UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION I
631 PARK AVENUE
KING OF PRUSSIA, PENNSYLVANIA 19406

JUN 1 0 1985

MEMORANDUM FOR: Robert L. Baer, Chief, Engineering and Generic Communication Branch, IE

FROM: Jacque P. Durr, Chief, Engineering Branch, Division of Reactor Safety, RI

SUBJECT: POTENTIAL GENERIC ISSUE CONCERNING RING SETTINGS OF CROSBY SAFETY VALVES

Our memorandum to you dated March 6, 1985, Enclosure 1, advised you of a potential generic issue concerning ring settings which produce only 50% lift on Crosby Steam Generator Safety Valves (SG-SVs). A 10 CFR 50.55(e) Construction Deficiency report was issued by Public Service of New Hampshire (PSNH) advising of the ring setting problem at Seabrook Station. Our major concern was not with the installation at Seabrook, a plant under construction, but that operating plants could have SG-SVs that may not achieve full lift; whereby, actual flow capacities and/or accumulation pressure are not representative of analyses.

We have held several phone conversations with both PSNH and Crosby to determine if the observed lift deficiences are common to this model of valve and possible effects on other plants. To date, we have been unsuccessful in establishing this fact. PSNH sent us a copy of the Wyle test data, Enclosure 2, which we forwarded to MEB (NRR). PSNH is planning additional tests to resolve their specific installation problem. In our conversations with Crosby, we were told they were awaiting a purchase order to perform testing. Regarding Crosby's response, we believe there should be a stronger commitment and confirmation that their factory guide ring setting of +150 is correct if such is the case.

Based on our review of the PSNH/Wyle test data of the Crosby 6R10 SV (this valve is in the "R" orifice size category), it is apparent that design lift is not achieved with the factory ring settings. We call to your attention that the "R" orifice is at the upper end of commercial valve sizes and has a nozzle bore dia. of 4.513 in. (16.00 sq. in. area). Generally, the capacity certification and functional tests required by ASME Section III are performed by the 9 valve coefficient of discharge method where the manufacturers testing is done on much smaller valves and the design is then extrapolated to the larger size valves. This raises the concern that functional demonstration of these large SVs may never have been performed. Additionally, the problem may extend to other SG-SV suppliers.

We are recommending that NRC Headquarters advise licensee's of the problem through an IE Information Notice, Bulletin or Generic Letter and request licensee's to show that their SG-SVs have full lift.

H. Gregg of my staff is available to provide assistance or additional information at FTS 488-1295.

Jacque P. Durr

Jacque P. Durr, Chief
Engineering Branch, DRS, RI

XA

Enclosures: As Stated

8604230319
2pp.

cc w/o encls:
S. Ebneter, Director, DRS
S. Collins, DRP
R. Gallo, DRP
A. Cerne, SRI
F. Cherny, NRR
H. Gregg, EB

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

SEP 0 9 1985

MEMORANDUM FOR: Edward L. Jordan, Director
Division of Emergency Preparedness
and Engineering Response
Office of Inspection and Enforcement

FROM: Hugh L. Thompson, Jr., Director
Division of Licensing
Office of Nuclear Reactor Regulation

SUBJECT: IE NOTICE REGARDING MAIN STEAM SAFETY VALVE TEST FAILURES AT WYLE LABORATORIES

The purpose of this memorandum is to transmit a proposed IE Information Notice which describes failures of Crosby PWR Main Steam Safety Valves (MSSV) to attain full disc lift during full flow tests performed at Wyle Laboratories for Public Service Company of New Hampshire's (PSNH) Seabrook plant. The tests showed that the disc only traveled 50% of the required distance. A lift of 50% in these valves translates to a flow capacity which is approximately 50% of rated capacity at rated pressure.

The test failures were originally reported by PSNH as a potential design deficiency per 10 CFR 50.55(e). The test reports have been under review by NRR Division of Engineering per the February 5, 1985 request of your Engineering and Generic Communications Branch Chief. Our review, which included discussions with both Crosby and Wyle Laboratories, has been completed. Based on this review, the Mechanical Engineering Branch agrees with the PSNH evaluation which concluded that the vendor specified, factory set guide ring position was too high, resulting in an inadequate lift force on the valve disc. They believe the problem may exist in operating plant MSSVs due to similarities among vendors in valve design and methods for determining ring settings. A preliminary evaluation of the consequences of a 50% degradation in MSSV flow capacity indicates that such a degradation may likely result in overpressurization of the main steam system in some plants should a full load rejection event occur with the steam dump and bypass system and anticipatory reactor trip unavailable. The pressure transient could be more severe in B&W designed units due to the relatively small liquid inventory in the once-through steam generator design.

CONTACT:
M. Caruso, NRR
x27940

XA

~~8509190396~~

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The enclosed draft Information Notice describes the Crosby valve test failures and the findings from the Division of Engineering's review of those tests. In light of the significance of this potential equipment deficiency, we recommend that you issue the proposed IE Notice. NRR will continue to investigate the generic implications and safety significance of this equipment deficiency to determine if additional staff action is required.

We have discussed this proposed IEN with R. Oller and Bob Baer of your office.

Frank J. Miraglia

for/ Hugh L. Thompson, Jr., Director
Division of Licensing
Office of Nuclear Reactor Regulation

Enclosure:
Draft IE Notice

cc: M. Wegner, IE
L. Marsh
J. Durr, RI
R. Baer, IE
J. Knight
F. Cherny
G. Hammer
V. Nerses
H. Nicolaras
R. Oller
J. Stolz
G. Knighton

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT
WASHINGTON, D.C. 20555
SEPTEMBER , 1985

IE Information Notice No.: 85-xx MAIN STEAM SAFETY VALVE TEST FAILURES AND RING SETTING ADJUSTMENTS

Addressees:

All PWR nuclear power reactor facilities holding an operating license (OL) or a construction permit (CP)

Purpose:

This Information Notice is being provided as a notification of a potentially significant problem pertaining to spring-actuated main steam safety valves (See Figure 1), that may possess less than the full rated flow capacity required for overpressure protection of the secondary cooling system in PWRs. It is expected that recipients will review the information for applicability to their facilities and consider actions, if appropriate, to preclude a similar problem at their facilities. However, suggestions contained in this Information Notice do not constitute NRC requirements; therefore, no specific action or written response is required.

Description of Circumstances:

Between October 16, 1984, and December 1, 1984, Wyle Laboratories conducted several full flow steam tests on two separate main steam safety valves (MSSVs) manufactured by Crosby Valve and Gage Company. These Crosby 6R10 MSSVs are to be installed by Public Service of New Hampshire on the Seabrook main steam system. The tests were conducted in order to determine the adequacy of various MSSV discharge piping arrangements. During the tests the valves were instrumented to measure valve disk lift. The valves were installed on the test facility with the settings of the valve adjusting rings (see Figure 1) as received from the valve vendor. With these ring settings the valve achieved about 50% of the full disk lift required to develop full steam flow capacity within the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code required 3% accumulated overpressure limit. Adequate lift was not achieved for either valve with these factory adjusted ring settings, even for the largest diameter (least flow resistance) vent pipe tested. The guide ring of both valves was subsequently adjusted to a lower position by a significant amount (150 notches) during the course of testing and full disk lift was subsequently achieved.

These types of full flow tests are not normally performed by either reactor owners or the valve vendor on MSSVs, nor are such tests required for capacity certification according to the ASME Code, Section III. In general, these valves are capacity certified by tests on much smaller size valves, and the capacities then extrapolated to larger size valves. The MSSVs on most PWRs, while not necessarily the same model or supplied by the same vendor, are like those at Seabrook in that they are generally at the upper end of the valve size range. This raises the concern that full flow functional demonstration of some valve types may never have been performed, and that due to incorrect ring settings, the valve may not be capable of providing relief capacity in accordance with facility design requirements.

Based on the full flow tests performed at Wyle Laboratories, Public Service Company of New Hampshire (PSCNH) has concluded that the guide ring setting for the Seabrook MSSVs should be adjusted downward 150 notches to ensure full flow capacity. The MSSVs will be installed at Seabrook with the guide ring adjusted downward 150 notches from the as delivered, factory adjusted setting.

No specific action or written response is required by this Information Notice. If you have any questions about this matter, please contact the Regional Administrator of the appropriate regional office or this office.

Edward L. Jordan, Director
Division of Emergency Preparedness
and Engineering Response
Office of Inspection and Enforcement

Technical Contact: G. Hammer, NRR
301-492-8963

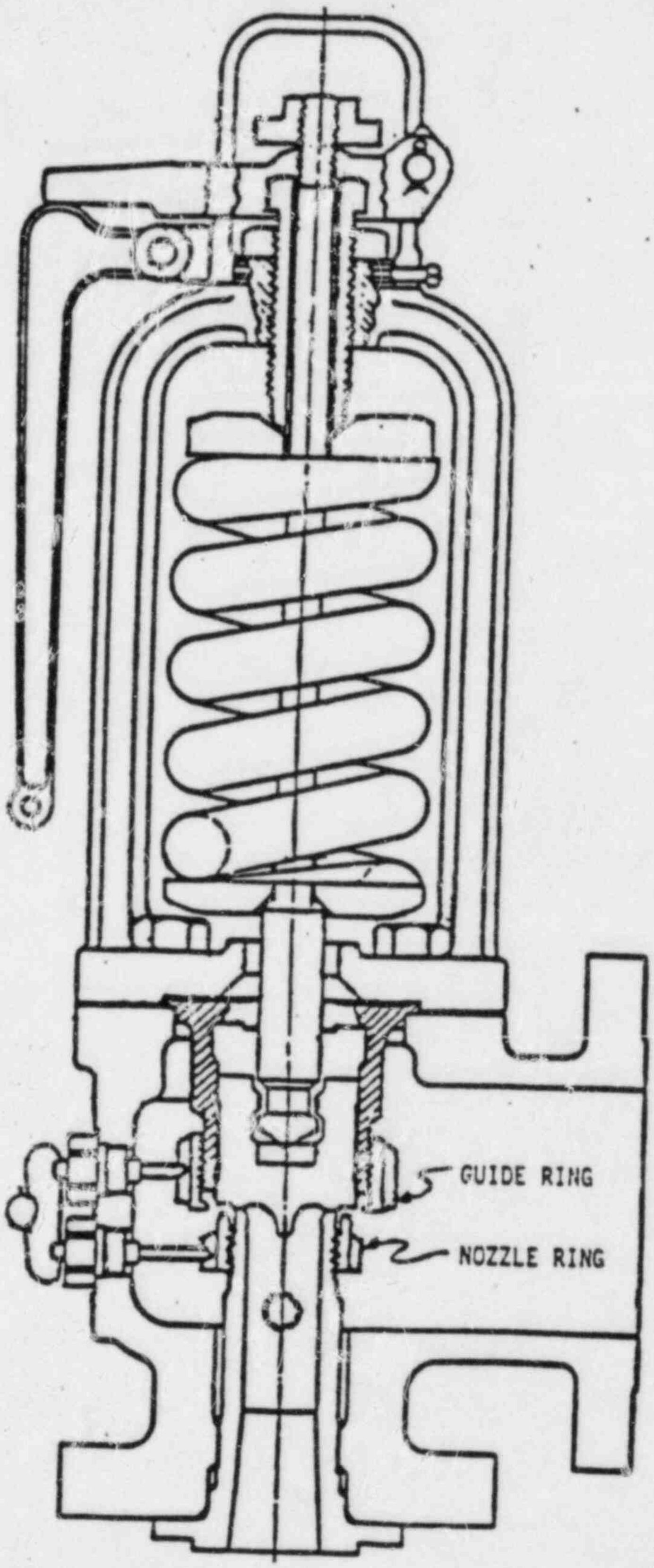

FIG. 1

TYPICAL MAIN STEAM SAFETY VALVE

D. Hammer

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

SEP 9 1985

MEMORANDUM FOR: Gary Holahan, Chief
Operating Reactors Assessment Branch
Division of Licensing

FROM: Frank C. Cherny, Acting Chief
Mechanical Engineering Branch
Division of Engineering

SUBJECT: MEB COMMENT ON DRAFT NUREG-0844 "NRC INTEGRATED PROGRAM FOR THE RESOLUTION OF USI A-3, A-4 AND A-5 REGARDING STEAM GENERATOR TUBE INTEGRITY

In the subject draft NUREG, there is information regarding the probability of a main steam safety valve (MSSV) sticking open after the valve has been challenged with water resulting from an overfilled steam generator. The estimated probability given for this event is 3.0 X 10^{-2}/demand. It is further stated that this probability is largely based on the results of liquid testing of pressurizer safety valves performed by EPRI in response to NUREG-0737, Item II.D.1. However, based on statements in the draft NUREG, we feel obliged to point out that the EPRI data has been misinterpreted in the NUREG. Our review of this data indicates that the probability of such valves sticking open or leaking severely after relieving is probably much higher. Of the approximately 26 safety valve water tests performed by EPRI, about 1/3 resulted in disk to seat chatter which either caused severe internal damage to the valves or was terminated within a very short time (few seconds) by manually opening the valves in order to prevent severe valve damage. Therefore, this data alone, we believe, would seem to conservatively indicate a failure probability of about 0.3. However, the actual probability for a specific event would be dependent on the liquid temperature, the flow rate, and the duration of flow.

Some of the EPRI water tests were conducted with saturated liquid and much better performance was observed (mild fluttering instead of chatter) for these than for more subcooled liquid tests. The information we have indicates that safety valves, originally designed for steam service, perform better for saturated liquid because the water flashes to steam directly under the valve disk thus assisting the disk to lift. Conversely, such safety valves are likely to chatter at more subcooled liquid conditions because less flashing takes place under the disk.

The flow rates during the EPRI water tests were considerably less than the full flow the valves would be capable of. This is clearly shown in the data since the measured disk lifts were usually significantly less than maximum. However, the disk to seat chattering was still severe in some tests and we suspect that as full liquid flow conditions are approached the chattering could become yet more severe since greater disk travel would occur. Also, the EPRI tests were conducted for short time periods (a few seconds) especially for those tests which were terminated because of chattering. Longer periods of liquid discharge would result in more disk and seat damage.

In summary, it is difficult to precisely characterize the expected damage to a MSSV for a given water flow event and to accurately quantify the probability of a stuck open or severely leaking valve. However, it is our opinion that the probability is much higher than the 3×10^{-2} value given in the draft NUREG. At the very least, we recommend that the final NUREG not convey the impression that the EPRI test results show that spring safety valves designed for steam work well on water. The exact opposite is the case.

Frank C. Cherny

Frank C. Cherny, Acting Chief
Mechanical Engineering Branch
Division of Engineering

cc: J. Knight
R. Bosnak
D. Crutchfield
T. Marsh
E. Murphy
G. Hammer
T. Speis
B. Sheron

E1B
FYI

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT
WASHINGTON, D.C. 20555

January 31, 1986

IE INFORMATION NOTICE NO. 86-05: MAIN STEAM SAFETY VALVE TEST FAILURES AND RING SETTING ADJUSTMENTS

Addressees:

All pressurized-water-reactor (PWR) facilities holding an operating license (OL) or a construction permit (CP).

Purpose:

This notice is being provided to alert recipients of a potentially significant problem pertaining to spring-actuated main steam safety valves that may possess less than the full-rated flow capacity. It is expected that recipients will review the information for applicability to their facilities and consider actions, if appropriate, to preclude a similar problem at their facilities. However, suggestions contained in this information notice do not constitute NRC requirements; therefore, no specific action or written response is required.

NRC is continuing to obtain and evaluate pertinent information. If specific actions are determined to be required by NRC, an additional notification will be made.

Description of Circumstances:

In the fall of 1984, Public Service of New Hampshire sent the main steam safety valves (MSSVs) for its Seabrook plant to Wyle Laboratories for full-flow testing to determine the proper vent stack size. To determine full flow, Wyle measured disc travel of the model number 6R10 valves manufactured by the Crosby Valve and Gage Company. The results of the tests indicated that the valves could not achieve the required disc travel with the factory-set ring setting (+155 notches). The disc travel achieved was 50% of the full lift necessary to develop required steam flow capacity. Adequate lift was not attainable even with the largest diameter tailpipe.

Additional tests were done in July 1985 to determine the appropriateness of the ring settings. Specifically, the tests were to determine if the "as-shipped" ring settings of the valves would allow the required disc travel with minimum tailpipe backpressure and to determine the effects on valve disc travel for a range of backpressures between 180 and 390 psig. During these tests, the upper (guide) ring setting was adjusted from +155 to 0 and then to +25 to achieve the required disc travel. This is a substantial adjustment. Subsequently, the licensee consulted with the valve manufacturer and agreed on ring settings of +25 for the guide ring and -25 (the original setting) for the lower (nozzle) ring (see figure 1).

Full flow, full size tests of the sort described in this notice are not normally performed by the licensee or valve vendor for large secondary safety valves, nor are they required by the ASME Code, Section III. Instead the valves are certified by extrapolations on data from tests of smaller valves.

The MSSVs on most PWRs, while not necessarily the same model or manufacturer as those at Seabrook, are generally at the upper end of the valve size range. This raises the concern that full-sized flow demonstration may never have been performed for many MSSVs and these may have incorrect ring settings. In addition, similar problems with ring settings have been found when full-size tests were performed for PWR primary safety valves. Thus, these MSSVs may not be capable of providing full-relief capacity in accordance with facility design requirements.

NRC is continuing to obtain and evaluate pertinent information. If specific actions are determined to be required by NRC, an additional notification will be made.

No specific action or written response is required by this information notice. If you have any questions about this matter, please contact the Regional Administrator of the appropriate regional office or this office.

E L Jordan

Edward L. Jordan, Director
Division of Emergency Preparedness
and Engineering Response
Office of Inspection and Enforcement

Technical Contact: Mary S. Wegner
(301) 492-4511

Attachments:
1. Figure 1, Typical Main Steam Safety Valve
2. List of Recently Issued IE Information Notices

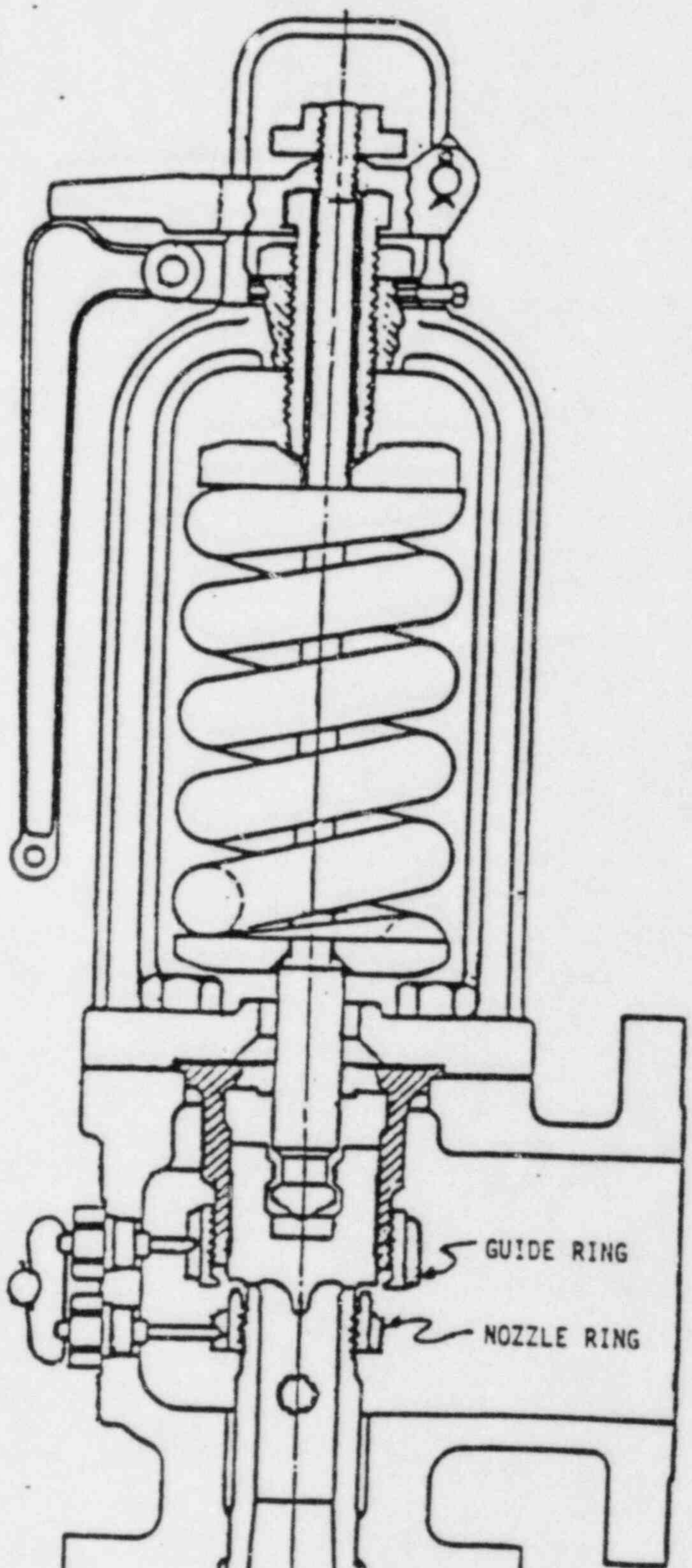

FIG. I

TYPICAL MAIN STEAM SAFETY VALVE

L. Hammer

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

SEP 27 1985

MEMORANDUM FOR: Dennis Crutchfield, Assistant Director for Safety Assessment
Division of Licensing

FROM: Robert J. Bosnak, Acting Assistant Director for Components and Structures Engineering
Division of Engineering

SUBJECT: JULY 23, 1985 OPERATING REACTOR EVENTS BRIEFING OPEN ITEM - INADEQUATE MAIN STEAM SAFETY VALVE CAPACITY AT SEABROOK

At the Operating Reactor Events briefing on the subject issue on July 23, 1985, DE was asked to investigate the adequacy of testing of PWR main steam safety valves (MSSVs) and the validity of extrapolating test data from small valves to larger full size MSSVs. The Mechanical Engineering Branch (MEB) has planned the following tasks which we think adequately respond to this request:

(1) MEB has prepared a proposed IE Information Notice to advise the industry of the MSSV capacity problem as it relates to proper ring settings. The proposed Notice was transmitted to DL with our August 9, 1985 memorandum from R. Bosnak to D. Crutchfield.

(2) MEB will formally request DST to prioritize a potential generic issue dealing with MSSV operability problems including that of inadequate flow capacity. The procedure outlined in Office Letter No. 40 will be followed.

(3) MEB will discuss with the ASME Section III Subgroup on Pressure Relief possible changes to the ASME Section III Code Class 2 safety valve certification requirements. Currently CL. 2 safety valves can be capacity certified based on tests performed on prototypical valves much smaller in size and at much lower pressures than are applicable for PWR, Main Steam Safety Valves. The ring adjustment problem encountered with the Seabrook MSSVs raises one of the same questions that arose during the recent EPRI testing of ASME Section III CL. 1 pressurizer safety valves. That is, do the valve manufacturers have an adequate understanding of how to extrapolate ring adjustments, that affect lift and blowdown, from very small test valves to the very large safety valves used on PWR plants?

Recently changes to the Code safety valve certification procedure, proposed by MEB, to address this concern for CL. 1 safety valves were accepted by ASME for incorporation into the Code. The change will require that new CL. 1 safety valve designs be prototypically tested in sizes and at pressures, temperatures, and flow rates that envelope those that the valve design will be used for in service.

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MEB will explore with the Code Committee the feasibility/desirability of making similar changes to the Code CL. 2 safety valve certification requirements.

We believe these actions should adequately resolve the problem of inadequate MSSV capacity.

/S/

Robert J. Bosnak, Acting Assistant Director
for Components and Structures Engineering
Division of Engineering

cc: J. Knight
F. Cherny
B. Sheron
G. Holahan
D. Tarnoff
R. Baer
H. Gregg

C

Please see previous concurrence.

C	:DE:MEB	:DE:MEB	:DE:MEB	:DE:AD/CSE	:	:	:
ME	:5520/vgt	:[redacted]	:FCherny	:RJBosnak	:	:	:
TE	:9/25/85	:9/ /85	:9/26/85	:9/27/85	:	:	:

OFFICIAL RECORD COPY

Calvert Cliffs Unit 2 MSSV's - Rec'd. 11-26-85 from BG&E

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.

Docket No. 50-346

License No. NPF-3

Serial No. 1193

October 18, 1985

TOLEDO EDISON

JOE WILLIAMS, JR.
Senior Vice President—Nuclear
(419) 249-2300
(419) 249-5223

Director of Nuclear Reactor Regulation
Attention: Mr. John F. Stolz
Operating Reactor Branch No. 4
Division of Licensing
United States Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Stolz:

This letter is in response for information requested during a September 30, 1985 telephone conversation between Mr. G. Hammer (NRC MEB) and members of the Toledo Edison Nuclear Staff. The requested information concerns details regarding Action Plan No. 16 contained in Toledo Edison's Course of Action (Section IV.C.1.1) submitted September 10, 1985 (Serial No. 1182).

Request No. 1

Provide information regarding the required size of the inlet piping to the Main Steam Safety Valves (MSSV's) versus the installed inlet piping size.

Response

The Dresser Instruction Manual for Type 3700 Consolidated Safety Valves, May 1978 edition, requires the inlet pipe for the MSSV's to have at least a six inch bore. This requirement does not appear in the original instruction manual nor on the drawings which were supplied with the valves. Piping for the inlet pipe was specified in the original installation to be six inch schedule 160 which has an inside dimension of 5.189 inches. This dimension has been verified by field measurements. The concern is that the "as built" configuration does not conform to the National Board configuration which was used for certification purposes.

Further evaluation indicates this condition will not impair MSSV performance. The maximum flow restriction occurs in the upper portion of the nozzle. In this area, the flow diameters for Dresser valves with Q and R orifices are respectively 3.750 and 4.515 inches. Furthermore, there has never been any concern raised regarding the valves capability to relieve an overpressure condition. Per Dresser recommendation, flow testing will be performed to ensure that all applicable ASME Code and National Board requirements are met. This concern is being tracked by Nonconformance Report 85-0117.

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NRR

Request No. 2

Provide information regarding any planned piping modifications resulting from analysis conducted under Action Plan No. 16.

Response

To ensure that the main steamlines properly support the MSSV's, analyses are underway to examine the flexibility of the Main Steam System piping. Fluid transient loads and structural system response will be included in the analysis. A modification is being considered, as an interim measure, to add a restraint to the steamline for Steam Generator No. 1-2. The purpose of this restraint is to increase the stiffness of the main steam line.

Request No. 3

Provide the testing history for the Main Steam Safety Valves (MSSV's), Atmospheric Vent Valves (AVV's), and Main Steam Isolation Valves (MSIV's).

Response

There have been several occurrences of MSSV blowdown in excess of rated 3%. Prior to the 1984 refueling outage, reseat pressure experienced during recent plant trips ranged from approximately 980 to 900 psig.

In March 1984, the A4 MSSV stuck open after a reactor trip resulting in boiling dry Steam Generator 1-2. The root cause of this occurrence was failure of a cotter pin permitting the release nut to travel unrestricted down the spindle threads. Maintenance Procedure MP 1401.28, Main Steam Safety Valve Disassembly Inspection/Repair and Reassembly, has been revised to require installation of new stainless steel cotter pins when maintenance is performed on a MSSV.

Previous maintenance experience on MSSV's has shown:

a. Excessive wear of guides and holders
b. Bending of spindles
c. Damage to the disc seats requiring replacement
d. Greater maintenance requirements for the low set pressure MSSV's

All MSSV's on the No. 2 ("A") header were rebuilt during the 1984 refueling outage. Valve B2 was rebuilt in March 1985. Valves B1 and B7 were rebuilt in 1983, and the other valves on the B header were last rebuilt in 1982.

Four of the eighteen installed MSSV's have a smaller capacity ("Q" orifice). These valves have required considerably less maintenance than the large ("R") orifice MSSV's.

Both AVV's were rebuilt during the 1984/1985 refueling outage.

Recent maintenance and testing histories of the MSSV's, AVV's and Integrated Control System were reviewed. Nothing of significance was noted.

Additional Inservice Test (IST) histories for the MSSV's, AVV's and MSIV's were previously been made available for NRC Mechanical Engineering Branch review.

Request No. 4

Provide the results of the "as received" testing of the first eight MSSV's which were tested at Wyle Laboratory.

Response

The Wyle Laboratory test results for the first eight MSSV's tested in the "as received" condition is provided as an Attachment to this letter. The first part of this testing was witnessed by NRC personnel from the Vendor Programs Branch.

Very truly yours,

JW Williams Jr

JW:LLH:DJS:lah
attachments

cc: DB-1 NRC Resident Inspector

Davis-Besse Main Steam Safety Valves (as received condition)

Wyle Test Data

Valve Position	Parameter	Run 1	Run 2	Run 3	Run 4	Run 5	Run 6	Leakage	Disposition
A1	Disc lift	1.16	0.20	1.16	1.18			No Leakage	Rebuild
	Setpoint	1024	995	993	1009				
	Blowdown	3.8	NM	1.1	2.8				
A2	Disc lift	1.06	1.06	1.04				Yes	Rebuild
	Setpoint	1020	1025	1015					
	Blowdown	4.3	19.0	2.7					
A3	Disc lift	1.13	1.14	0.25	1.14			No Leakage	Rebuild
	Setpoint	1025	1020	1031	1025				
	Blowdown	1.3	2.5	NM	1.8				
A4	Disc lift	0.62	1.12	1.12	1.12			No Leakage	Rebuild
	Setpoint	1014	1015	1006	1005				
	Blowdown	NM	5.3	5.7	5.2				
B1	Disc lift	1.15	1.14	1.16	NM	1.14		Yes	Rebuild
	Setpoint	1042	1041	1044	1044	1040			
	Blowdown	4.6	4.5	5.2	5.2	19.7			
B2	Disc lift	1.12	1.09	1.10	1.09*	1.10	1.10	No Leakage	Passed
	Setpoint	1051	1037	1037	1042	1043	1049		
	Blowdown	4.8	3.2	2.7	2.6	2.8	3.1		
B3	Disc lift	1.16	1.20	1.20	1.20			Yes	Rebuild
	Setpoint	1074	1070	1077	1070				
	Blowdown	1.8	2.8	2.4	3.4				
B4	Disc lift	1.20	1.20	1.18				Yes	Rebuild
	Setpoint	1080	1071	1069					
	Blowdown	3.3	2.5	2.3					

All disc lifts in inches
All setpoints in psi
All blowdown in percent
NM Indicates the parameter was not measured

*Compression screw was adjusted one flat clockwise prior to this run.

sm b/40

(~ 11 ~

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

AUG 2 1985

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

FROM: Dennis M. Crutchfield, Assistant Director
for Safety Assessment, DL

SUBJECT: SUMMARY OF THE OPERATING REACTORS EVENTS
MEETING ON JULY 23, 1985 - MEETING 85-12

On July 23, 1985, an Operating Reactor Events meeting (85-11) was held to brief the Office Director, the Division Directors and their representatives on events which occurred since our last meeting on July 1, 1985. The list of attendees is included as Enclosure 1.

The events discussed and the significant elements of these events are presented in Enclosure 2. In addition, the assignment of follow-up review responsibility was discussed. The assignments made during this meeting and the status of previous assignments are presented in Enclosure 3.

Completion dates have been assigned for items in Enclosure 3. Note that we have revised Enclosure 3 to facilitate computerized tracking and provide additional details regarding responsibilities and status. Each assignee should review Enclosure 3 with regard to their respective responsibilities and advise ORAB if the target completion date cannot be met. If an assignee has any questions, please contact D. Tarnoff, x29526.

Dennis M. Crutchfield
Dennis M. Crutchfield, Assistant Director
for Safety Assessment, DL

Enclosures:
As stated

cc w/encl:
See next page

~~8508120333~~
8pp.

cc: H. Denton
R. Bernero
J. Knight
T. Speis
C. Heltemes
T. Novak
W. Russell
J. Taylor
E. Jordan
F. Rowsome
W. Minners
L. Shao
T. Ippolito
S. Varga
J. Zwolinski
E. Sullivan
D. Beckham
G. Edison
K. Seyfrit
T. Murley, R-I
J. Nelson Grace, R-II
J. Kepper, R-III
R. D. Martin, R-IV
J. B. Martin, R-V
R. Starostecki, R-I
R. Walker, R-II
C. Norelius, R-III
R. Denise, R-IV
D. Kirsch, R-V
G. Lainas
Baranowski, RES
E. Rossi, IE
R. Hernan
F. Schroeder
G. Knighton
D. Silver
J. Lyons
D. Brinkman
E. Weiss
R. Baer
J. Stolz
E. Butcher
R. Bosnak
P. Morriette
W. Jones
G. Hammer
D. Osborne
R. Caruso
D. Lynch
D. McDonald
D. Neighbors
V. Nerses
H. Nicolaras
J. Wilson
P. O'Connor
T. Alexion
K. Jabbour
L. Olshan
H. Booher
B. Sheron
F. Cherney

ENCLOSURE 1

LIST OF ATTENDEES

OPERATING REACTORS EVENTS BRIEFING (85-11)

JULY 23, 1985

H. Denton, NRR

M. Caruso, NRR

J. Jackson, NRR/DE/EQB

B. Jones, IE/DEPER/EAB

F. Cherny, MEB/DE/NRR

W. Swenson, ORAB/DL/NRR

H. Nicolaras, NRR/DL/ORB#4

K. Mitchell, NRR/DL/ORAB/ORB#5

J. Stone, IE/VPB

G. Bagchi, NRR/DE/EQB

J. Wilson, LB#3/DL/NRR

T. Alexion, LB#1/DL/NRR

L. N. Olshan, LB#1/DL/NRR

P. O'Connor, LB#1/DL/NRR

W. J. Collins, IE/DEPER

N. P. Kadambi, NRR/DL/LB#3

D. Humenansky, OCM/COMM ZECH

T. Rotella, NRR/DL/ORB#5

B. Sheron, NRR/DSI/RSB

B. Bosnak, NRR/DE

A. W. Dromerick, IE/DEPER/EGCB

E. Weiss, IE/DEPER/EAB

N. Lauben, NRR/DSI/RSB

J. Stolz, NRR/DL/ORAB#4

D. Neighbors, NRR/DL/ORB#1

G. Knighton, NRR/DL/LB#3

H. Rood, NRC/DL/LB#3

J. Lyons, NRR/DL

R. Bernero, NRR/DSI

T. M. Novak, NRR/DL

D. Beckham, NRR/DHFS

R. Wessman, NRR/DL

T. Speis, NRR/DST

F. Schroeder, NRR/DST

W. Minners, NRR/DST

S. Varga, DL

D. Crutchfield, DL

J. P. Knight, NRR/DE

G. Lanik, IE/EAB

S. Schwartz, IE/DEPER

H. Thompson, NRR/DL

P. Morriette, NRR/DL

D. Tarnoff, NRR/DL

K. Seyfrit, AEOD/ROAB

D. Zukor, AEOD/ROAB

E. J. Brown, AEOD/ROAB

<u>SEABROOK - CROSBY MAIN STEAM SAFETY VALVE</u>

<u>FLOW DEFICIENCY - DECEMBER 1984</u>

<u>(G. HAMMER, NRR)</u>

- <u>PROBLEM</u> - FULL FLOW TEST RESULTS INDICATE SPRING-ACTUATED MAIN STEAM SAFETY VALVES MAY NOT ACHIEVE RATED FLOW CAPACITY.

- <u>SAFETY SIGNIFICANCE</u> - POSSIBLE INADEQUATE OVERPRESSURE PROTECTION OF SECONDARY SYSTEM IN PWRs USING THESE VALVES

- WYLE LAB TEST RESULTS: INADEQUATE LIFT OF VALVE DISK (ABOUT 50%) WITH THE VENDOR (CROSBY) RECOMMENDED RING SETTING ADJUSTMENTS. TESTS WERE CONDUCTED TO DETERMINE ADEQUACY OF DISCHARGE PIPING.

- CORRECTIVE ACTION - RINGS READJUSTED. OBTAINED FULL LIFT ON SEABROOK VALVES

- GENERIC IMPLICATION - SEABROOK VALVES AND DISCHARGE PIPING SIMILAR TO OTHER PWRs. FULL FLOW TESTS NOT NORMALLY RUN TO ADJUST RINGS.

- NRC FOLLOWUP ACTION: -
  (1) DEVELOPING IE INFORMATION NOTICE
  (2) STAFF MAY PURSUE AS A GENERIC ISSUE
  (3) DISCUSSIONS WITH CROSBY BY REGION 1 AND NRR REGARDING ADEQUACY OF VENDOR GUIDANCE AND SRV RING SETTINGS.

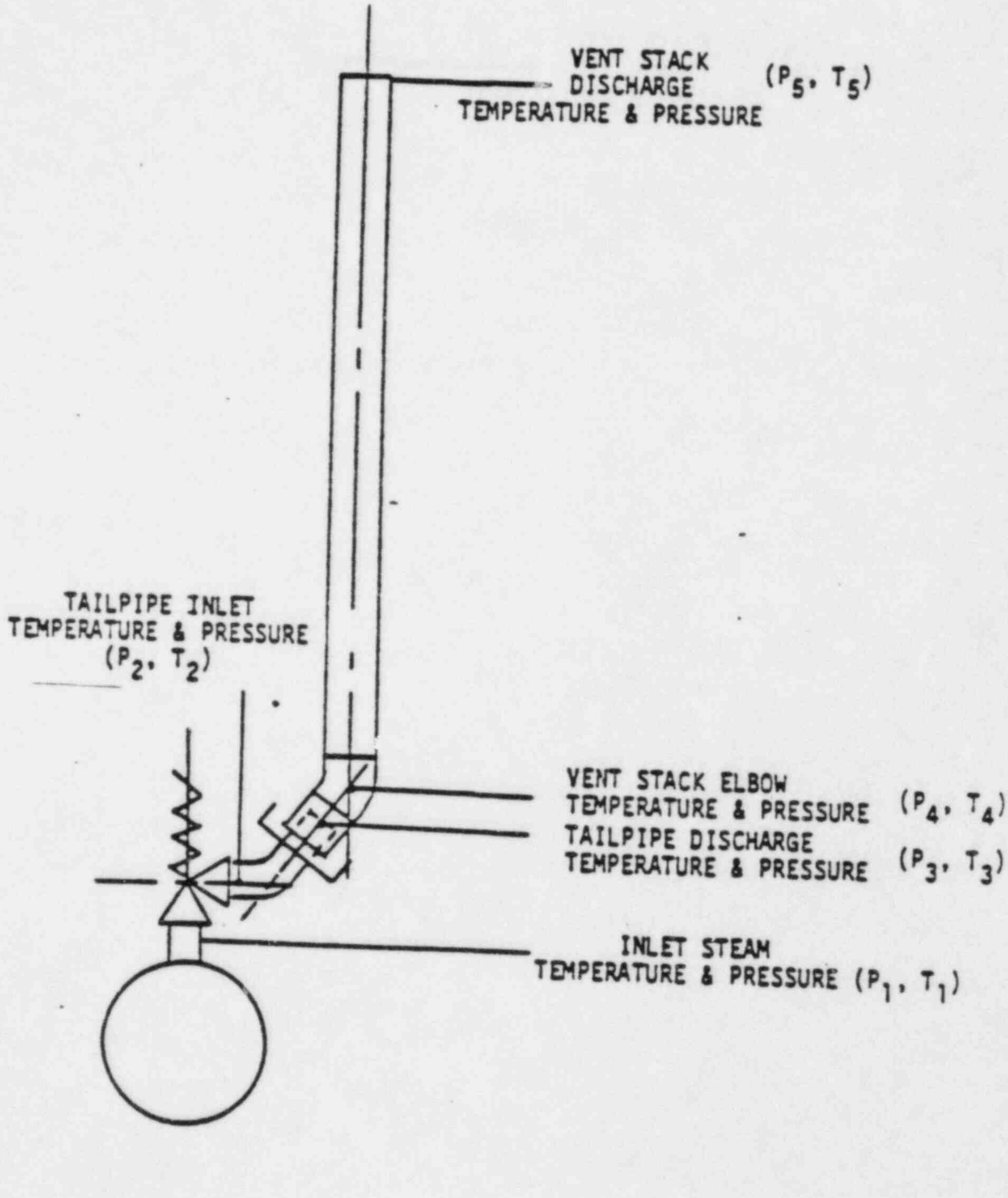

FIGURE 1. INSTRUMENTATION LOCATIONS (16, 18, AND 20-INCH VENT STACKS)

OCONEE 2 - EXTENDED BLOWDOWN FROM MAIN STEAM SAFETY VALVES
JULY 11, 1985 (H. NICOLARAS, NRR)

- OCONEE UNIT 2 REACTOR TRIP FROM 94% POWER CAUSED BY PERSONNEL ERROR

- TWO MAIN STEAM SAFETY VALVES DID NOT RESEAT AT SETPOINT - EXTENDED BLOWDOWN - TO ABOUT 990 PSI

- TO RESEAT VALVES, OPERATORS REDUCED STEAM PRESSURE THROUGH TURBINE BYPASS VALVES.

- FAILURE OF CROSBY MAIN STEAM SAFETY VALVES TO PROPERLY RESEAT HAS ALSO OCCURRED REPEATEDLY AT OCONEE UNIT 1

- IMPROPER RING SETTING IS A LIKELY CAUSE OF EXCESS BLOWDOWN, BUT NOT CONFIRMED.

- DUKE POWER COMMITTED CORRECTIVE ACTIONS TO REGION II

- SUMMARY OF PLANTS REPORTING SIMILAR BLOWDOWN PROBLEM

PLANT	KNOWN # OF EVENTS
OCONEE 1, 2, 3	31
TROJAN	1
SALEM	1

CAUSE: UNKNOWN

POSTULATED PROBLEM WITH RING SETTINGS

CORRECTIVE ACTIONS: READJUST, RESET SETPOINT, VALVE DISASSEMBLY

4 (OUT OF 16) VALVES REWORKED DURING EACH REFUELING

LICENSEE PURSUING METHODS FOR CHECKING BLOWDOWN SETTINGS

IMPLICATIONS: RCS OVERCOOLING

CHALLENGE TO ADDITIONAL SYSTEMS

POTENTIAL PRECURSOR TO STUCK OPEN VALVE OR FULL LIFT CAPACITY

OPERATING REACTORs' EVENTS MEETING FOLLOWUP ITEMS
AS OF MEETING 85-12 ON JULY 23, 1985

(ORDERED BY ASCENDING MEETING DATES, NSSS VENDORS, FACILITY)

EETING UMBER/ EETING ATE	FACILITY NSSS VENDOR/ EVENT DESCRIP.	RESPONSIBLE DIVISION/ INDIVIDUAL	TASK DESCRIPTION	SCHEDULE COMPLET. DATE(S)	CLOSED DATE BY DOCUMENT/ MEETING, ETC.	COMMENTS
5-12 /23/85	SEABROOK W / CROSBY MAIN STEAM SAFETY VALVE FLOW DEFICIENCY 12/84	DSI /MARSH /	ANALYZE SAFETY IMPLICATIONS OF VALVE FLOW DEFICIENCY	09/30/85 / / / /	OPEN / /	
5-12 7/23/85	SEABROOK W / CROSBY MAIN STEAM SAFETY VALVE FLOW DEFICIENCY 12/84	DE /CHERNEY /	INVESTIGATE ADEQUACY OF TESTING AND VALIDITY OF EXTRAPOLATING DATA FROM SMALL TO LARGE VALVES	09/30/85 / / / /	OPEN / /	
5-12 /30/85	HATCH 1 GE / STUCK OPEN SAFETY RELIEF VALVE	DRAB/CARUSO, M. /	WILL DEVELOP TIA TO COORDINATE IE NOTICE AND FURTHER INVESTIGATIVE EFFORTS.	07/30/85 / / / /	CLOSED 07/23/85 TIA IN CONCURRENCE	

PSNH

Public Service of New Hampshire

New Hampshire Yankee Division

Seabrook Station
Engineering Office

August 27, 1985

SBN- 863
T.F. Q2.2.2

E. Brown	Projects - WJD
A. Cerne	Projects - Chrono
R. Cummings	Ropes & Gray (3)
R. DeLoach	F. Sabadini
W. Derrickson	A. Shepard
J. DeVincentis	R. Sweeney
T. Feigenbaum	T.F. Q2.2.2
G. Gram	G. Thomas
W. Hall	H. Tracy
R. Harrison	J. Tribble
D. Hunter	UE&C & W (SB-19770)
W. Johnson	M. Wiley
G. Kingston	ASLB
G. F. McDonald	10CFR50.55(e)File
M. McKenna	J. Allen
B. Middleton	INPO
D. Moody	
NRC Subject File	

United States Nuclear Regulatory Commission
Region I
631 Park Avenue
King of Prussia, PA 19406

Attention: Mr. Richard W. Starostecki, Director
Division of Project and Resident Programs

References:
(a) Construction Permits CPPR-135 and CPPR-136, Docket Nos. 50-443 and 50-444
(b) Telecon of December 21, 1984, A. L. Legendre, Jr. (YAEC) to J. Grant (Region I)
(c) NHY Letter SBN-751 dated January 17, 1985, John DeVincentis to R. W. Starostecki, NRC Region I
(d) NHY Letter SBN-788 dated April 8, 1985, John DeVincentis to R. W. Starostecki

Subject: Final 10CFR50.55(e) Report, "Main Steam Safety Valve Ring Setting Deficiency," (CDR 84-00-19)

Dear Sir:

In References (c) and (d), we filed interim 10CFR50.55(e) reports regarding a ring setting deficiency for the main steam safety valves. The valves were sent to Wyle Laboratories for testing for determination of the proper ring settings. The tests were completed and the results are contained in Wyle Laboratories Report No. 47787-01 dated July 12, 1985.

The objectives of the tests were to:

1. Determine if the "as-shipped" ring settings of the valves would allow the required disc travel with minimum tailpipe backpressure.

2. Determine the effects on the valve disc travel for a range of backpressures between 180 and 390 psig.

The results of the "as-shipped" ring setting tests indicated that the valves could not achieve the required disc travel with 3% steam accumulation at minimum tailpipe pressures of 15-20 psig.

~~8509050195~~
3pp.

During the test, the upper (guide) ring setting was adjusted from +155 notches to 0 and +25 notches, and full required disc travel was achieved at 3% steam accumulation under the full range of tailpipe backpressure tested.

As a result of these test, we have agreed with Crosby, the valve manufacturer, that the optimum ring settings for the Seabrook main steam safety valves shall be -25 notches for the lower (nozzle) ring (original setting) and +25 notches for the upper (guide) ring. The corrections were completed by a Crosby service representative at the Wyle facility prior to returning the valves to the Seabrook Site.

This is our final report on this subject.

Very truly yours,

John DeVincentis

John DeVincentis, Director
Engineering and Licensing

cc: Atomic Safety and Licensing Board Service List

Director, Office of Inspection and Enforcement
U. S. Nuclear Regulatory Commission
Washington, DC 20555

William S. Jordan, III
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Harmon, Weiss & Jordan
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Suite 430
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Office of the Executive Legal Director
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Robert A. Backus, Esquire
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P.O. Box 516
Manchester, NH 03105

Philip Ahrens, Esquire
Assistant Attorney General
Augusta, ME 04333

Mr. John B. Tanzer
Designated Representative of
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Hampton, NH 03842

Roberta C. Pevear
Designated Representative of
the Town of Hampton Falls
Drinkwater Road
Hampton Falls, NH 03844

Mrs. Sandra Gavutis
Designated Representative of
the Town of Kensington
RFD 1
East Kingston, NH 03827

Jo Ann Shotwell, Esquire
Assistant Attorney General
Environmental Protection Bureau
Department of the Attorney General
One Ashburton Place, 19th Floor
Boston, MA 02108

Senator Gordon J. Humphrey
U.S. Senate
Washington, DC 20510
(ATTN: Tom Burack)

Diana P. Randall
70 Collins Street
Seabrook, NH 03874

Donald E. Chick
Town Manager
Town of Exeter
10 Front Street
Exeter, NH 03833

Brentwood Board of Selectmen
RED Dalton Road
Brentwood, NH 03833

Richard E. Sullivan, Mayor
City Hall
Newburyport, MA 01950

Calvin A. Canney
City Manager
City Hall
126 Daniel Street
Portsmouth, NH 03801

Dana Bisbee, Esquire
Assistant Attorney General
Office of the Attorney General
208 State House Annex
Concord, NH 03301

Anne Verge, Chairperson
Board of Selectmen
Town Hall
South Hampton, NH 03827

Patrick J. McKeon
Selectmen's Office
10 Central Road
Rye, NH 03870

Carole F. Kagan, Esquire
Atomic Safety and Licensing Board Panel
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Mr. Angi Machiros
Chairman of the Board of Selectmen
Town of Newbury
Newbury, MA 01950

Town Manager's Office
Town Hall - Friend Street
Amesbury, MA 01913

Senator Gordon J. Humphrey
1 Pillsbury Street
Concord, NH 03301
(ATTN: Herb Boynton)

PSNH

SEABROOK STATION
Engineering Office

Public Service of New Hampshire

New Hampshire Yankee Division

January 17, 1985

SBN- 751
T.F. Q2.2.2

United States Nuclear Regulatory Commission
Region I
631 Park Avenue
King of Prussia, PA 19406

Attention: Mr. Richard W. Starostecki, Director
Division of Project and Resident Programs

References: (a) Construction Permits CPPR-135 and CPPR-136, Docket Nos. 50-443 and 50-444
(b) Telecon of December 21, 1984, A. L. Legendre, Jr. (YAEC) to J. Grant (Region I)

Subject: Interim 10CFR50.55(e) Report, "Main Steam Safety Valve Ring Setting Deficiency"

Dear Sir:

We previously reported [Reference (b)] a potential 10CFR50.55(e) item to Region I regarding the apparent deficiency in Crosby Main Steam Safety Valve ring settings. The deficiency was discovered at Wyle Labs during full flow testing of the Crosby valves to determine the proper vent stack size. In order to determine full flow, the valve disc travel was measured and a disc travel of 1.12 inches was confirmed to us as being required by Crosby Valve Company via telecon on December 3, 1984. The tests determined that the disc only traveled approximately 50% of the required distance with 3% accumulation. The same limited disc travel occurred on later tests with larger diameter vent stacks. After discussions with Crosby, the ring setting was adjusted. The valve was retested and the required disc travel was achieved.

The problem and its reportability occurs because uncorrected reduced valve disc travel may invalidate certain assumptions on the Main Steam System in the FSAR Accident Analysis. The Seabrook FSAR Accident Analysis performed by Westinghouse Corporation assumes that the Main Steam Safety Valves are fully open with 3% accumulation. The ring settings are adjusted at Crosby Valve Company and the valve disc travel is not normally measured to confirm the required lift in field test.

8502080237

In conclusion, we are performing additional testing on the Crosby Main Steam Safety Valve to verify the ring setting problem without any stack influence whatsoever, and to determine the new ring settings. The results of these tests will be reported in a future interim 10CFR50.55(e) report by March 31, 1985.

done in may
at wyle

Very truly yours,

John DeVincentis

J. DeVincentis, Director
Engineering and Licensing

cc: Atomic Safety and Licensing Board Service List

Director, Office of Inspection and Enforcement
U. S. Nuclear Regulatory Commission
Washington, DC 20555

CALL M. WEGNER or
D. KIESSEL

RE: Seabrook MSSV's

LASALLE 1 [redacted]

GROUP 1 ISOL NICK

2 LIFTED

3300 GAL LOST EVERY LIFT

8-9/1ST HOUR

THEN 45 MIN

VALVES WORKED VERY GOOD - AS DESIGNED

RCIC PROVIDED SLOW REFILL

HPCS NOT USED - AFRAID OF OVERFILL

→ TELL KULIN TUES. WHEN HE CALLS

B/26

Seabrook MSSV [redacted]

Hal Gregg

50.55 e recd. by Reg. I

Memo to Baer from R-I

150 guide
(-)25 nozzle } 1.21" lift

Wyle verified w. Crosby

went to 0 guide → got full lift

Ring setting stamped on valve

utility owes Crosby info to evaluate

Tests were more than req'd.

How many valves tested ?

Discharge arrangement ?

Inlet arrangement ?

Bellows ?

B/27

U.S. NUCLEAR REGULATORY COMMISSION

OPERATOR/SENIOR OPERATOR LICENSING EXAMINATION
ANSWERS

2

Concerning three main steam relief valves that have been slow in reseating, the licensee has scheduled disassembly and rebuilding of the valves during the next refueling outage in early 1986. New nozzles are being ordered in case they are needed. The licensee stated that maintenance will be prepared and the work will be done prior to the refueling outage in the event of a Unit 1 shutdown requiring cooldown for a period of several days. (paragraph 10)

3. Licensee Action on Previous Inspection Findings

Not inspected

4. Unresolved Items

Unresolved items were not identified on this inspection

B/28

10 Unit 1 Main Steam Relief Valves (MSRV's)

Following four reactor trips in the past nine months, two Unit 1 MSRV's, MS 2 and MS 10, have failed to reseat properly until main steam header pressure was reduced to 850 to 900 psig. One such event on May 12, 1984 was discussed in Report No. 50-269/84-11 and LER 269/84-02. Events on December 2 and 3, 1984 were discussed in Report No. 50-269/84-32 and LER's 269/84-~~04~~ 06 + 07. IFI in 84-32 Following the trip on January 22, 1985, MS 2 and 10 ~~again~~ again failed to reseat properly until steam pressure was reduced to approximately 850 psig.

MSRV's 2 and 10 ~~which~~ have a nominal relief set point of 1065 psig. The lift set point of these valves was checked and reset prior to unit startup after refueling in November 1984. They were checked and reset again ~~during~~ following the trips in December using a modified procedure attempting to eliminate possible differences in technique by different mechanics. However, this does not appear to have eliminated the problem.

The MSRVs ~~they~~ relief, or trip, settings are checked and set with 900 psig pressure of steam boosted by a pneumatic device to reach the set point. The blowdown, or reseat, value ~~value~~ is set by adjustment of blowdown rings a given number of notches based on manufacturer's (Crosby) specifications. This is set on the bench before valves are installed or

Blowdown
4% ⇒ 43psi
7% ⇒ 75

actual
$\frac{1065-850}{1065} \times 100$ = 20%

$\frac{1065-900}{1065} \times 100$ = .15%

15-20%

reinstalled after maintenance. Ring settings are not changed with the valves on line due to personnel safety considerations and because there is no reliable means to check the reseat values.

Blowdown, according to the vendor manual, is 4% of relief point; however, the licensee stated that in reality it is about 7%. The licensee has checked with one vendor who checks blowdown settings but found that the accuracy claimed did not appear to be beneficial. Methods of checking blowdown are still under study by the licensee.

MSRV's 2, 4 and 10 have been scheduled for disassembly and rework on Unit 1's next refueling outage in early 1986. The resident inspectors have discussed this matter with licensee representatives on several occasions. Parts are on hand except for spare nozzles and they are being ordered. The licensee has stated that they will ~~p~~ be prepared to do the valve rework on any shutdown which requires plant cooldown for a period of several days, but that the plant will not be cooled down specifically for rework of the MSRVs.

To date the valves have always reseated by the time steam header pressure has been dropped to 850 psig, ~~the~~ and reactor cooldown has not ~~been~~ appeared to be excessive. The resident inspectors will stay abreast of developments

U.S. NUCLEAR REGULATORY COMMISSION
OPERATOR/SENIOR OPERATOR LICENSING EXAMINATION
ANSWERS

NAME

18

concerning the MSRV's and the inspector followup item opened in Report No 269/84-32 will remain open.

No violations or deviations were identified.

PSNH

SEABROOK STATION
Engineering Office

Public Service of New Hampshire

New Hampshire Yankee Division

January 17, 1985

SBN- 751
T.F. Q2.2.2

United States Nuclear Regulatory Commission
Region I
631 Park Avenue
King of Prussia, PA 19406

Attention: Mr. Richard W. Starostecki, Director
Division of Project and Resident Programs

References: (a) Construction Permits CPPR-135 and CPPR-136, Docket Nos. 50-443 and 50-444
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Subject: Interim 10CFR50.55(e) Report, "Main Steam Safety Valve Ring Setting Deficiency"

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~~8501220187~~
2PP

In conclusion, we are performing additional testing on the Crosby Main Steam Safety Valve to verify the ring setting problem without any stack influence whatsoever, and to determine the new ring settings. The results of these tests will be reported in a future interim 10CFR50.55(e) report by March 31, 1985.

Very truly yours,

John DeVincentis

J. DeVincentis, Director
Engineering and Licensing

cc: Atomic Safety and Licensing Board Service List

Director, Office of Inspection and Enforcement
U. S. Nuclear Regulatory Commission
Washington, DC 20555

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

[redacted]

MEMORANDUM FOR: Robert Bosnak, Chief
Engineering Issues Branch
Division of Safety Review and Oversight

FROM: Frank C. Cherny, Section Leader
Section B
Engineering Issues Branch

SUBJECT: TRIP REPORT - MEETING OF ASME SECTION III SUBGROUP ON PRESSURE RELIEF, MARCH 12 & 13, 1985

The referenced meeting was held at the United Engineering Center in New York City. A synopsis of major items of interest to NRC is contained in the attachment to this memorandum. Information on all the agenda items is available in my file.

Frank C. Cherny

Frank C. Cherny, Section Leader
Section B
Engineering Issues Branch

Attachment:
As stated

cc: T. Speis
B. Sheron
J. Richardson
W. Campbell
W. Norris
G. Millman
H. Gregg, RI
K. Kniel
G. Hammer
E. Brown
M. Wegner
M. Caruso
EIB Members

11pp

B/73

ASME CODE COMMITTEE: ASME Section III Subgroup
on Pressure Relief (SGPR)

DATE OF MEETING: March 12 and 13, 1985

NRC COMMITTEE MEMBER: F. Cherny

MEETING ATTENDED AND REPORTED BY: F. Cherny

I. ADMINISTRATIVE ITEM

As previously reported, F. Catudal, the SGPR Chairman is retiring by mid 1986 and resigning from the Subgroup. This was his last SGPR meeting. His last Code Committee meeting as SGPR Chairman will be the May S/C III meeting in Baltimore. The writer has received both NRC and ASME approvals to succeed as SGPR Chairman and will fully assume that position after the Baltimore meeting.

II. TECHNICAL ITEMS

A. Definition of System Operating Condition - NP-3-82

The proposal, prepared by the writer, to delete the "Upset, Emergency, etc. System Condition "terminology from NB-7000 was passed by the MC at the February meeting. The SGPR agreed that the same changes should be incorporated in NC/ND-7000 as part of the changes made pursuant to the ongoing re-review effort (NP-1-85).

B. Ring Adjustment Problems on ASME CL.2 Main Steam Safety Valves (MSSVS) NP-1-86

In accordance with agreements reached at the last SGPR meeting, the writer submitted a formal proposal at this meeting that would revise NC-7000 to require that each CL.2 MSSV production valve be full flow, full pressure tested by the manufacturer prior to shipment. These tests would confirm that the value adjusting rings were in the correct position to assure full stamped relieving capacity and that blowdown was in accordance with the valve specified in the valve Design Specification. With minor editorial changes the proposal was unanimously passed. It will be incorporated as part of the revisions made pursuant to the NC-7000 re-review (NP-1-85). As previously reported, Crosby Valve currently has the capability to perform such tests. Target Rock will have the capability in the near future. Dresser has stated that valves will have to be shipped to Wyle Laboratory in Huntsville, Alabama. As part of the discussion on this item, the writer specifically asked if there was anything less expensive than testing each valve that could be done to assure that the rings were reliably adjusted properly. None of the SGPR members had any alternative suggestions. Representatives from Crosby, Dresser, and Target Rock were in attendance. The Chairman asked what the bases for ring setting has been up to the present time for MSSVs. Only the Crosby representative responded, saying they were extrapolated based on data from much smaller valves and "field experience".

C. Re-Review of NB/NC/ND-7000 NP-1-85

Again for this meeting the review focused on NB-7000. Significant actions include:

1. Safety Valves - reaffirming restriction for use to only stem, air, or gas service. Even with the addition of the full size Demonstration of Function Text requirement, SGPR concluded that the Code still should not permit the use of this type of valve for liquid service.

2. Safety Valves With Auxilary Actuating Devices - Revised to permit use of this type of valve for air, gas and liquid service in addition to steam. This is consistent with EPRI and other test data obtained over the last several years.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT
WASHINGTON, D.C. 20555

IE INFORMATION NOTICE NO. ___ : ~~INADEQUATE CAPACITY OF~~ MAIN STEAM SAFETY VALVES ~~IN PWR PLANTS~~ TEST FAILURES AND RING SETTING ADJUSTMENTS

Addresses:

All PWR nuclear power reactor ~~facilities~~ holding an operating license (OL) or a construction permit (CP)

Purpose:

This information notice is being provided as a notification of a potentially significant problem pertaining to spring-actuated main steam safety valves (see Figure 1) ~~(MSSVs)~~, that may ~~not~~ possess less than the full rated flow ~~capability~~ capacity required for overpressure protection of the secondary cooling system in PWR s. It is expected that recipients will review the information for applicability to their facilities and consider actions, if appropriate, to preclude a similar problem ~~occurring~~ at their facilities. However, suggestions contained in this information notice do not constitute NRC requirements; therefore, no specific action or written response is required.

Description of Circumstances:

Between October 16, 1984 and ~~November~~ December 1, 1984, Wyle Laboratories conducted several full flow steam tests on two separate Main Steam Safety valves (MSSVs) manufactured by Crosby Valve and Gage

B/74

Company. These ~~two~~ Crosby 6R10 MSSVs are to be installed by Public Service of New Hampshire on the Seabrook main steam ~~Secondary cooling~~ system. The tests were conducted in order to determine the adequacy of various MSSV discharge piping arrangements. During the tests the valves were instrumented to measure valve disk lift. The valves were installed set up on the test facility with the Crosby recommended settings of the valve adjusting rings. With these factory ring settings the valve achieved ~~only~~ about 50% of the full disk lift required to develope full steam flow capacity within the ASME Code ~~required~~ 3% accumulated overpressure limit. Adequate lift was not achieved for either valve with the factory ring settings, (least flow resistance) even for the largest diameter vent pipe tested. ~~The rings were subsequently adju~~. The ~~middle~~ guide ring of both valves was subsequently adjusted a significant amount (150 notches) ~~in the~~ during the course of testing and full disk lift was subsequently achieved.

These types of full flow tests ~~are~~ normally not performed by either utility or the valve vendor on ~~the safety valves~~ MSSVs, nor are such tests required for capacity certification according to the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III. In general, these valves are capacity certified by ~~testing~~ tests on much smaller size valves, ~~which are~~ and the capacities then extrapolated to larger size valves. The MSSVs ~~of~~ on most PWRs, while not ~~the~~ necessarily the same model, are like those ~~on~~ at Seabrook in that these valves are generally at the upper end of the valve size range. This raises the concern that full flow functional demonstration of these type valves may never have been performed.

~~It was revealed during the industry sponsored EPRI Electric Power Research Institute testing of the PWR pressurizer safety valves that full flow capacity of spring-actuated safety valves can be is sensitive to valve ring settings.~~

A related MSSV problem which has occurred at several PWRs in the past few years pertains to excessive blowdown of main steam system pressure during transients which have actuated some MSSVs. On separate occasions at the Oconee, Salem, Trojan, and Davis-Besse nuclear facilities, MSSVs have remained open below the correct reseat pressure and have blown down excessive amounts of steam. The design blowdown value is usually 5% of setpoint pressure and is also dependent on the specific valve ring setting adjustments. Some valves at these facilities have exhibited as much as 10% blowdown. This raises the concern that some MSSVs

--- may remain open too long, relieving excessive quantities of steam possibly adversely affecting cooling of the primary system and causing excessive thermal stresses on ~~the~~ primary system components.

No specific action or written response is required by this information notice. If you have any questions about this matter, please contact the Regional Administrator of the appropriate regional office or this office.

Edward L. Jordan, Director
Division of Emergency Preparedness
and Engineering Response
Office of Inspection and Enforcement

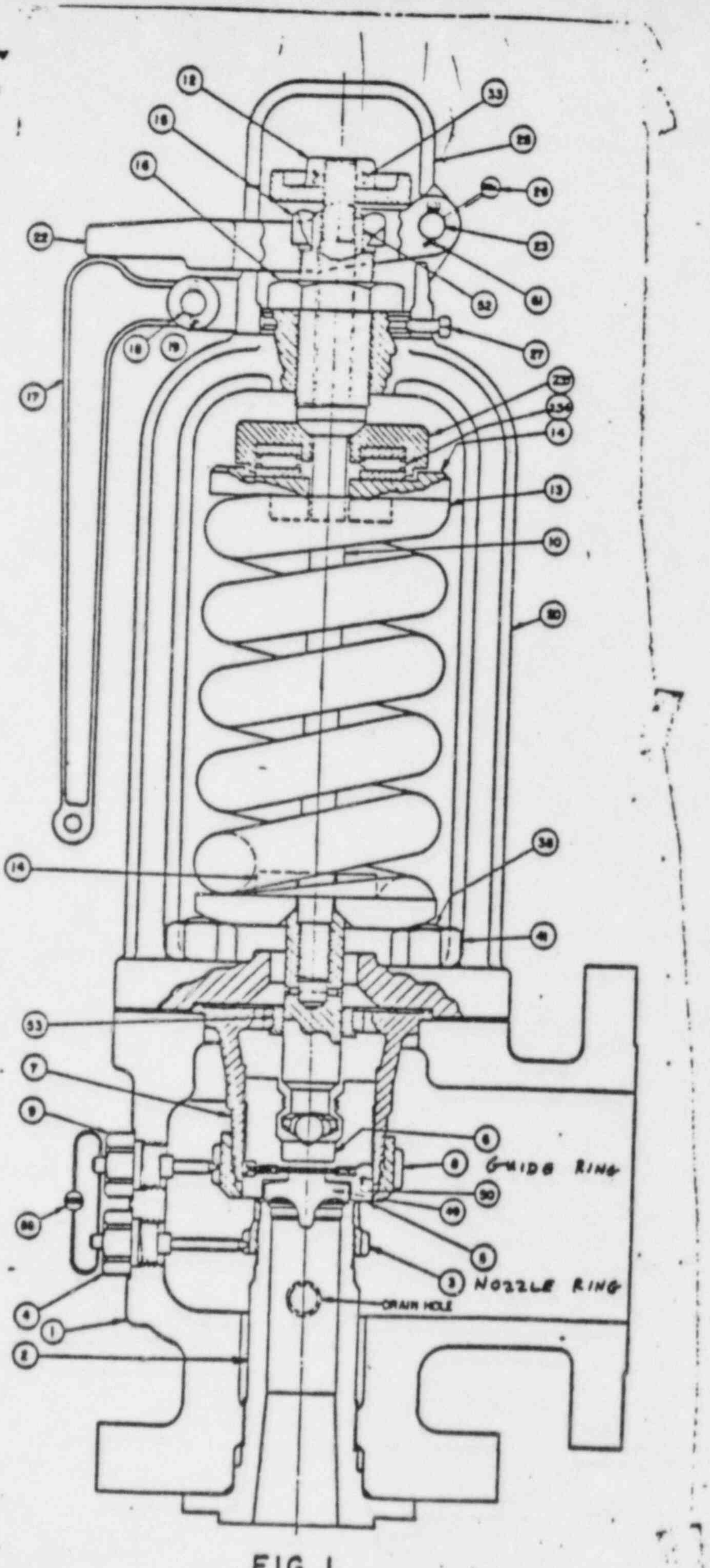


FIG. 1

TYPICAL MAIN STEAM SAFETY VALVE

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

[redacted]

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

FROM: Dennis M. Crutchfield, Assistant Director
for Safety Assessment, DL

SUBJECT: SUMMARY OF THE OPERATING REACTORS EVENTS
MEETING ON JULY 23, 1985 - MEETING 85-12

On July 23, 1985, an Operating Reactor Events meeting (85-11) was held to brief the Office Director, the Division Directors and their representatives on events which occurred since our last meeting on July 1, 1985. The list of attendees is included as Enclosure 1.

The events discussed and the significant elements of these events are presented in Enclosure 2. In addition, the assignment of follow-up review responsibility was discussed. The assignments made during this meeting and the status of previous assignments are presented in Enclosure 3.

Completion dates have been assigned for items in Enclosure 3. Note that we have revised Enclosure 3 to facilitate computerized tracking and provide additional details regarding responsibilities and status. Each assignee should review Enclosure 3 with regard to their respective responsibilities and advise ORAB if the target completion date cannot be met. If an assignee has any questions, please contact D. Tarnoff, x29526.

Dennis M. Crutchfield

Dennis M. Crutchfield, Assistant Director
for Safety Assessment, DL

Enclosures:
As stated

cc w/encl:
See next page

~~8508120333~~

3 pp.

B/40

cc: H. Denton
R. Bernero
J. Knight
T. Speis
C. Heltemes
T. Novak
W. Russell
J. Taylor
E. Jordan
F. Rowsome
W. Minners
L. Shao
T. Ippolito
S. Varga
J. Zwolinski
E. Sullivan
D. Beckham
G. Edison
K. Seyfrit
T. Murley, R-I
J. Nelson Grace, R-II
J. Kepper, R-III
R. D. Martin, R-IV
J. B. Martin, R-V
R. Starostecki, R-I
R. Walker, R-II
C. Norelius, R-III
R. Denise, R-IV
D. Kirsch, R-V
G. Lainas
Baranowski, RES
E. Rossi, IE
R. Hernan
F. Schroeder
G. Knighton
D. Silver
J. Lyons
D. Brinkman
E. Weiss
R. Baer
J. Stolz
E. Butcher
R. Bosnak
P. Morriette
W. Jones
G. Hammer
D. Osborne
R. Caruso
D. Lynch
D. McDonald
D. Neighbors
V. Nerses
H. Nicolaras
J. Wilson
P. O'Connor
T. Alexion
K. Jabbour
L. Olshan
H. Booher
B. Sheron
F. Cherney

ENCLOSURE 1

LIST OF ATTENDEES

OPERATING REACTORS EVENTS BRIEFING (85-11)

JULY 23, 1985

H. Denton, NRR
M. Caruso, NRR
J. Jackson, NRR/DE/EQB
B. Jones, IE/DEPER/EAB
F. Cherny, MEB/DE/NRR
W. Swenson, ORAB/DL/NRR
H. Nicolaras, NRR/DL/ORB#4
K. Mitchell, NRR/DL/ORAB/ORB#5
J. Stone, IE/VPB
G. Bagchi, NRR/DE/EQB
J. Wilson, LB#3/DL/NRR
T. Alexion, LB#1/DL/NRR
L. N. Olshan, LB#1/DL/NRR
P. O'Connor, LB#1/DL/NRR
W. J. Collins, IE/DEPER
N. P. Kadambi, NRR/DL/LB#3
D. Humenansky, OCM/COMM ZECH
T. Rotella, NRR/DL/ORB#5
B. Sheron, NRR/DSI/RSB
B. Bosnak, NRR/DE
A. W. Dromerick, IE/DEPER/EGCB
E. Weiss, IE/DEPER/EAB
N. Lauben, NRR/DSI/RSB

J. Stolz, NRR/DL/ORAB#4
D. Neighbors, NRR/DL/ORB#1
G. Knighton, NRR/DL/LB#3
H. Rood, NRC/DL/LB#3
J. Lyons, NRR/DL
R. Bernero, NRR/DSI
T. M. Novak, NRR/DL
D. Beckham, NRR/DHFS
R. Wessman, NRR/DL
T. Speis, NRR/DST
F. Schroeder, NRR/DST
W. Minners, NRR/DST
S. Varga, DL
D. Crutchfield, DL
J. P. Knight, NRR/DE
G. Lanik, IE/EAB
S. Schwartz, IE/DEPER
H. Thompson, NRR/DL
P. Morriette, NRR/DL
D. Tarnoff, NRR/DL
K. Seyfrit, AEOD/ROAB
D. Zukor, AEOD/ROAB
E. J. Brown, AEOD/ROAB

OPERATING REACTORS EVENTS BRIEFING (85-12)

JULY 23, 1985

INDIAN POINT UNIT 3	-	STEAM GENERATOR WELD INDICATIONS
SEABROOK	-	MAIN STEAM SAFETY VALVE TEST FAILURE
OCONEE UNIT 2	-	EXTENDED BLOW DOWN FROM MAIN STEAM SAFETY VALVES
WATERFORD UNIT 3	-	PLANT TRIPS JULY 4-7, 1985
WATERFORD/WOLFCREEK CALLAWAY/CATAWBA/BYRON	-	STARTUP EXPERIENCE COMPARISON.
COMBUSTION ENGINEERING		LOCA ANALYSIS ERROR
MOJAVE GENERATING STATION	-	REHEAT LINE FAILURE
PALUEL UNITS 1, 2	-	IN-CORE INSTRUMENTATION TUBE VIBRATION PROBLEMS

OTHER EVENTS OF INTEREST

MILLSTONE UNIT 2	-	PRESSURIZER SPRAY VALVE FAILURES
LASALLE UNIT 1	-	RHR FLOW SWITCHES IMPROPERLY INSTALLED
FERMI UNIT 2	-	INADVERTENT CRITICALITY
TURKEY POINT UNIT 3	-	REACTOR TRIP AND AFW VALVE FAILURE

INDIAN POINT 3 - SG WELD INDICATIONS

JUNE 27, 1985 (D. NEIGHBORS, NRR)

- PLANT IN REFUELING STATUS
- T.S. REQUIRES INSPECTIONS OF SG TRANSITION ZONE UPPER GIRTH WELDS
- INDICATIONS FOUND BY UT:

  SG 31 - 1
  SG 32 - 2
  SG 33 - 0
  SG 34 - 23

- SG-34 HAD WELD REPAIR OF LEAK FOUND IN 1983
- MT ON SG-34 SHOWED CLEAN ON 16 OF 23 INDICATIONS
- REMAINING 7 WELD INDICATIONS ON 34, AND 3 ON 31 AND 32 WERE FOUND TO BE CODE ACCEPTABLE BY VIRTUE OF SIZE OR BY FRACTURE MECHANICS ANALYSIS
- LICENSEE STILL EVALUATING
- NRR HAS LEAD (SINCE 7/15/85)
- NRR & IE DEVELOPING GENERIC CORRESPONDENCE

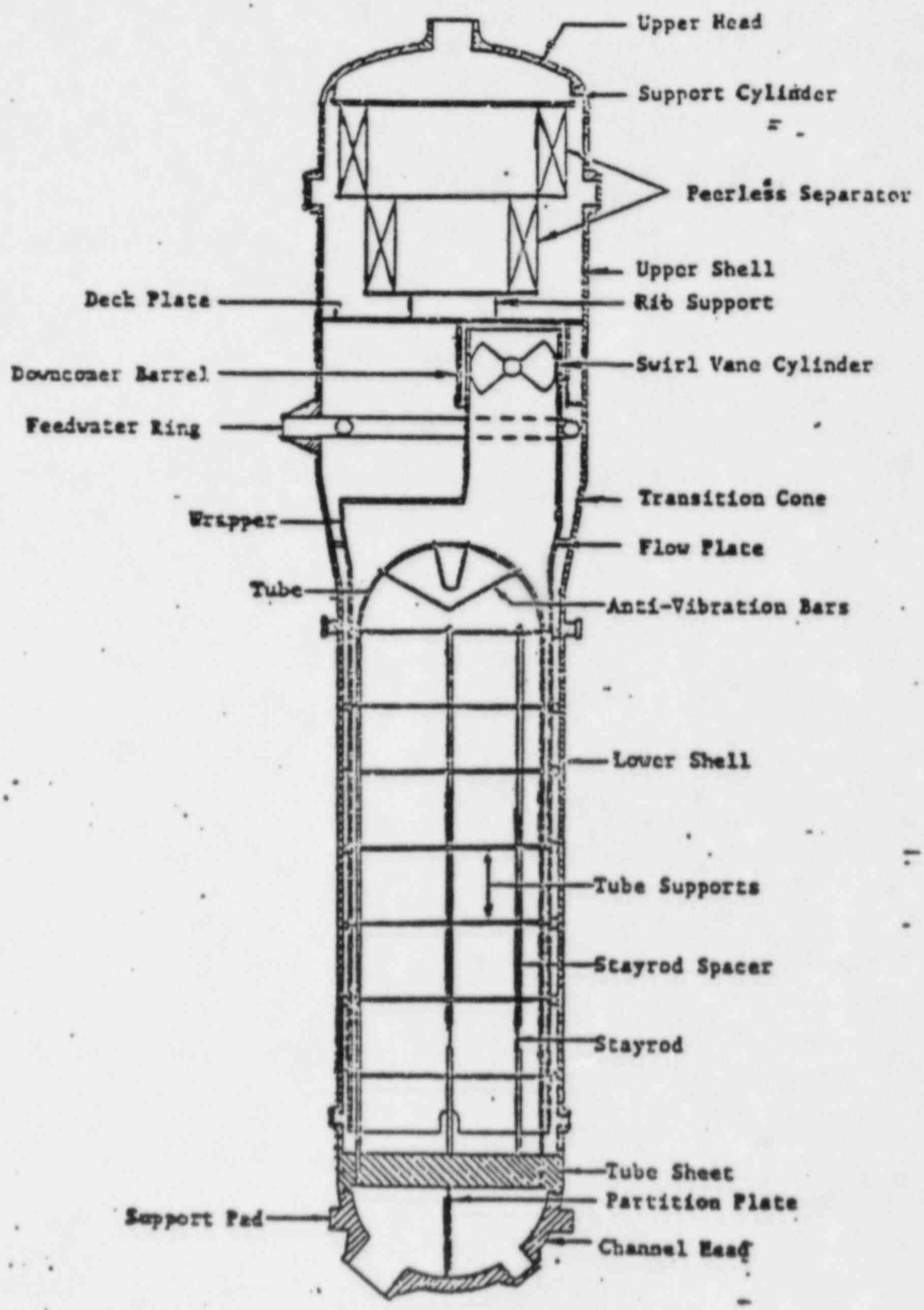


FIGURE 2.3-1
SERIES 51 STEAM GENERATOR

SEABROOK - CROSBY MAIN STEAM SAFETY VALVE FLOW DEFICIENCY - DECEMBER 1984

(G. HAMMER, NRR)

- PROBLEM - FULL FLOW TEST RESULTS INDICATE SPRING-ACTUATED MAIN STEAM SAFETY VALVES MAY NOT ACHIEVE RATED FLOW CAPACITY.

- SAFETY SIGNIFICANCE - POSSIBLE INADEQUATE OVERPRESSURE PROTECTION OF SECONDARY SYSTEM IN PWRs USING THESE VALVES

- WYLE LAB TEST RESULTS: INADEQUATE LIFT OF VALVE DISK (ABOUT 50%) WITH THE VENDOR (CROSBY) RECOMMENDED RING SETTING ADJUSTMENTS. TESTS WERE CONDUCTED TO DETERMINE ADEQUACY OF DISCHARGE PIPING.

- CORRECTIVE ACTION - RINGS READJUSTED. OBTAINED FULL LIFT ON SEABROOK VALVES

- GENERIC IMPLICATION - SEABROOK VALVES AND DISCHARGE PIPING SIMILAR TO OTHER PWRs. FULL FLOW TESTS NOT NORMALLY RUN TO ADJUST RINGS.

- NRC FOLLOWUP ACTION: -
  (1) DEVELOPING IE INFORMATION NOTICE
  (2) STAFF MAY PURSUE AS A GENERIC ISSUE
  (3) DISCUSSIONS WITH CROSBY BY REGION 1 AND NRR REGARDING ADEQUACY OF VENDOR GUIDANCE AND SRV RING SETTINGS.

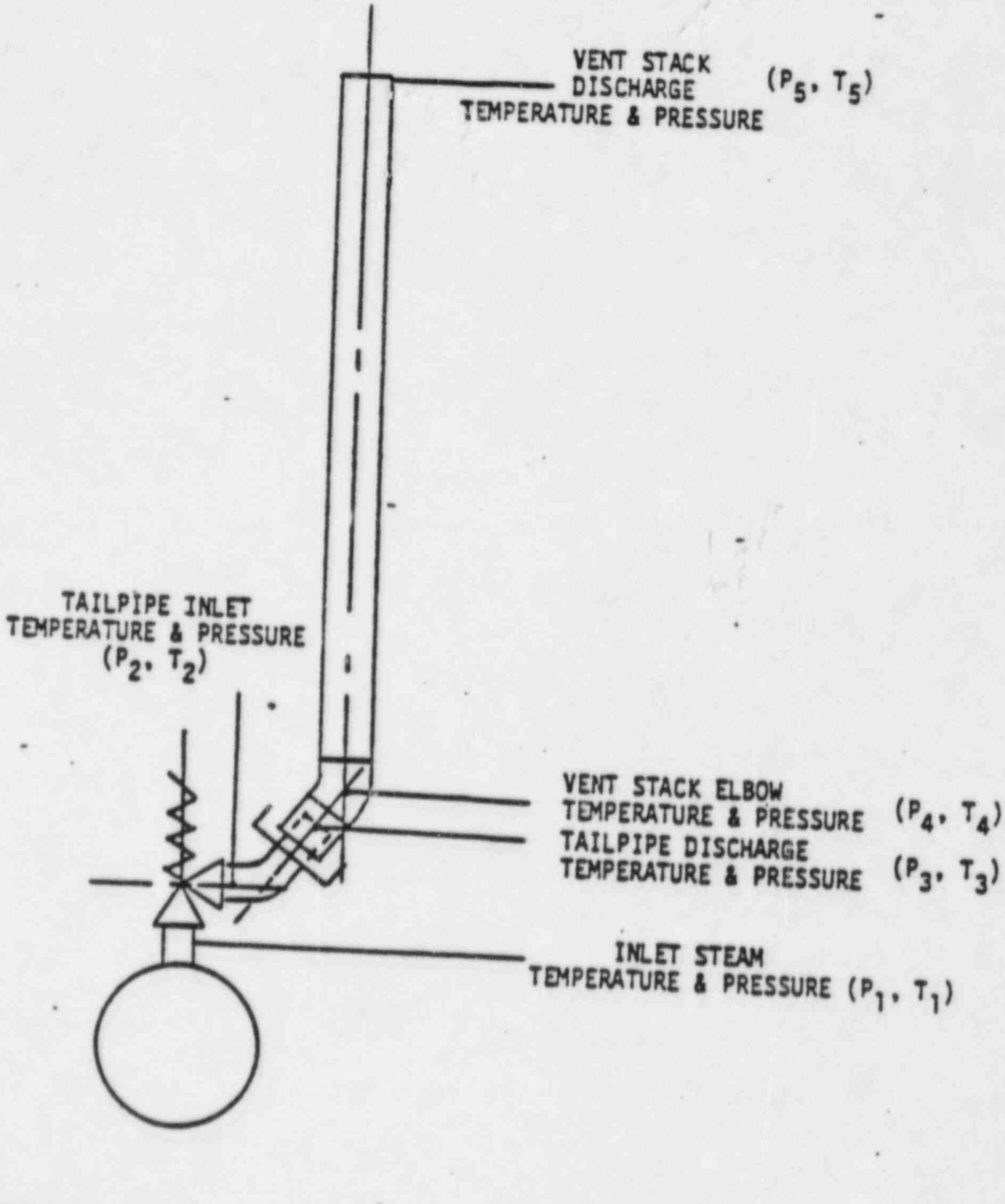


FIGURE 1. INSTRUMENTATION LOCATIONS (16, 18, AND 20-INCH VENT STACKS)

OCONEE 2 - EXTENDED BLOWDOWN FROM MAIN STEAM SAFETY VALVES

JULY 11, 1985 (H. NICOLARAS, NRR)

- OCONEE UNIT 2 REACTOR TRIP FROM 94% POWER CAUSED BY PERSONNEL ERROR
- TWO MAIN STEAM SAFETY VALVES DID NOT RESEAT AT SETPOINT - EXTENDED BLOWDOWN - TO ABOUT 990 PSI
- TO RESEAT VALVES, OPERATORS REDUCED STEAM PRESSURE THROUGH TURBINE BYPASS VALVES.
- FAILURE OF CROSBY MAIN STEAM SAFETY VALVES TO PROPERLY RESEAT HAS ALSO OCCURRED REPEATEDLY AT OCONEE UNIT 1
- IMPROPER RING SETTING IS A LIKELY CAUSE OF EXCESS BLOWDOWN, BUT NOT CONFIRMED.
- DUKE POWER COMMITTED CORRECTIVE ACTIONS TO REGION II
- SUMMARY OF PLANTS REPORTING SIMILAR BLOWDOWN PROBLEM

PLANT	KNOWN # OF EVENTS
OCONEE 1, 2, 3	31
TROJAN	1
SALEM	1

CAUSE: UNKNOWN

POSTULATED PROBLEM WITH RING SETTINGS

CORRECTIVE ACTIONS: READJUST, RESET SETPOINT, VALVE DISASSEMBLY

4 (OUT OF 16) VALVES REWORKED DURING EACH REFUELING

LICENSEE PURSUING METHODS FOR CHECKING BLOWDOWN SETTINGS

IMPLICATIONS: RCS OVERCOOLING

CHALLENGE TO ADDITIONAL SYSTEMS

POTENTIAL PRECURSOR TO STUCK OPEN VALVE OR FULL LIFT CAPACITY

WATERFORD 3 - PLANT TRIPS JULY 4-7, 1985

(J. WILSON, NRR)

- WATERFORD 3 EXPERIENCED FOUR REACTOR TRIPS IN LESS THAN THREE DAYS

- DURING A PORTION OF THIS TIME, THE EFW TURBINE-DRIVEN PUMP WAS UNAVAILABLE DUE TO BEING INADVERTENTLY TRIPPED

- JULY 4 AT 0950 HOURS - 100% PWR - LOW SG LEVEL-HIGH VIBRATION ON "A" MAIN FEEDWATER PUMP

- JULY 4 AT 2217 HOURS - 6% PWR - CPC AUXILIARY TRIP ON AXIAL SHAPE INDEX - XE OSCILLATIONS

- JULY 5 AT 2219 HOURS - 60% PWR - HIGH SG LEVEL DUE TO OVERFEEDING SG WHILE IN MANUAL CONTROL WITH ONE MAIN FEEDWATER PUMP RUNNING

- JULY 6 AT 0915 HOURS - 70% PWR - EFW PUMP TERRY TURBINE OVERSPEED LATCH WAS FOUND TO BE TRIPPED

- JULY 7 AT 0121 HOURS - 90% PWR - LOW SG LEVEL - LOSS OF MAIN FEEDWATER PUMPS ON LOW SUCTION WHILE AN OPERATOR WAS ATTEMPTING TO BACKWASH A CONDENSATE POLISHING SYSTEM FILTER

- LP&L CORRECTIVE ACTIONS:
    - REMOVING TRIP ON MAIN FEEDWATER PUMP VIBRATION--ALARM ONLY
    - REVISING OPERATING PROCEDURES
    - TRAINING, NIGHT ORDERS
- REGION IV AND IE MONITORING STARTUP ACTIVITIES

WATERFORD - 3, WOLF CREEK, CALLAWAY, CATAWBA AND BYRON - STARTUP EXPERIENCE COMPARISON (W. JONES, IE)

FACILITY	FULL POWER LICENSE DATE
WATERFORD - 3	(03-16-85)
WOLF CREEK	(06-04-85)
CALLAWAY	(10-18-84)
CATAWBA	(01-17-85)
BYRON	(02-14-85)

- BASED ONLY ON 50.72 NOTIFICATIONS-- SIGNIFICANT EVENTS ONLY (I.E. NOT CONTROL ROOM ISOLATIONS ETC)

- FOR 4 MONTHS FOLLOWING FULL POWER LICENSE (ADJUSTED FOR SIGNIFICANT SHUTDOWNS)

REPORTED EVENT COMPARISON SUMMARY

	MONTH			
	1ST	2ND	3RD	4TH
WATERFORD	4	2	2	9
WOLF CREEK	9	--	--	--
CALLAWAY	11	2	3	1
BYRON	7	3	2	3
CATAWBA	5	0	1	5*
MONTHLY AVG W/O WATERFORD & WOLF CREEK	7.6	1.6	2	3

*1 MONTH SHUTDOWN NOT INCLUDED IN PERIOD

REACTOR TRIP AND FEEDWATER COMPARISON DETAIL

	MONTH	SIGNIFICANT EVENT REPORTS	REACTOR+ TRIPS	INVOLVING LOFW
WATERFORD 3	1	4	4	3
	2	2	2	0
	3	2	2	1
	4	9	8	3
	SUBTOTAL	17	16	7
WOLF CREEK	1	9	5	8
CALLAWAY	1	11	6	8
	2	2	2	0
	3	3	3	2
	4	1	1	0
		17	12	10
CATAWBA	1	5	3	1
	2	0	0	0
	3	1	1	1
	4	5*	2	1
		11	6	3
BYRON	1	7	6	2
	2	3	3	0
	3	2	1	1
	4	3	3	0
		15	13	3

*MAY REFLECT RETURN TO POWER AFTER SHUTDOWN

+BASED ON AEOD & NRR STUDY FOR 1983/84, AVERAGE NUMBER OF TRIPS FOR 4 MONTH PERIOD WAS 2.1, FOR ALL PWR's.

PERSONNEL ERROR & EQUIPMENT PROBLEMS COMPARISON DETAIL

	MONTH	SIGNIFICANT EVENTS	PERS ERROR	EQP PROB	OTHER
WATERFORD 3	1	4	3	1	0
	2	2	1	1	0
	3	2	1	1	0
	4	9	2	6	1
		17	7	9	1
WOLF CREEK	1	9	3	5	1
		9	3	5	1
CALLAWAY	1	11	2	7	2
	2	2	0	1	1
	3	3	0	3	0
	4	1	0	1	0
		17	2	12	3
CATAWBA	1	5	3	1	1
	2	0	0	0	0
	3	1	1	0	0
	4	5	2	1	2
		11	6	2	3
BYRON	1	7	2	4	1
	2	3	2	0	1
	3	2	0	1	1
	4	3	0	3	0
		15	4	8	3

CE LOCA ANALYSIS ERROR

JULY 2, 1985 (H. ROOD, NRR)

- NON-CONSERVATIVE ERROR FOUND IN CE LARGE-BREAK LOCA MODEL
- CENTER PEAK AXIAL POWER SHAPE YIELDS 34°F HIGHER PEAK CLAD TEMPERATURE (PCT) THAN PREVIOUSLY ASSUMED TOP-PEAKED SHAPE.
- FOR THREE CE PLANTS THAT ARE ON 1ST CYCLE THIS WOULD YIELD A PCT IN EXCESS OF THE 2200°F LIMIT OF 10 CFR 50.46. PLANTS ARE:

  PALO VERDE 1
  SAN ONOFRE 3
  WATERFORD 3

- BASED ON CE REANALYSIS, OTHER FACTORS IN LARGE-BREAK LOCA MODEL WILL REDUCE PCT TO BELOW 2200°F.
- LETTERS FROM THESE 3 LICENSEES BEING SUBMITTED GIVING BASIS FOR CONTINUED OPERATION.
- OTHER CE LICENSEES BEYOND CYCLE 1 AND (EVEN WHEN OTHER FACTORS NOT INCLUDED) HIGHER PCT DOES NOT REACH 2200°F LIMIT.

MOHAVE GENERATING STATION - REHEAT LINE FAILURE

JUNE 9, 1985 (R. BOSNAK, NRR)

- FAILURE OCCURRED JUNE 9, 1985 WHEN A 30" REHEAT LINE SUDDENLY SPLIT LONGITUDINALLY

- FRACTURE WAS FISH MOUTH RUPTURE APPROXIMATELY 20' x 6' FIG 1A &B

- SAFETY SIGNIFICANCE

    - FOSSIL PLANTS OF SIMILAR VINTAGE
    - NUCLEAR PLANTS

REHEAT LINE - VITAL STATISTICS

- DESIGNED TO B31.1 CODE FOR STEAM CONDITIONS OF 1000°F AND 600 PSIG

- COMMENCED OPERATION 1971 CONSTRUCTION LATE 1960's

- FAILURE IN A HORIZONTAL SPOOL 30"-DIAMETER ROLLED AND WELDED OF A-378C PLATE (1 1/4 CR-1/2 MO) TO MEET A-155 WELDED PIPE

COMPARISON WITH LWR PIPING

- MATERIAL NOT USUALLY USED IN LWR

- UPPER TEMPERATURE NOT IN CREEP RUPTURE AND CREEP FATIGUE RANGE IN LWR

- FABRICATION CONTROLS INCLUDING NDE SUPERIOR IN LWR

- LEAK DETECTION REQUIREMENTS IN LWR

- INSERVICE INSPECTION IN LWR

FAILURE ANALYSIS

- RESULTS EXPECTED FROM SCE BY EARLY AUGUST

PALUEL 1 & 2, IN-CORE INSTRUMENTATION TUBE VIBRATION PROBLEMS

MARCH 29, 1985 (P. MORIETTE, NRR)

- INITIAL EVENT: MARCH 29, 1985, PALUEL 1 IN COLD SHUTDOWN.
- LEAK DETECTED ON ONE THIMBLE TUBE, WHILE LEAK TESTING IN-CORE INSTRUMENTATION SYSTEM.
- SUBSEQUENT FINDINGS:

  APRIL 5: MECHANICAL WEAR (WITHOUT LEAK) ON 4 OTHER THIMBLES.

  APRIL 16: A PROBE CANNOT BE COMPLETELY INSERTED IN ONE THIMBLE (PALUEL 1).

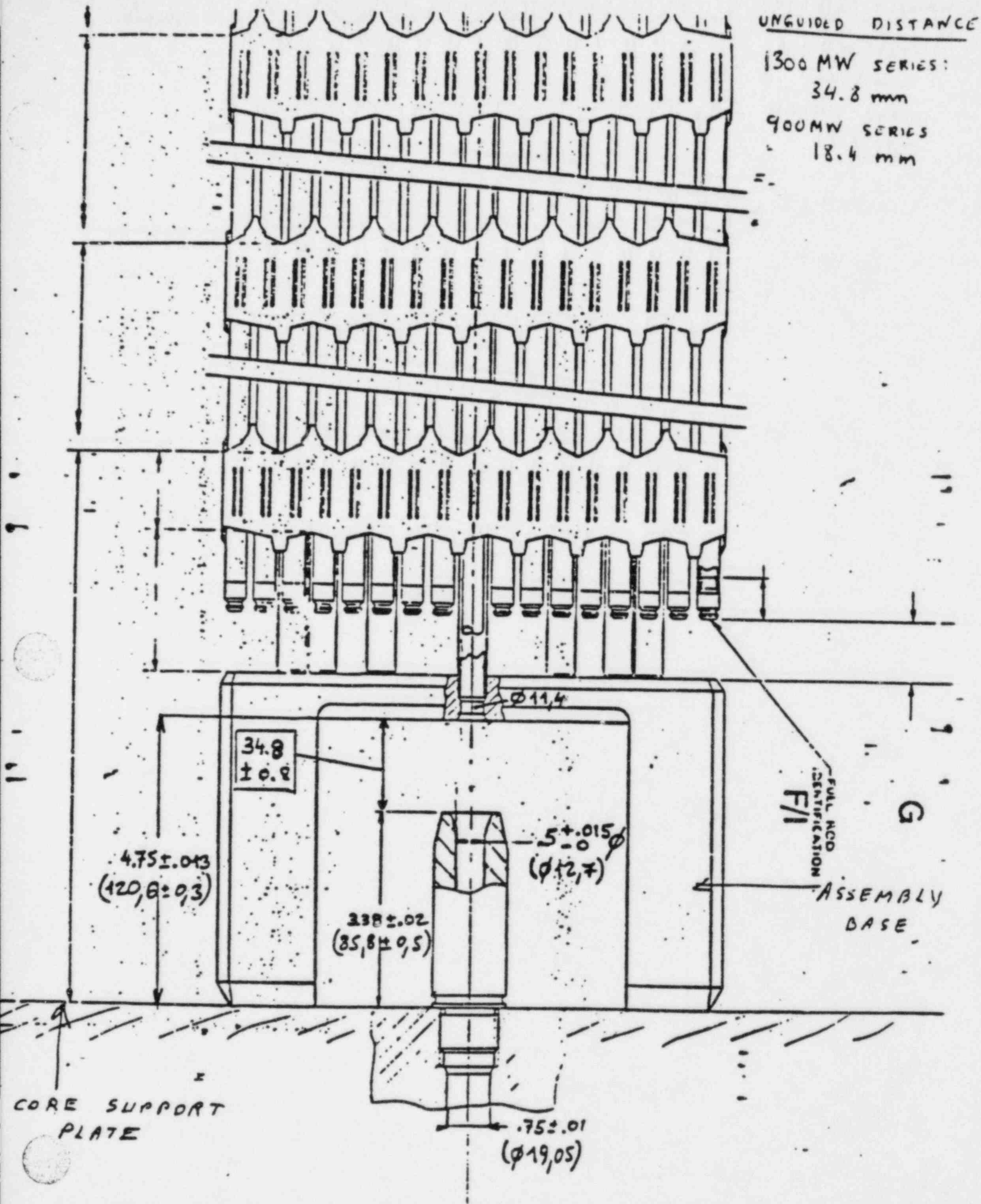
  MAY-JUNE: 2 LEAKS ON PALUEL 2, ANOTHER LEAK ON PALUEL 1
- SAFETY SIGNIFICANCE: REACTOR COOLANT LEAKS, OR: NO FLUX MAPS. POSSIBILITY OF MIGRANT OBJECTS.
- MAJOR POINTS:
  - DEFECTS (OR LEAKS) LOCATED AT DISCONTINUITY IN GUIDING STRUCTURE
  - CAUSE THOUGHT TO BE HYDRAULIC EXCITATION DUE TO TURBULENCE IN THE CORE SUPPORT PLATE - BOTTOM OF FUEL ASSEMBLY REGION.
  - DIFFERENCES (FROM 900MWE SERIES) IN LOWER INTERNALS DESIGN AND MEASURED FLOW PARAMETERS SUPPORT THIS HYPOTHESIS.
  - LOWER INTERNALS W DESIGN. CORE INSTRUMENTATION SYSTEM (OUTSIDE VESSEL) FRAMATOME DESIGN.
- GENERIC IMPLICATIONS: ALL 1300MWE SERIES REACTORS AFFECTED IN FRANCE

- LICENSEE CORRECTIVE ACTIONS:

  SHORT TERM: JUSTIFY OPERATION WITHOUT IN-CORE INSTRUMENTATION FOR 1 1/2 MONTH.

  LONG TERM: MODIFY THIMBLE GUIDING PIECES ON TOP OF CORE SUPPORT PLATE FOR BETTER PROTECTION, REDUCE TURBULENT FLOW AROUND THIMBLES.

- ONLY AFFECTED US FACILITY: SOUTH TEXAS PROJECT 1 & 2


UNGUIDED DISTANCE
1300 MW SERIES:
34.8 mm
900MW SERIES
18.4 mm
Ø11,4
34.8
± 0.8
4.75±.013
(120,6±0,3)
.5 +.015 -0 Ø
(Ø12,7)
.338±.02
(85,8±0,5)
FULL RCC IDENTIFICATION
F/I
G
ASSEMBLY
BASE
CORE SUPPORT
PLATE
.75±.01
(Ø19,05)

MILLSTONE UNIT 2 - FAILURE OF PRESSURIZER SPRAY VALVES TO SHUT

JULY 15, 1985 (D. OSBORNE, NRR)

- RETURNED TO 100% POWER ON 7/11/85 AFTER 135 DAY REFUELING OUTAGE
- REACTOR TRIPPED ON JULY 15, 1985 ON THERMAL MARGIN/LOW PRESSURE SETPOINT (2130 PSI)
- PRESSURE DROP HALTED AT 1725 PSI
- TRIP ATTRIBUTED TO FAILURE OF BOTH PRESSURIZER SPRAY VALVES TO SHUT
- FISHER CONTROL AIR OPERATED 3 INCH ANGLE VALVES
- ONE MISADJUSTED - (WOULD NOT FULLY CLOSE)
- OTHER HAS MECHANICAL BINDING (PRELIMINARY)
- DURING INVESTIGATION DISCOVERED SPRAY VALVES WIRED TO WRONG CONTROLLER, AS SPECIFIED IN THE OUTAGE DESIGN CHANGE (NOT CAUSE OF FAILURE)
- LICENSEE HAS REVIEWED OTHER REFUELING OUTAGE DESIGN CHANGES AND TESTING
- CORRECTIVE ACTIONS: ONE VALVE WAS READJUSTED TO FULLY CLOSE
OTHER VALVE WAS REPACKED AND REBUILT
- REGION REVIEWED PRIOR TO STARTUP
- PLANT RESTARTED ON JULY 20, 1985

FERMI 2 - INADVERTENT CRITICALITY

JULY 2, 1985 (D. LYNCH, NRR)

- PROBLEM - INADVERTENT CRITICALITY JULY 2, 1985, DUE TO OPERATOR ERROR
- SAFETY SIGNIFICANCE -

  PLANT WAS IN ANALYZED CONDITION AND NEVER EXCEEDED SOURCE RANGE.
- REACTOR OPERATOR PULLED ROD BANK (ABOUT 8 TO 10 RODS) FULL OUT (NOTCH 48) RATHER THAN NOTCH 4, AS SPECIFIED.
- JULY 2 - INADVERTENT CRITICALITY DECLARED BY OPERATOR AND RODS REINSERTED. SHIFT SUPERVISOR REVIEWED OPERATOR ACTIONS AND DETERMINED THAT INADVERTENT CRITICALITY HAD NOT OCCURRED. NORMAL STARTUP COMPLETED.
- POSSIBLE LACK OF EFFECTIVE ASSISTANCE OR OBSERVATION BY SRO
- SHIFT TECHNICAL ADVISOR WAS IN TRAINING
- SUBSEQUENT SEQUENCE OF EVENTS:

  JULY 7 - LICENSEE TECHNICAL STAFF CONFIRMED INADVERTENT CRITICALITY

  JULY 8 - INADVERTENT CRITICALITY RECONFIRMED INDEPENDENTLY BY LICENSEE

  JULY 10 - COMMISSION BRIEFING ON FULL POWER LICENSE

  JULY 15 - REGION III NOTIFIED BY LICENSEE (1:30 PM CDT)

  JULY 15 - FULL POWER LICENSE ISSUED BY NRC (3:15 PM EDT)
- REGION III ISSUED CONFIRMATORY ACTION LETTER LIMITING FERMI 2 TO 5 PERCENT POWER
- ENFORCEMENT CONFERENCE SCHEDULED FOR JULY 23
- INVESTIGATION CONTINUING

LA SALLE UNIT 1 - RHR FLOW SWITCHES IMPROPERLY INSTALLED

JULY 17, 1985 (R. CARUSO, NRR)

- PROBLEM - FOUR RHR FLOW ISOLATION SWITCHES FOUND TO BE PIPED BACKWARDS SINCE MARCH 1985 EQ UPGRADE, AS A RESULT OF BREAKDOWN IN MANAGEMENT CONTROL AND QA

- EVENT IS A REPEAT OF PROBLEM WITH UNIT 2 ECCS AND RHR SWITCHES IDENTIFIED ON 6/10/85

- SAFETY SIGNIFICANCE - RHR SWITCHES SERVE DIVERSE AND REDUNDANT FUNCTION TO ISOLATE RHR IN CASE OF BREAK - MINOR SIGNIFICANCE.

- LICENSEE WAITED FROM JUNE 10 TO JULY 17 TO TEST INSTALLATION, SINCE PLANT SHUTDOWN FOR UNRELATED ACTIVITY.

- CORRECTIVE ACTION - REGION III REVIEWING LICENSEE PLANS TO REVIEW MODIFICATION CONTROL SYSTEM. DRAWING CHANGES TO BE RE-REVIEWED, BOTH UNITS TO REMAIN SHUTDOWN UNTIL REVIEW COMPLETE.

- ESCALATED ENFORCEMENT ACTION BEING CONSIDERED.

UNPLANNED REACTOR TRIPS*

- AVERAGE WEEKLY TRIP FREQUENCY FOR PAST 7½ WEEKS IS APPROXIMATELY 10 TRIPS/WEEK, WHICH IS NEAR AVERAGE

- BREAKDOWN OF REPORTED CAUSES

  AUTOMATIC

  - EQUIPMENT FAILURES 52%
  - PERSONNEL ACTIVITIES 40%

  MANUAL 8%

*BASED ON 10 CFR 50.72 REPORTS FOR PLANTS WITH LICENSES FOR FULL POWER OPERATION

Enclosure 3

OPERATING REACTORS' EVENTS MEETING FOLLOWUP ITEMS
AS OF MEETING 85-12 ON JULY 23, 1985

(ORDERED BY ASCENDING MEETING DATES, NSSS VENDORS, FACILITY)

MTG NUMBER/ MEETING DATE	FACILITY NSSS VENDOR/ EVENT DESCRIP.	RESPONSIBLE DIVISION/ INDIVIDUAL	TASK DESCRIPTION	SCHEDULE COMPLET. DATE(S)	CLOSED DATE BY DOCUMENT/ MEETING, ETC.	COMMENTS
04/10/84	RANCHO SECO 1 BW / MAIN GEN HYDROGEN EXPLOSION - PARTIAL LOSS OF NNI 3/19/84	DL /VISSING /	SUMMARIZE B&W LIC. RESPONSES TO QUESTIONS SUBSEQUENT TO RANCHO SECO LOSS OF NNI EVENT AND PRESENT AT FOLLOW-UP OR EVENTS BRIEFING	08/30/85 07/30/85 / /	OPEN / /	STAFF REVIEW OF B&W OWNERS GROUP SUBMITTAL OF 1/11/85 IS IN PROGRESS (THIS REACTIVATED- CRYS.RIV. 3)
08/07/84	SALEM 2 W / STUCK OPEN RELIEF VALVE/ ECCS ACTUATION JULY 25, 1984	DL /FISCHER, D. /	DETERMINE IF VELAN (PORV) BLOCK VALVE QUALIFIED TO CLOSE AGAINST 7/25/84 STEAM BLOWDOWN TRANSIENT AT SALEM 2. CHECK EPRI TEST PROGRAM RESULTS.	08/30/85 07/30/85 / /	OPEN / /	AEOD REPORT IN PREPARATION WILL ADDRESS THAT ISSUE.
01/03/85	CRYSTAL RIVER 3 BW / TEMP. LOSS OF NNI 12/28/84	ICSB/ROSA, F. /	CONSIDER NEED FOR ADDITIONAL REQUIREMENTS ON ALARMS / ANNUNCIATORS	09/30/85 07/30/85 / /	OPEN / /	ICSB IS CONSIDERING OCONEE 1 LOSS OF ANNUNCIA. (4/25/84) IN ANAL. OF REQUIREMENTS.
05/07/85	CATAWBA 2 W / BOTH RHR TRAINS OVERPRESSURIZED 4/19/85	DL /JABBOUR, K. /	FOLLOW UP BRIEFING AFTER EVAL. OF OVERPRESS. EFFECTS ON VARIOUS SYSTEMS AND CORRECTIVE ACTIONS.	09/30/85 08/01/85 08/07/85	OPEN / /	PRELIM. REPORT REC'D. ADD'L INFO. REQUESTED FROM LICENSEE. RESOLUTION PENDING REVIEW BY NRR & RII.
05/07/85	MILLSTONE 2 CE / EDDY CURRENT TEST OF STEAM GEN. TUBES, 4/10/85	DRAB/MURPHY, E. DE /CONRAD, H.	REVIEW SGOG LETTER ON EDDY CURRENT TESTING ISSUE, IN VIEW OF MILLSTONE 2 FINDINGS.	10/01/85 08/01/85 08/07/85	OPEN / /	ORNL CONTRACTED TO EVALUATE EFFECT OF COPPER ON EDDY CURRENT TESTING
85-10 06/12/85	DAVIS BESSE 1 BW / LOSS OF ALL MAIN AND AUX. FEEDWATER 6/9/85	DST /SPEIS, T. DL /DEAGAZIO, A.	DETERMINE STATUS OF IMPLEMENTATION OF TMI ITEMS, GENERIC ISSUES, MPAs AT DAVIS BESSE THAT MAY BE RELATED TO 6/9/85 EVENT.	08/30/85 06/19/85 / /	OPEN / / STATUS OF IMPLEMENT. OF TMI ISSUES CLOSED- H. THOMPSON TO H. DENTON MEMO OF 6/20/85.	REMAINING 2 ISSUES (GENERIC & MPAs) IN PROGRESS. IIT REPORT UNDER REVIEW
85-1[illegible] [illegible]	DAVIS BESSE 1 BW / LOSS OF ALL MAIN AND AUX. FEEDWATER 6/9/85	DSI /PARR, O. DL /DEAGAZIO, A.	RE-EXAMINE STAFF REVIEW OF ACCEPTABILITY/ DIVERSITY OF DAVIS BESSE AFW SYSTEM.	08/30/85 06/19/85 / /	OPEN / /	IIT REPORT UNDER REVIEW

OPERATING REACTORs' EVENTS MEETING FOLLOWUP ITEMS
AS OF MEETING 85-12 ON JULY 23, 1985

(ORDERED BY ASCENDING MEETING DATES, NSSS VENDORS, FACILITY)

MTG NUMBER/ MEETING DATE	FACILITY NSSS VENDOR/ EVENT DESCRIP.	RESPONSIBLE DIVISION/ INDIVIDUAL	TASK DESCRIPTION	SCHEDULE COMPLET. DATE(S)	CLOSED DATE BY DOCUMENT/ MEETING, ETC.	COMMENTS
85-10 06/12/85	DAVIS BESSE 1 BW / LOSS OF ALL MAIN AND AUX. FEEDWATER 6/9/85	DL /DEAGAZIO, A. /	PROVIDE H. DENTON WITH 1) COMPARISON BETWEEN DAVIS BESSE 1 AND TMI-1 OF TMI ACTION PLAN IMPLEMENTATION STATUS 2) COMPARISON OF MFW AND AFW SYSTEMS AT DAVIS BESSE 1 & TMI-1 3) STATUS OF STARTUP FW PUMP UPGRADE.	08/30/85 06/19/85 / /	OPEN / / ITEM 1-6/20/85 MEMO H. THOMPSON TO H. DENTON PROVIDED UPDATE TO NUREG 1066	ITEMS 2 & 3 IN PROGRESS.
85-11 07/01/85	LASALLE 2 GE / ED MODIFICATION PROBLEMS AND LOSS OF ALL ECCS JUNE 5-10, 1985	IE /BAER, R. /	CONSIDER ISSUANCE OF IE NOTICE.	08/30/85 / / / /	OPEN / /	
85-11 07/01/85	OYSTER CREEK 1 GE / UNCONTROLLED LEAKAGE OF REACTOR COOLANT OUTSIDE CONTAINMENT 6/12/85	DST /MINNERS LEAD DE /CHERNY ASST.	PRESSURE ISOLATION VALVE TESTING REQUIREMENT TO BE ADDRESSED IN CRGR BRIEFING.	08/30/85 / / / /	OPEN / /	
85-11 07/01/85	OYSTER CREEK 1 GE / UNCONTROLLED LEAKAGE OF REACTOR COOLANT OUTSIDE CONTAINMENT 6/12/85	DHFS/BOOHER, H. /	DETERMINE EFFICACY OF EPG/ OPERATOR ACTION AND EXCESSIVE RESPONSE TIME.	08/30/85 / / / /	OPEN / /	
85-11 07/01/85	RANCHO SECO 1 BW / RCS HIGH POINT VENT LEAK, 6/23/85	DL /MINER, S. /	DETERMINE STATUS OF IEB 79-14.	08/30/85 / / / /	OPEN / /	THE LICENSEE IS COMPLETING THE INSPECTION
85-11 07/01/85	RANCHO SECO 1 BW / STARTUP PROBLEMS- JUNE 1985	DL /MINER, S. /	SCHEDULE CONFERENCE CALL WITH REGION AND LICENSEE PRIOR TO RESTART TO DETERMINE PLANT READINESS FOR OPERATIONS.	08/15/85 07/12/85 / /	OPEN / /	EVALUATION IN PROGRESS
8[illegible] 07/[illegible]	OCONEE 2 BW / EXTENDED BLOWDOWN FROM MAIN STEAM SAFETY VALVES	DL /NICOLARIS, H /	SCHEDULE CONFERENCE CALL WITH LICENSEE TO DISCUSS CROSBY VALVE PERFORMANCE	09/30/85 / / / /	OPEN / /	

OPERATING REACTORs' EVENTS MEETING FOLLOWUP ITEMS
AS OF MEETING 85-12 ON JULY 23, 1985

(ORDERED BY ASCENDING MEETING DATES, NSSS VENDORS, FACILITY)

MEETING NUMBER/ MEETING DATE	FACILITY NSSS VENDOR/ EVENT DESCRIP.	RESPONSIBLE DIVISION/ INDIVIDUAL	TASK DESCRIPTION	SCHEDULE COMPLET. DATE(S)	CLOSED DATE BY DOCUMENT/ MEETING, ETC.	COMMENTS
85-12 07/23/85	SEABROOK W / CROSBY MAIN STEAM SAFETY VALVE FLOW DEFICIENCY 12/84	DSI /MARSH /	ANALYZE SAFETY IMPLICATIONS OF VALVE FLOW DEFICIENCY	09/30/85 / / / /	OPEN / /	
85-12 07/23/85	SEABROOK W / CROSBY MAIN STEAM SAFETY VALVE FLOW DEFICIENCY 12/84	DE /CHERNEY /	INVESTIGATE ADEQUACY OF TESTING AND VALIDITY OF EXTRAPOLATING DATA FROM SMALL TO LARGE VALVES	09/30/85 / / / /	OPEN / /	
85-12 05/30/85	HATCH 1 GE / STUCK OPEN SAFETY RELIEF VALVE	ORAB/CARUSO, M. /	WILL DEVELOP TIA TO COORDINATE IE NOTICE AND FURTHER INVESTIGATIVE EFFORTS.	07/30/85 / / / /	CLOSED 07/23/85 TIA IN CONCURRENCE	

TURKEY POINT 3 - REACTOR TRIP AND AFW VALVE FAILURE

JULY 21, 1985 AND JULY 22, 1985 (T. ROTELLA, NRR)

LICENSEE: FLORIDA POWER & LIGHT

FACILITY: TURKEY POINT UNIT - 3

VENDOR: WESTINGHOUSE

INITIAL CONDITIONS: 100% POWER; STEADY STATE

CURRENT CONDITIONS: PLANT IN COLD SHUTDOWN

EVENT SEQUENCE:

EVENT HRS.	DATE	TIME		
0	7/21/85	23:41	-	LIGHTNING STRIKE IN OR NEAR PLANT
			-	REACTOR TRIP FROM 100% POWER
			-	FIRST-OUT ANNUNCIATOR:TURBINE TRIP (CAUSE UNVER INVESTIGATION)
+1	7/22/85	00:40	-	LO-LO LVL S/G #3B DUE MFW BYPASS VALVE FAILURE TO OPEN UPON OPERATOR INITIATION (CAUSE UNDER INVESTIGATION)
			-	AFW PUMPS AUTO STARTED PROPERLY HOWEVER THE A&C PUMPS TRIPPED ON MECHANICAL OVERSPEED (CAUSE UNDER INVESTIGATION)
+3.5	7/22/85	04:05	-	MFW PUMP TRIP DUE TO HI LVL S/G #3C (CAUSE WAS DUE TO LEAKING MFW BYPASS REGULATING VALVE)
			-	ALL 3 AFW PUMPS SUCCESSFULLY STARTED AND RAN
+5.5	7/22/85	06:04	-	PLANT BEGIN COOLDOWN FROM HOT S/D TO COLD S/D
			-	AFW FLOW CONTROL VALVE FAILED TO CLOSE, (AFW V-2833 TO S/G #3C) CAUSE UNDER INVESTIGATION)

[illegible]

~~Frank, Nitu[?] I listed as as technical contacts. Gary~~

Concur: Hamner
Cherny
Bizenek[?]

Viola! Please change as marked. F.C.

[redacted]

D. Crutchfield

MEMO FOR: ~~[illegible]~~, AD Safety Assessment
DL

FROM: ~~[illegible]~~ RJB

SUBJECT: TRANSMITTAL OF PROPOSED IE INFORMATION NOTICE RE: MAIN STEAM SAFETY VALVES

Please find attached ~~[illegible]~~ our draft of a proposed ~~[illegible]~~ IE Information Notice entitled "Main Steam Safety Valve Test Failures and Ring Setting Adjustments" ~~[illegible]~~

~~[illegible]~~

for your transmittal

~~[illegible]~~

to the Office of Inspection and Enforcement. The basis for this Notice was discussed at the Operating Reactor Events Briefing of July 24, 1985. The specific [illegible] have been coordinated with M. Caruso of ORAB, DL. RJB

cc: J. Knight, H. Thompson, H. Nicolaras, V. Nerses, E. Jordan, IE
F. Cherny, G. Holahan, R. Oller, IE
G. Hammer, R. Weisman, R. Baer, IE
M. Caruso, J. Durr, RI
H. Gregg, RI

B/41

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT
WASHINGTON, D.C. 20555

AUGUST , 1985

IE Information Notice No. 85-xx: Main Steam Safety Valve Test Failures and Ring Setting Adjustments } ALL CAP

Addressees:

All PWR nuclear power reactor facilities holding an operating license (OL) or a construction permit (CP)

Purpose:

This Information Notice is being provided as a notification of a potentially significant problem pertaining to spring-actuated main steam safety valves (See Figure 1), that may possess less than the full rated flow capacity required for over pressure protection of the secondary cooling system in PWRs. It is expected that recipients will review the information for applicability to their facilities and consider actions, if appropriate, to preclude a similar problem at their facilities. However, suggestions contained in this Information Notice do not constitute NRC requirements; therefore, no specific action or written response is required.

Description of Circumstances:

Between October 16, 1984, and December 1, 1984, Wyle Laboratories conducted several full flow steam tests on two separate main steam safety valves (MSSV~~s~~) manufactured by Crosby Valve and Gage Company. These Crosby 6R10 MSSVs are to be installed by Public Service of New Hampshire on the Seabrook main steam system. The tests were conducted in order to determine the adequacy of various MSSV discharge piping arrangements. During the tests the valves were instrumenta~~l~~ed to measure valve disk lift. The valves were installed on the test facility with the Crosby recommended settings of the valve adjusting rings. With these factory ring settings the valve achieved about 50% of the full disk lift required to develop full steam flow capacity within the ~~ASME~~ Code required 3% accumulated overpressure limit. Adequate lift was not achieved for either valve with the factory ring settings, even for the largest diameter (least flow resistance) vent pipe tested. The guide ring of both valves was subsequently adjusted a significant amount (150 notches) during the course of testing and full disk lift was subsequently achieved.

American Society of Mechanical Engineers (ASME) ~~Boiler and~~

Boiler and Pressure Vessel

~~-2-~~

These types of full flow tests are normally not performed by either utility or the valve vendor on MSSVs, nor are such tests required for capacity certification according to the ~~American Society of Mechanical Engineers~~ (ASME) ~~Boiler and Pressure Vessel~~ Code, Section III. In general, these valves are capacity certified by tests on much smaller size valves, and the capacities then extrapolated to larger size valves. The MSSVs on most PWRs, while not necessarily the same model, are like those at Seabrook in that these valves are generally at the upper end of the valve size range. This raises the concern that full flow functional demonstration of these type valves may never have been performed.

A related MSSV problem which has occurred at several PWRs in the past few years pertains to excessive blowdown of main steam system pressure during transients which have actuated some MSSVs. On separate occasions at the Oconee, Salem, Trojan, and Davis-Besse nuclear facilities, MSSVs have remained open below the correct reseat pressure and have blown down excessive amounts of steam. The design blowdown value is usually 5% of setpoint pressure and is also dependent on the specific valve ring setting adjustments. Some valves at these facilities have exhibited as much as 10% blowdown. This raises the concern that some MSSVs may remain open too long, relieving excessive quantities of steam possibly adversely affecting cooling of the primary system and causing excessive thermal stresses on primary system components.

No specific action or written response is required by this Information Notice. If you have any questions about this matter, please contact the Regional Administrator of the appropriate regional office or this office.

Edward L. Jordan, Director
Division of Emergency Preparedness
and Engineering Response
Office of Inspection and Enforcement

Technical Contacts: Frank Cherny, NRR
492-8437
Gary Hammer, NRR
492-8963

Frank —

[redacted]

— A few questions on this

1. ~~What~~ How do the valve manufacturers feel about this? Seems to say that at least one NFG's recommended settings are no good. Do they have a response? Crosby?

2. What do we expect each owner to do who receives the notice?

3. Presumably I/E will get ELD to review

Bob

Call D. Zahorsky Mon. 8-12
- Read Notice
  Can't send him draft
- Using NSSS figure

B/42

Mark,
Attached is
a clean copy
to type on.

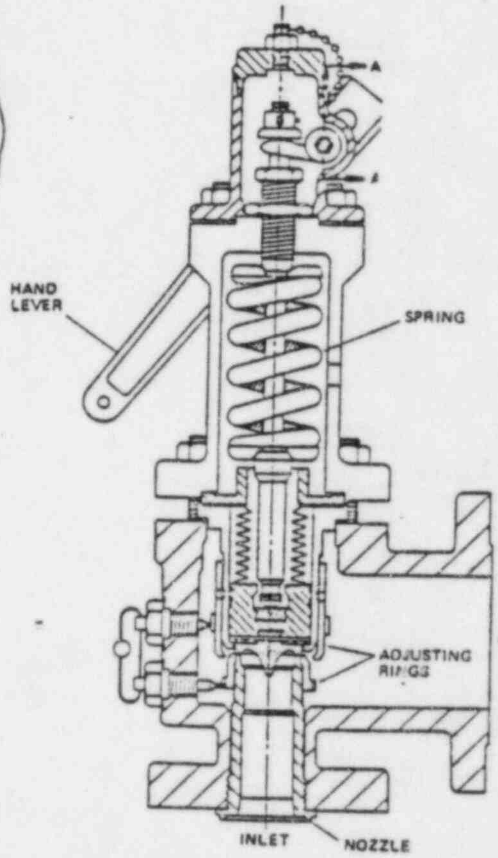
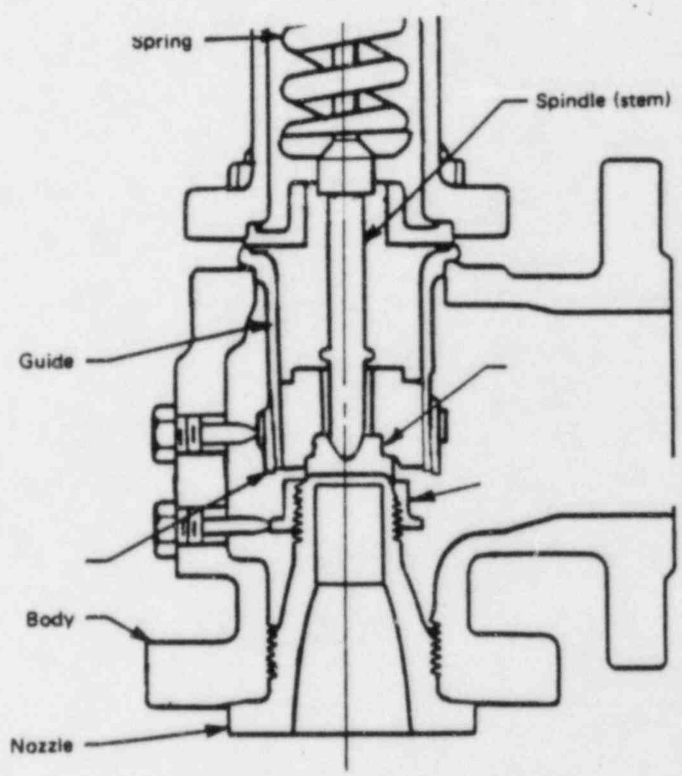


FIGURE 1

Typical Spring
Safety Valve


Spring
Spindle (stem)
Guide
Body
Nozzle

ROUTING AND TRANSMITTAL SLIP

Date [redacted]

TO: (Name, office symbol, room number, building, Agency/Post)	Initials	Date
1.		
2. Frank Cherny	Gary -	Info.
3. P-522A	Note	G.L. 6-16
4.	+ Iconce.	F.C. 8-9
5.		

Action	File	Note and Return
Approval	For Clearance	Per Conversation
As Requested	For Correction	Prepare Reply
Circulate	X For Your Information	See Me
Comment	Investigate	Signature
Coordination	Justify	

REMARKS

You may find the attached item of interest.

DO NOT use this form as a RECORD of approvals, concurrences, disposals, clearances, and similar actions

FROM: (Name, org. symbol, Agency/Post)	Room No.—Bldg.
	531
Warren H. Swenson NRR/DL/ORAB	Phone No. 2-7876

5041-102

OPTIONAL FORM 41 (Rev. 7-76)
Prescribed by GSA
FPMR (41 CFR) 101-11.206

* U.S.G.P.O.: 1984-421-529/412

B/43

MORNING REPORT - REGION II
DATE: August 8, 1985

LICENSEE/FACILITY	NOTIFICATION/SUBJECT	DESCRIPTION OF ITEM OR EVENT
Grand Gulf DN: 50-416	HQ Duty Officer, 8/7	On 8/7, at 3:02 p.m., Unit 1 tripped from 94% power due to a main turbine generator trip. The turbine generator tripped on "low cooling water flow to the rotor" due to a flow transmitter failing low. Safety relief valves lifted and reseated as they should have. The licensee is making necessary repairs and plans to restart the unit today. For information only.
[redacted] 50-287	Resident Inspector, 8/8	Unit 3 Shut Down on 8/8, for a scheduled fifty-seven day refueling outage. The major maintenance tasks planned are sludge lancing and eddy current testing of steam generators, Aris inspection (inservice) of reactor hot legs, rebuilding one reactor coolant pump, overhaul of four code safety valves (main steam), overhaul of two-low pressure turbines, overhaul of both main feedwater turbines, and testing and plugging of moisture separator reheater tubes.
St. Lucie 2 DN: 50-389	HQ Duty Officer, 8/8 Reactor Shutdown	At 2:06 a.m., 8/8, a fuse blew in the normal power feeder to the "A" safeguards actuation cabinet. Immediately, a fuse blew in the backup power feed. These two power feeds are rectified - then auctionered to provide uninterrupted power to the cabinet. With both power feeds lost, the actuation cabinet properly initiated "A" side safeguards features: high and low pressure safety injection pumps, start of the "A" emergency diesel generator, and containment isolation. The injection pumps did not inject because plant pressure never reduced to the injection point. The emergency diesel generator did not load onto the bus because normal power was available. Containment isolation interrupted component cooling water (CCW) to all four reactor coolant pumps (RCPs) at their containment penetrations, and subsequently these valves were reopened; however individual RCP CCW lines to 2A1 and 2B2 RCPs were isolated because these individual RCP CCW isolation valves are powered from a power supply which is stripped upon safeguards initiation. These valves could not be reopened. The plant used natural circulation to remove decay heat while the pumps were stopped. 2A1 and 2B2 RCPs remain stopped pending shaft seal inspection at cold shutdown. "B" safeguards initiated due to low pressurizer pressure and was reset in about five minutes when pressure was restored above the setpoint. Auxiliary Feedwater operated as required. The unit is being cooled to cold shutdown for RCP seal inspection. An Unusual Event was declared at 3:10 a.m. and terminated at 6:31 a.m. PN issued.

low

Copy to HEB (Cheney) PM

Frank Cherny Make P-522 A

Copy to PM

Daily Highlight

T.R./PM - Refine bullets for DR Brief

Steve Long Doing TEN

D. Verrelli: 2/4 Seals failed

Mod

MORNING REPORT - REGION II
DATE: August 8, 1985
-2-

General

Representatives of the Mississippi Power and Light Company are in the Region II Office to attend a meeting regarding the apparent inadequate safety evaluation of the Appendix R revised Safe Shutdown Systems at the Grand Gulf facility.

Public Service of New Hampshire

New Hampshire Yankee Division

Seabrook Station
Engineering Office

E. Brown
A. Cerne
R. Cummings
R. DeLoach
W. Derrickson
J. DeVincentis
[redacted]eigenbaum
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W. Hall
R. Harrison
D. Hunter
W. Johnson
G. Kingston
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M. McKenna
B. Middleton
D. Moody
NRC Subject File
Projects - WJD
Projects - Chrono
Ropes & Gray (3)
F. Sabadini
A. Shepard
R. Sweeney
T.F. Q2.2.2
G. Thomas
H. Tracy
J. Tribble
UE&C & W (SB-19770)
M. Wiley
ASLB
10CFR50.55(e)File
J. Allen
INPO

SBN- 863
T.F. Q2.2.2

United States Nuclear Regulatory Commission
Region I
631 Park Avenue
King of Prussia, PA 19406

Attention: Mr. Richard W. Starostecki, Director
Division of Project and Resident Programs

References:
(a) Construction Permits CPPR-135 and CPPR-136, Docket Nos. 50-443 and 50-444
(b) Telecon of December 21, 1984, A. L. Legendre, Jr. (YAEC) to J. Grant (Region I)
(c) NHY Letter SBN-751 dated January 17, 1985, John DeVincentis to R. W. Starostecki, NRC Region I
(d) NHY Letter SBN-788 dated April 8, 1985, John DeVincentis to R. W. Starostecki

Subject: Final 10CFR50.55(e) Report, "Main Steam Safety Valve Ring Setting Deficiency," (CDR 84-00-19)

Dear Sir:

In References (c) and (d), we filed interim 10CFR50.55(e) reports regarding a ring setting deficiency for the main steam safety valves. The valves were sent to Wyle Laboratories for testing for determination of the proper ring settings. The tests were completed and the results are contained in Wyle Laboratories Report No. 47787-01 dated July 12, 1985.

The objectives of the tests were to:

1. Determine if the "as-shipped" ring settings of the valves would allow the required disc travel with minimum tailpipe backpressure.

2. Determine the effects on [illegible] disc travel for a range of backpressures between 180 and 390 psig.

The results of the "as-shipped" ring setting tests indicated that the valves could not achieve the required disc travel with 3% steam accumulation at minimum tailpipe pressures of 15-20 psig.

8509250143

B/47

During the test, the upper (guide) ring setting was adjusted from +155 notches to 0 and +25 notches, and full required disc travel was achieved at 3% steam accumulation under the full range of tailpipe backpressure tested.

As a result of these test, we have agreed with Crosby, the valve manufacturer, that the optimum ring settings for the Seabrook main steam safety valves shall be -25 notches for the lower (nozzle) ring (original setting) and +25 notches for the upper (guide) ring. The corrections were completed by a Crosby service representative at the Wyle facility prior to returning the valves to the Seabrook Site.

This is our final report on this subject.

Very truly yours,

John DeVincentis

John DeVincentis, Director
Engineering and Licensing

cc: Atomic Safety and Licensing Board Service List

Director, Office of Inspection and Enforcement
U. S. Nuclear Regulatory Commission
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