1	SPRING
7	1	COMPRESSION SCREW
7A	1	COMPRESSION SCREW LOCKNUT
7B	1	COMPRESSION SCREW ADAPTER
7C	1	THRUST BEARING
8	1	SPINDLE
9	1	UPPER ADJ. RING
10	1	LOWER ADJ. RING
11	1	UPPER RING PIN
11A	1	LOWER RING PIN
12	1	DISC HOLDER
13	1	DISC COLLAR
13A	1	DISC COLLAR COTTER PIN
14	1	GUIDE
15	1	CAP
15A	1	CAP SET SCREW
16	1	RELEASE NUT
16A	1	RELEASE NUT COTTER PIN
17	1	TOP LEVER
17A	1	TOP LEVER PIN
18	1	DROP LEVER
18A	1	DROP LEVER PIN
19	2	LEVER COTTER PIN



Valve closed



- *Valve in full lift*



Valve closing

MSSV UNIT 2 LIFT SETPOINT TEST
(AS FOUND/AS LEFT PSIG)

Valve	Setpoint (± 1% PSIG)	Oct. 1982	Apr. 1984	Oct. 1985	As Found With Hydroset Correction
3992	985		1009/982	991/991	987
3993	985		963/975	1015/985	1011
3994	995		1022/996	1001/1001	997
3995	995		962/989	1035/1004	1031
3996	1015	1021/1021	1037/1006	1024/1024	1020
3997	1015	1015/1015	1044/1008	1020/1020	1016
3998	1035	929/1035±10	1065/1038	1044/1044	1040
3999	1035	1032/1032	1053/1038	1057/1038	1053
4000	985		974/981	1037/993	1033
4001	985	987/987	972/985	1040/992	1036
4002	995	998/998	1004/1004	1059/993	1055
4003	995	996/996	1014/990	1047/997	1043
4004	1015		1018/1018	1070/1018	1065
4005	1015		1012/1012	1054/1018	1050
4006	1035		991/1036	1104/1035	1100
4007	1035		980/1034	1106/1039	1102
Test Results:		1 Low 0 High 6 Sat	6 Low 7 High 3 Sat	0 Low 11 High 5 Sat	0 Low 11 High 5 Sat

mssvu2

ACCIDENTS PREDICTING MSSV OPERATION

- Loss of Load
- Loss of Load to One Steam Generator
- CEA Withdrawal
- Feed Line Break
- Loss of Non-Emergency AC Power
- Loss of Feedwater
- Small Break LOCA

MAIN STEAM SAFETY VALVE SAFETY ANALYSIS

	Nominal Setpoint PSIG	As-Found Setpoint PSIG	As-Found Analysis PSIG	U1C8* Analysis PSIG
RV-3992/4000	985	991/1037	1037/1037	1035**
RV-3993/4001	985	1015/1040	1040/1040	1035**
RV-3994/4002	995	1001/1059	1059/1059	1035
RV-3995/4003	995	1035/1047	1047/1047	1035
RV-3996/4004	1015	1024/1070	1070/1070	1065
RV-3997/4005	1015	1020/1054	1054/1054	1065
RV-3998/4006	1035	1044/1104	1104/1104+	1065
RV-3999/4007	1035	1057/1106	1106/1106+	1065

* This configuration has been shown to be applicable for U2C6.

** Small Break LOCA assumes 995 PSIG.

\+ RV-4006 and RV-4007 were assumed stuck closed for Loss of Load analysis. RV-3998 and RV-3999 opened at 991 and 1001 PSIG, respectively.

RESULTS

SMALL BREAK LOCA

- Available High Pressure Safety Injection flow is higher than assumed in analysis.

- Higher flow compensates for higher MSSV setpoints.

NON-LOCA SAFETY ANALYSIS

LIMITING EVENTS

- Loss of Load
- Loss of Load to one steam generator

LIMITING PARAMETER

- Peak secondary system pressure: Limit less than 1100 PSIA.

RESULTS

NON-LOCA SAFETY ANALYSIS

LOSS OF LOAD TO ONE STEAM GENERATOR

- As-found setpoints used
- Otherwise identical to U1C8 analysis
- Peak secondary system pressure = 1080 psia

LOSS OF LOAD

- As-found setpoints used
- MTC = 0.0E-4 delta rho/F vs. +0.7E-4 delta rho/F (U1C8)
- Otherwise identical to U1C8 analysis
- Peak secondary system pressure = 1093 psia

MSSV TESTING PROGRAM

2-RV-3993

1. At $70^{o}F$ ambient, $500^{o}F$ steam

   a. Thermal equilibrium
   b. Test set pressure
   c. Full Flow Test
   d. Check leakage
   e. Retest set pressure.

2. At $70^{o}F$ ambient, $530^{o}F$ steam, repeat 1a - e.

3. Heat up transient at $120^{o}F$ ambient, $530^{o}F$ steam prior to thermal equilibrium, repeat 1b - d.

4. Heat up to thermal equilibrium at $120^{o}F$ ambient, $530^{o}F$ steam, repeat 1b - e.

5. Set with new hydroset, check with old hydroset at 985, 995, 1015, and 1035 psig.

6. Reset to 985 psig at $120^{o}F$ ambient and $530^{o}F$ steam, repeat 1b - e and check with 2 full flow tests.

2-RV-3992

1. At $120^{o}F$ ambient, $530^{o}F$ steam

   a. Thermal equilibrium
   b. Test set pressure
   c. Full Flow Test
   d. Check leakage
   e. Retest set pressure

NOTE: For all tests, record value of temperature vs. time for inlet nozzle, body, spring, and outlet flange.

2RV

Wyle Test Valve BM-7771
11/12 - 14/85

1. Valve Set at 985 $\pm$ 10
   Ambient - Avg. 86.3°F
   System - Avg. 495.5°F

   Hydroset - Avg. 975.6 psi set pressure
   Full Flow - Avg. 986.3 psi pop pressure

2. Ambient - Avg. 74.6°F
   System - Avg. 537.7°F

   Hydroset - Avg. 966.3 psi set pressure
   Full Flow - Avg. 979 psi pop pressure

3. Ambient - Avg. 119.8°F — 1 hr. heat up
   System - Avg. 526.2°F

   Hydroset - Avg. 973.8 psi
   Full Flow - Avg. 979 psi

4. Ambient - Avg. 118.3°F — 4 hr. heat up
   System - Avg. 527°F

   Hydroset - Avg. 965.8 psi set pressure
   Full Flow - Avg. 976 psi pop pressure

5. Valve set at 1035 $\pm$ 10

   Ambient - Avg. 119.5°F
   System - Avg. 527.5°F

   Hydroset - Avg. 1035.6 psi set pressure
   Full Flow - Avg. 1030.5 pop pressure

6. Valve Reset at 985 $\pm$ 10

   Ambient - Avg. 119°F
   System - Avg. 523.5°F

   Hydroset - Avg. 967.5 psi set pressure
   Full Flow - Avg. 981.3 psi pop pressure

7. Wyle Test Valve BM 7787 - 11/14/85

   Ambient - Avg. 122.7°F
   System - Avg. 523°F

   Hydroset - Avg. 966.3 psi set pressure
   Full Flow - Avg. 984.5 psi pop pressure

wyle

Valve 2-RV	Ring Setting Top (Note 2)	Ring Setting Bottom (Note 2)	Surface Film/Cond.	Stem Run Out (Note 1)	Avg. Disk to Guide Clearance **	Maximum Set Pressure (PSIG)	1985 As Found Set Pressure (PSIG)
3992	-7t	-2	very light/good	OK	19t	995	987
3993	+22t	+12	very light/good	OK	17t	995	1011
3994	113t	-15	very light/good	8t	18t	1005	997
3995	6t	-2	very light/some wear	OK	15t	1005	1031
3996	-11t	-3	very light/good	15t	15t	1025	1020
3997	12t	-2	very light/good	8t	13t	1025	1016
3998	13t	-	heavy/good	10t	14t	1045	1040
3999	-7t	-1	heavy/good	24t	13t	1045	1053
4000	4t	-9	very light/good	21t	14t	995	1033
4001	12t	-1	heavy/good	14t	14t	995	1036
4002	95t	-3	very light/good	10t	14t	1005	1055
4003	12t	-7	heavy/good	13t	13t	1005	1043
4004	25t	-3	very light/good	29t	11t	1025	1065
4005	52t	-3	very light/good	20t	15t	1025	1050
4006	27t	-1	very light/good	13t	13t	1045	1100
4007	22t	-1	very light/good	18t	12t	1045	1102

$t = 10^{-3}$ inches

**Min. old disk to guide clearance 10t

**Min. new disk to guide clearance 15t

1. No effect below 0.0625"
2. As-Found ring positions affect setpoint by less than 1% and yield 15% or less blowdown.

U-2

CONCLUSIONS

Apparent setpoint changes not explained by as-found condition of valves.

Apparent setpoint changes may possibly be the result of measurement technique.

Rebuilt valves will perform as designed.

FUTURE ACTIONS

Procedural Enhancements

Set valves at 530°F vice 500°F

Provide QC coverage while verifying setpoints

Independently reverify setpoints of 4 valves 12 hours after initial setting.

Verify setpoints of 4 valves during first outage after 4 month operation.

ENCLOSURE 3

AGENDA
U-1

- Past Test Results
- Estimated Current Condition of Valves
- Conclusions
- Future Actions

MSSV UNIT 1 LIFT SETPOINT TEST
(AS FOUND/AS LEFT PSIG)

Valve	Setpoint (±1% PSIG)	Hydroset Test Oct. 11, 1983	Hydroset Test April, 1985	As Left Hydroset Test June, 1985
3992	985		959/975	933
3993	985		975/988	
3994	995		995/995	
3995	995		987/987	990
3996	1015	989/1024	1024/1024	
3997	1015		1014/1014	
3998	1035		1023/1028	
3999	1035		1010/1034	
4000	985	990/990	980/980	978
4001	985	981/981	964/983	983
4002	995		961/996	990
4003	995		987/987	999
4004	1015		1024/1024	1016
4005	1015		986/1010	1012
4006	1035	1058/1040		1030
4007	1035	1043/1043		
Test Results:		1 Low 1 High 3 Sat	6 Low 0 High 8 Sat	

mssvu1

U-1

CONCLUSIONS

Valves will open, provide full capacity, and reset as designed.

No safety implications

FUTURE ACTIONS

Verify setpoints of all 16 valves during next outage.

Reset any valves outside ± 1%

If necessary to reset any valve, verify setpoint of valves during first outage after 3 months operation.