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Closeout of IE Bulletin 80-18: Maintenance of Adequate Minimum Flow Thru Centrifugal Charging Pumps Following Secondary Side High Energy Line Rupture

Prepared by W. J. Foley, R. S. Dean, A. Hennick

PARAMETER, Inc.

Prepared for
U.S. Nuclear Regulatory
Commission

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PARAMETER IE-159

Closeout of IE Bulletin 80-18: Maintenance of Adequate Minimum Flow Thru Centrifugal Charging Pumps Following Secondary Side High Energy Line Rupture

Manuscript Completed: July 1987
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Prepared by
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Washington, DC 20555
NRC FIN B8729

ABSTRACT

On May 8, 1980, Westinghouse reported to the NRC/IE per Title 10 CFR Part 21 that one or more centrifugal charging pumps could be damaged by low flow at certain plants before satisfactory termination of safety injection after a secondary side high energy line rupture. The plants of concern had been notified, a calculational method of evaluation and plant-specific reviews had been recommended, and interim modifications had been proposed by Westinghouse. The NRC/IE issued IE Bulletin 80-18 on July 24, 1980 because of this potential safety-related problem. Licensees and near-term licensees of pressurized water reactors (PWRs) were required to take specific actions and submit written responses. Utilities with PWRs under construction were issued the bulletin for information, in preparation for the licensing process. Evaluation of utility responses and NRC/Region inspection reports shows that the bulletin can be closed out per specific criteria for 44 (96%) of the 46 facilities to which it was issued for action. A followup item for the remaining dual plant is proposed for use by the NRC to ensure satisfactory completion of required and corrective actions.

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CLOSEOUT OF IE BULLETIN 80-18: MAINTENANCE OF ADEQUATE MINIMUM FLOW THRU CENTRIFUGAL CHARGING PUMPS FOLLOWING SECONDARY SIDE HIGH ENERGY LINE RUPTURE

INTRODUCTION

In accordance with the Statement of Work in Task Order 013 under NRC Contract 05-85-157-02, this report provides documentation for the closeout status of IE Bulletin 80-18. Documentation is based on the records obtained from the NRC Document Control System.

IE Bulletin 80-18 was issued to licensees for action July 24, 1980 because of concern about the potential for damage to centrifugal charging pumps in safety injection systems of PWR facilities. Although only certain plants had been identified by Westinghouse, the bulletin was issued for action and written response to all operating PWRs and to non-operating PWRs which were nearing licensing.

For background information and required actions, IE Bulletin 80-18 and its enclosed 10 CFR Part 21 Westinghouse letter to the NRC/IE are included in Appendix A. The letter contains the recommended calculational method of evaluation, the description of preferred and alternative interim modifications, and a list of plants of concern. Evaluation of utility responses and NRC/Region inspection reports is documented in Appendix B as the basis for bulletin closeout. Facilities to which the bulletin was issued for information are listed in Appendix B. A followup item is proposed in Appendix C for use by the NRC in assuring satisfactory completion of corrective action. Utility manhours expended on the bulletin are tabulated in Appendix D. Abbreviations used in this report and associated documents are presented in Appendix E.

SUMMARY

In the following summary items, and in the closeout criteria on Page 3, the underlined terms <u>listed</u> or <u>unlisted</u> refer to the listing of plants of concern in the Westinghouse 10 CFR Part 21 letter enclosed with the bulletin (see Appendix A, Page A-8).

1. The bulletin has been closed out for the following <u>listed</u> facilities per Criterion 1 (see Page 3):

   None

2. The bulletin has been closed out for the following 11 listed facilities per Criterion 2 (see Page 3):

Beaver Valley 1	North Anna 1,2	Surry 1,2
Diablo Canyon 1	Salem 1,2	Trojan
McGuire 1	Sequoyah 1	

3. The bulletin has been closed out for the following 21 unlisted facilities per Criterion 3 :

Arkansas 2	Oconee 1,2,3	San Onofre 1
Calvert Cliffs 1,2	Palisades	St. Lucie 1
Haddam Neck	Point Beach 1,2	Turkey Point 3,4
Indian Point 2,3	Prairie Island 1,2	Yankee-Rowe 1
Maine Yankee	Robinson 2	

4. The bulletin has been closed out for the following four (4) unlisted facilities per Criterion 4 :

Arkansas 1	Rancho Seco 1	TMI 1
Crystal River 3		

5. The bulletin has been closed out for the following facility per Criterion 5 :

   Ginna

6. The bulletin has been closed out for the following seven (7) facilities per Criterion 6:

Cook 1,2	Farley 1	Kewaunee
Davis-Besse 1	Fort Calhoun 1	Millstone 2

7. The bulletin is called open for the following two (2) facilities:

   Zion 1,2

REMAINING AREAS OF CONCERN

There are none because a followup item is proposed in Appendix C to ensure satisfactory completion of required and corrective actions at Zion 1,2.

CRITERIA FOR CLOSEOUT OF BULLETIN

In the following closeout criteria, and in the summary items on pages 1 and 2, the underlined terms listed or unlisted refer to

the listing of plants of concern in the Westinghouse 10 CFR Part 21 letter enclosed with the bulletin (see Appendix A, Page A-8).

The bulletin is closed out for facilities to which one of the following criteria applies:

1. A response for the listed facility indicates that minimum cooling flow is assured for all conditions by the required calculations, and an NRC/Region inspection report indicates compliance with required actions and no need for corrective actions.

2. A response for the listed facility indicates that minimum cooling flow is not assured for all conditions by the required calculations, and an NRC/Region inspection report indicates that required and corrective actions have been completed satisfactorily.

3. A response for the unlisted facility indicates that minimum cooling flow is assured for all conditions (without calculations) because an open line assumed for all safety-related analyses is provided to guarantee minimum cooling flow during high pressure injection, and an NRC/Region inspection report indicates compliance with required actions and no need for corrective actions.

4. A response for the unlisted facility indicates that minimum cooling flow is assured for all conditions per the calculations recommended by Westinghouse, and an NRC/Region inspection report indicates compliance with required actions and no need for corrective actions.

5. Although a response for the unlisted facility states only that centrifugal charging pumps are not used, an NRC/Region inspection report indicates that all safety injection pumps were reviewed and calls the bulletin closed.

6. The NRC/Region contact for the facility has assured the Technical Monitor assigned to this task at NRC Headquarters that all required and corrective actions have been completed satisfactorily.

APPENDIX A

Background Information and Required Actions

Notes:

1. For actions required by the bulletin, refer to pages A-2 and A-3.

2. For interim modifications recommended by Westinghouse, refer to Page A-6.

3. For identification by Westinghouse of plants of potential concern, refer to Page A-8.

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

IE Bulletin No. 80-18
Date: July 24, 1980
Page 1 of 3

MAINTENANCE OF ADEQUATE MINIMUM FLOW THRU CENTRIFUGAL CHARGING PUMPS FOLLOWING SECONDARY SIDE HIGH ENERGY LINE RUPTURE

Description of Circumstances:

Letters similar to the May 8, 1980 notification made pursuant to Title 10 CFR Part 21 (enclosure) were sent from Westinghouse to a number of operating plants and plants under construction (list, within enclosure) in early May, 1980.

The letters and the enclosed "Part 21" letter contain a complete description of the potential problem summarized below. The letters indicated that under certain conditions the centrifugal charging pumps (CCPs) could be damaged due to lack of minimum flow before presently applicable safety injection (SI) termination criteria are met. The particular circumstances that could result in damage vary somewhat from plant to plant, but involve unavailability of the pressurizer power operated relief valves (PORVs), with operation of one or more CCPs repressurizing the reactor during SI following a secondary system high energy line break. Since the SI signal automatically isolates the CCP mini-flow return line, the flow through the CCPs is determined by the individual pump characteristic head vs. flow curve, the pressurizer safety valve setpoint, and the flow resistances and pressure losses in the piping and in the reactor core. That minimum flow may not be adequate to insure pump cooling, and resulting pump damage could violate design criteria before current SI termination criteria are met.

Westinghouse recommends that plant specific calculations outlined in the letter (enclosure) be performed to determine if adequate minimum flow is assured under all conditions. If adequate minimum flow is not assured, Westinghouse recommends specific equipment and procedure modifications which will result in adequate minimum flow. The recommended modifications assure availability of the necessary minimum flow by assuring that the mini-flow bypass line will be open when needed, but will be closed at lower pressures when the extra flow resulting from bypass line closure might be necessary for core cooling.

Actions to be taken by PWR licensees listed in the enclosure as "operating plants," and those listed as "non-operating plants" which are nearing licensing* are listed below:

1. Perform the calculations, outlined in the enclosure, for your plant.

2. If availability of minimum cooling flow for the CCPs is not assured for all conditions by the calculations in 1:

   a. Make modifications to equipment and/or procedures, such as those suggested in the enclosure, to insure availability of adequate minimum flow under all conditions. If modifications are made as described in the attachment for interim modification II, verify that the Volume Control Tank Relief Valve is operable and will actuate at its design setpoint.

   b. Justify that any manual actions necessary to assure adequate minimum flow for any transient or accident requiring SI can and will be accomplished in the time necessary.

   c. Verify that any manipulations required (valve opening or closing, along with the instrumentation necessary to indicate need for the action or accomplishment of the action, etc.) can be accomplished without offsite power available.

   d. Justify that flow available from the CCPs with the modifications in place will be sufficient to justify continued applicability of any safety related analyses which take credit for flow from the CCPs (LOCA, HELB, etc.).

   e. Justify that all Technical Specifications based on the Item 2.d analyses remain valid.

3. Provide the results of calculations performed under Item 1, and describe any modifications made as a result of Item 2 (include the justifications requested).

Actions to be taken by PWR licensees not listed in the enclosure are listed below:

1. In a quantitative manner similar to 1 above, determine whether or not minimum cooling is provided to centrifugal pumps used for high pressure injection, for all conditions requiring SI, prior to satisfying SI

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*Those listed in the enclosure considered to be "nearing licensing" are: North Anna 2, Diablo Canyon 1, McGuire 1, Salem 2, and Sequoyah. These plants must respond in writing within the specified time. Other non-licensed plants whether or not listed in the enclosure, are not required to submit a written response at this time.

termination criteria. If a "minimum flow bypass" line is present which remains open during high pressure injection, and if that line guarantees that minimum cooling flow will be provided to the pumps under such conditions, then no further calculations are required if all safety related analyses (Item 2.d above) assumed presence of the open line.

2. Same as 2 above.

3. Same as 3 above.

Licensees of all operating PWR power reactor facilities and those nearing licensing* shall submit the information requested within 60 days of the date of this letter. Include in your response to this Bulletin, (a) your schedule for any changes proposed, (b) if reactor operation is to continue prior to completion of the proposed changes, include your justification for continued operation.

Reports shall be submitted to the Director of the appropirate NRC Regional Office and a copy forwarded to the Director, NRC Office of Inspection and Enforcement, Division of Reactor Operations Inspection, Washington, D. C. 20555.

Approved by GAO, B280225 (R0072); clearance expires 7-31-80. Approval was given under a blanket clearance specifically for identified generic problems.

Enclosure:
Ltr from T. M. Anderson, W
to V. Stello, IE (w/attachments) dtd 5/8/80

*Those considered to be "nearing licensing" are: North Anna 2, Diablo Canyon 1, McGuire, Salem 2, and Sequoyah.

Westinghouse Electric Corporation

Water Reactor Divisions

Nuclear Technology Division

Box 355
Pittsburgh Pennsylvania 15230

May 8, 1980

NS-TMA-2245

Mr. V. Stello, Director
Office of Inspection and Enforcement
U. S. Nuclear Regulatory Commission
1717 H Street
Washington, D. C. 20555

Subject: Centrifugal Charging Pump Operation Following Secondary Side High Energy Line Rupture

Dear Mr. Stello:

This letter is to confirm the telephone conversation of May 8, 1980 between Westinghouse and Mr. Ed Blackwood of Division of Reactor Operations Inspection, Office of Inspection and Enforcement, regarding notification made pursuant to Title 10 CFR Part 21.

A review of the Westinghouse Safety Injection (SI) Termination Criteria following a secondary side high energy line rupture (feedline or steamline rupture at high initial power levels) has revealed a potential for consequential damage of one or more centrifugal charging pumps (CCPs) before the SI termination criteria are satisfied and CCP operation terminated. Such consequential damage may adversely impact long-term recovery operations for the initiating event and is not permitted by design criteria. This concern exists for plants which utilize the CCPs as Emergency Core Cooling System (ECCS) pumps, where the CCPs are automatically started, and where the CCP miniflow isolation valves are automatically isolated upon safety injection initiation. Attachment A identifies plants potentially subject to this concern. A summary of the concern and recommendations follow.

Following a secondary side high energy line rupture and associated reactor trip, Reactor Coolant System (RCS) pressure and temperature initially decrease. Safety injection is actuated and the CCPs start to increase RCS inventory. Reactor Coolant System pressure and temperature subsequently increase due to the loss of secondary inventory, steamline and feedline isolation, RCS inventory addition and reactor core decay heat generation. The accident scenario may vary with rupture size and specific plant design, but it will develop into a RCS heatup transient with accompanying increase in RCS pressure. As RCS pressure increases, the pressurizer power-operated relief valves (PORVs) are designed to limit RCS pressure to 2350 psia. Although these valves are normally available, they are not designed as safety-related equipment. It can be postulated that, due to either loss of offsite power,

adverse environment inside containment, the pressurizer PORV in manual mode, or the PORV block valve in a closed position, due to PORV leakage, the pressurizer PORVs may not be operable. As a result of the RCS heatup and inventory increase, the RCS pressure could rise to the pressurizer safety valve setpoint of 2500 psia within approximately 200 seconds and remain at that pressure until transient "turnaround." Transient "turnaround" ca[illegible] occur between 1800 and 4200 seconds depending on operator action and available equipment. During the initial portion of this transient, the SI termination criteria may not be satisfied. Consequently, the RCS pressure can reach the pressurizer safety valve relief pressure before CCP operation is terminated. During this period, the minimum flow required for CCP operation must be satisfied by flow to the RCS since the CCP miniflow isolation valves are automatically closed on safety injection initiation. This requires that the CCPs be able to deliver their minimum required flow to the RCS at the safety valve setpoint pressure.

To evaluate this concern, Westinghouse has develope[illegible] a calculational method and has reviewed typical CCP head versus flow performance curves and other representative plant parameters. The calculational method considers the effects of safety valve relief setpoint accuracy, RCS piping resistance, ECCS piping resistance, number of CCPs operating, technical specification allowable CCP head degradation, and uncertainties associated with in-plant verification testing. The analyses for two CCP operation, the best estimate condition, is similar to the analysis for one CCP operation except that the flowrate used to determine ECCS piping line loss must ensure the minimum flow through each pump. For example, at a specific required head, the pump with the higher developed head may be required to deliver greater than the minimum flow in order to permit the lower head pump to meet the minimum flow requirement. This generic evaluation indicates that sufficient flow to satisfy CCP minimum flow requirements to avoid pump degradation may not be ensured for a secondary system high energy line rupture under the conditions described above.

Based on the generic evaluation, Westinghouse recommends that operating plants perform a plant specific evaluation to assess this concern. Attachment B provides the Westinghouse calculational method and a sample calculation which can be used in this evaluation. Based on Westinghouse generic review, satisfactory results may not be obtained. Should a plant specific concern be identified, the following recommendations have been developed and can be tailored to specific plant applications for the interim until necessary design modifications can be implemented. The interim modifications consist of system alignment and [illegible]ting procedure changes to provide backup to the pressurizer PORVs in ensur[illegible] that CC[illegible]nimum flow requirements are satisfied. In conjunction with t[illegible]rim mo[illegible]ons, it is recommended that plants, (a) review the pressu[illegible] to maximize the availability of these valves to limit [illegible]surizer safety valves, and (b) review the maintenan[illegible]ical specifications for the backup (i.e., third) charging pump [illegible] availability for long-term recovery from a secondary si[illegible]e recommendations, in combination with the interim

modifications described below, are considered sufficient to address this concern in the interim until necessary design modifications can be implemented.

Interim Modification I

This interim modification is preferred and requires that component cooling water be supplied to the seal water heat exchanger following safety injection initiation in order to provide cooling for CCP miniflow.

1. Verify that CCP miniflow return is aligned directly to the CCP suction during normal operation with the alternate return path to the volume control tank isolated (lock closed).

2. Remove the safety injection initiation automatic closure signal from the CCP miniflow isolation valves.

3. Modify plant emergency operating procedures to instruct the operator to:

   a. Close the CCP miniflow isolation valves when the actual RCS pressure drops to the calculated pressure for manual reactor coolant pump trip.

   b. Reopen the CCP miniflow isolation valves should the wide range RCS pressure subsequently rise to greater than 2000 psig.

Interim Modification II

This modification is an alternative for plants in which component cooling water is not supplied to the seal water heat exchanger following safety injection initiation. Since miniflow cooling is not provided, this alternative directs miniflow to the volume control tank to permit the CCP minimum flow requirements to be satisfied with cool uncirculated water. The volume control tank acts as a surge tank to collect miniflow following safety injection initiation with excess flow directed to a holdup tank via the volume control tank relief valve.

1. Align the CCP miniflow to the volume control tank during normal operation with the miniflow return path direct to the CCP suction isolated (lock closed). Verify that the volume control tank relief valve and discharge line capacity exceeds the miniflow requirements of all CCPs plus the reactor coolant pump seal return flow.

2. Same as Interim Modification I, Item 2.

3. Same as Interim Modification I, Item 3.

Based on the generic evaluation, Westinghouse has initiated efforts to perform additional plant specific analyses for non-operating plants and to develop design modifications to resolve any identified concerns. The modifications will be designed to safety-related standards and will be compatible with Westinghouse SI termination criteria and standardized technical specifications.

If you require further information, please call Ray Sero (412-373-4189) of my staff.

Very truly yours,

T. M. Anderson

T. M. Anderson, Manager
Nuclear Safety Department

TMA/jaw

Attachments

ATTACHMENT A

OPERATING PLANTS

3-Loop	4-Loop
Baaver Valley 1	Cook 1 & 2
Farley 1	Salem 1 & 2
Surry 1 & 2	Trojan
North Anna 1 & 2	Zion 1 & 2
	Sequoyah 1

NON-OPERATING PLANTS

Beaver Valley 2	Braidwood 1 & 2
Farley 2	Byron 1 & 2
Shearon Harris 1, 2, 3 & 4	Calloway 1 & 2
Virgil Summer	Catawba 1 & 2
	Comanche Peak 1 & 2
	Diablo Canyon 1 & 2
	Jamesport 1 & 2
	Haven
	Marble Hill 1 & 2
	McGuire 1 & 2
	Millstone 3
	Seabrook 1 & 2
	Sequoyah 2
	Sterling
	Vogtle 1 & 2
	Watts Bar 1 & 2
	Tyrone
	Wolf Creek

MINIMUM CENTRIFUGAL CHARGING PUMP FLOW DURING TWO PUMP PARALLEL SAFETY INJECTION OPERATION

In order to ensure that minimum pump flow is maintained during parallel safety injection operation of two centrifugal charging pumps (CCPs), Westinghouse provides below a sample calculation utilizing actual plant data and determines what actual CCP developed head at the miniflow flowrate must be available.

Step 1: Individually determine the developed head of each CCP at the miniflow flowrate of 60 gpm from field test data. (two pumps for 4-loop plants and three pumps for 3-loop plants)

Sample: Maximum developed head pump
2571.4 psid = 5940 ft. @ 60 gpm

Minimum developed head pump
2554.1 psid = 5900 ft. @ 60 gpm

Step 2: Correct the pump head for testing error. Add the appropriate error in determining the above measured developed head, i.e., instrument error plus reading error, to the maximum developed head and subtract this error from the minimum developed head.

Sample: Pressure instrument accuracy of ± 0.5 percent x span of measuring instrument of 3000 psig = 15 psi (35 ft. of head), plus 10 psi (23 ft.) reading accuracy = 58 ft.

The resultant CCP developed heads at miniflow which can be supported are a maximum developed head of 5998 ft. for the maximum head pump, and a minimum developed head of 5842 ft. for the minimum head pump.

Step 3: Determine total CCP flow. Construct a pump curve for the maximum head pump that is parallel to the actual "as-built" vendor pump curve and passes through the above determined developed head at the miniflow flowrate which is the measured developed head plus the determined measurement accuracy. (See attachment Figure 1.)

Use this head versus flow curve to determine the flow delivered by the maximum head pump (strong pump) at the developed head of the minimum head pump (weak pump) at the miniflow flowrate (i.e., 5842 ft. as determined in Step 1).

Sample: As illustrated in Figure 1, the delivered flow of the strong pump at 5842 ft. is 150 gpm. Therefore, the total flow from both CCPs which guarantees that the weak CCP will be delivering at least 60 gpm is 210 gpm (150 gpm + 60 gpm).

Step 4: Determine Injection Piping Head Loss. The head loss due to friction in the safety injection/RCP seal injection piping is determined as follows:

The Δh_f is equal to the strong CCP developed head at runout flow. This resistance is established during the CCP flow balance testing which limits CCP flow to the runout limit. The injection piping resistance (k) is equal to the developed head of the strong CCP at its runout flow divided by the (runout flowrate)^2.

e.g. $$k = \frac{\text{developed head}}{(\text{runout flowrate})^2} = \frac{\Delta h_f}{Q^2} = \frac{1500\ \text{ft.}}{(550\ \text{gpm})^2}$$

$$k = 4.96 \times 10^{-3}\ \text{ft./gpm}^2$$

The resistance of the injection piping (Δh_f), at the total CCP flow required to maintain 60 gpm through the weak CCP is:

$$\Delta h_f = kQ^2 \text{ or } \Delta h_f = (4.96 \times 10^{-3} \frac{ft.}{gpm^2}) (210 \text{ gpm})^2 = 219 \text{ ft.}$$

Step 5: Determine head loss through the Reactor Coolant System. Consider that the reactor coolant pumps are operating, therefore, the pressure drop from the CCP cold leg injection nozzles through the reactor vessel to the pressurizer surge line off the hot leg at full RCS flow are to be included. This pressure drop is approximately 50 psid (116 ft.) for 4-loop plants and 48 psid (111 ft.) for 3-loop plants. This pressure drop must be overcome by the CCPs in order to deliver flow to the RCS at the hot leg/pressurizer pressure.

Step 6: Determine the elevational head between the RWST and the pressurizer safety valves.

e.g.	RWST elevation	- 160 ft.
	CCP suction elevation	- 100 ft.
	RCS cold leg injection nozzle elevation	- 126 ft.
	Pressurizer safety valve elevation	- 187 ft.
	RWST to CCP suction	- 60 ft.
	minus CCP suction to RCS	- (-26 ft.)
	minus RCS to pressurizer safety valves (61 ft. assuming a full pressurizer) corrected for density difference	- (-44 ft.)
		-10 ft.

Thus, in this example the CCPs must provide an additional 10 ft. of elevational head.

Step 7: Calculate the pressurizer safety valve relief pressure.

e.g. relief pressure = safety valve nominal relief pressure + 1% setting tolerance

relief pressure = 2485 psig + 25 psig = 2510 psig (5798 ft.)

Step 8: Determine the maximum RCS pressurizer pressure at which 60 gpm minimum flow is maintained through the weak CCP.

Maximum RCS pressure = (CCP developed head at total CCP flowrate) (injection piping head loss) - (head loss through RCS) - (elevation head loss)

Maximum RCS pressure = 5842 ft. - 219 ft. - 116 ft. - 10 ft. = 5497 ft. = 2380 psig

Comparing this pressure to the pressurizer safety valve relief pressure (Step 7) of 2510 psig, it is evident that the 60 gpm flow required for the weak CCP will not be maintained.

Attachment 5
page 5

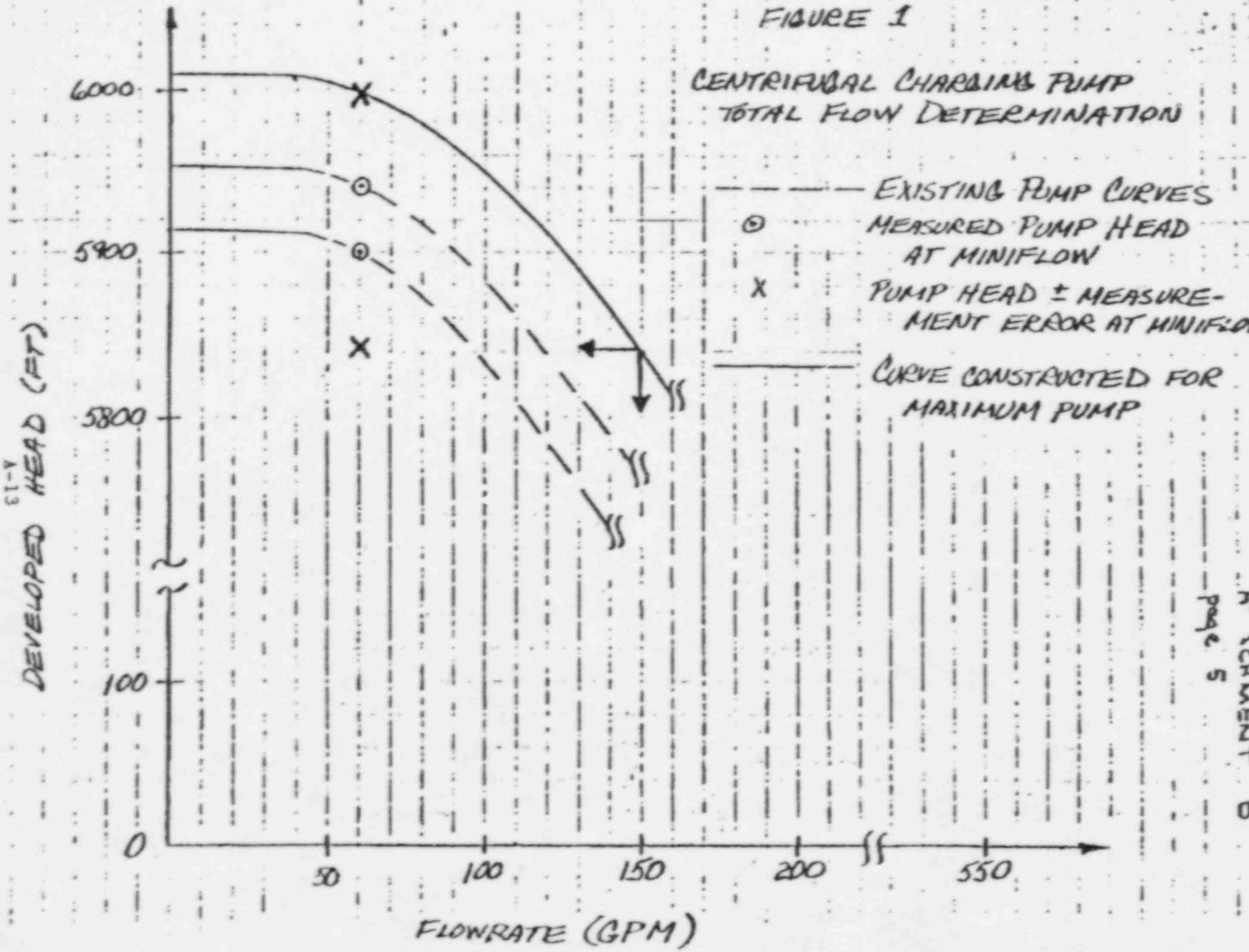

FIGURE 1

CENTRIFUGAL CHARGING PUMP TOTAL FLOW DETERMINATION

APPENDIX B

Documentation of Bulletin Closeout

TABLE B.1 BULLETIN CLOSEOUT STATUS

Facility	Utility	Docket	Facility Status	NRC Region	NSSS	Utility Response Date	Inspection Report and Date	Closeout Status and Criterion
Arkansas 1	AP&L	50-313	OL	IV	B&W	09-22-80	81-30(12-07-81)	Closed 4
Arkansas 2	AP&L	50-368	OL	IV	C-E	09-22-80	81-29(12-07-81)	Closed 3
*Beaver Valley 1	DLC	50-334	OL	I	W	09-24-80	82-01(03-18-82)	Closed 2
Calvert Cliffs 1	BG&E	50-317	OL	I	C-E	08-21-80	81-24(12-28-81)	Closed 3
Calvert Cliffs 2	BG&E	50-318	OL	I	C-E	08-21-80	81-23(12-28-81)	Closed 3
*Cook 1	IMECO	50-315	OL	III	W	09-23-80	80-15(12-12-80)	Closed 6
*Cook 2	IMECO	50-316	OL	III	W	09-23-80	80-14(12-12-80)	Closed 6
Crystal River 3	FPC	50-302	OL	II	B&W	09-22-80 10-24-80	81-15(09-24-81)	Closed 4
Davis-Besse 1	TECO	50-346	OL	III	B&W	09-23-80	80-27(11-10-80)	Closed 6
*Diablo Canyon 1	PG&E	50-275	OL	V	W	10-16-80	81-19(08-12-81)	Closed 2
*Farley 1	APCO	50-348	OL	II	W	09-23-80	82-10(04-27-82)	Closed 6
Fort Calhoun 1	OPPD	50-285	OL	IV	C-E	08-05-80	81-14(07-23-81)	Closed 6
Ginna	RG&E	50-244	OL	I	W	09-04-80	83-19(10-05-83)	Closed 5
Haddam Neck	CYAPCO	50-213	OL	I	W	09-22-80	84-12(07-24-84)	Closed 3
Indian Point 2	Con Ed	50-247	OL	I	W	09-22-80	83-11(05-11-83)	Closed 3
Indian Point 3	PASNY	50-286	OL	I	W	09-19-80	82-03(03-19-82)	Closed 3
Kewaunee	WPS	50-305	OL	III	W	09-22-80	81-02(03-02-81)	Closed 6
Maine Yankee	MYAPCO	50-309	OL	I	C-E	09-19-80	80-14(10-22-80)	Closed 3
*McGuire 1	DUPCO	50-369	OL	II	W	09-22-80 09-29-80	80-32(01-14-81)	Closed 2
Millstone 2	NNECO	50-336	OL	I	C-E	09-22-80	83-06(03-15-83)	Closed 6
*North Anna 1	VEPCO	50-338	OL	II	W	09-26-80 12-22-80	84-06(02-01-85)	Closed 2
*North Anna 2	VEPCO	50-339	OL	II	W	09-26-80 12-22-80	84-06(02-01-85)	Closed 2
Oconee 1	DUPCO	50-269	OL	II	B&W	08-07-80	82-30(09-02-82)	Closed 3
Oconee 2	DUPCO	50-270	OL	II	B&W	08-07-80	82-30(09-02-82)	Closed 3
Oconee 3	DUPCO	50-287	OL	II	B&W	08-07-80	82-30(09-02-82)	Closed 3

See notes at end of table.

TABLE B.1 (contd)

Facility	Utility	Docket	Facil-ity Status	NRC Region	NSSS	Utility Response Date	Inspection Report and Date	Closeout Status and Criterion
Palisades	CPC	50-255	OL	III	C-E	09-23-80	81-02(03-24-81)	Closed 3
Point Beach 1	WEPCO	50-266	OL	III	W	08-26-80	80-15(10-15-80)	Closed 3
Point Beach 2	WEPCO	50-301	OL	III	W	08-26-80	80-15(10-15-80)	Closed 3
Prairie Island 1	NSP	50-282	OL	III	W	08-15-80	80-14(09-19-80)	Closed 3
Prairie Island 2	NSP	50-306	OL	III	W	08-15-80	80-15(09-19-80)	Closed 3
Rancho Seco 1	SMUD	50-312	OL	V	B&W	09-11-80 10-27-80	80-31(12-16-80)	Closed 4
Robinson 2	CP&L	50-261	OL	II	W	09-22-80 12-04-80	81-32(12-23-81)	Closed 3
*Salem 1	PSE&G	50-272	OL	I	W	09-23-80 09-29-80 07-14-81	80-32(01-20-81) 82-05(04-06-82)	Closed 2
*Salem 2	PSE&G	50-311	OL	I	W	09-23-80 09-29-80 07-14-81	80-22(01-20-81) 82-08(04-06-82)	Closed 2
San Onofre 1	SCE	50-206	OL	V	W	09-18-80	80-30(11-04-80)	Closed 3
*Sequoyah 1	TVA	50-327	OL	II	W	09-22-80 10-16-80 11-17-80 11-18-80 07-02-81	81-32(10-13-81)	Closed 2
St. Lucie 1	FPL	50-335	OL	II	C-E	09-22-80	81-02(03-12-81)	Closed 3
*Surry 1	VEPCO	50-280	OL	II	W	09-26-80 12-22-80	81-33(01-07-81)	Closed 2
*Surry 2	VEPCO	50-281	OL	II	W	09-26-80 12-22-80	81-33(01-07-81)	Closed 2
TMI 1	Met-Ed	50-289	OL	I	B&W	10-16-80 01-21-81	81-24(10-19-81)	Closed 4
*Trojan	PGE	50-344	OL	V	W	09-24-80 05-01-81	82-24(09-02-82)	Closed 2
Turkey Point 3	FPL	50-250	OL	II	W	09-23-80	81-13(06-09-81)	Closed 3
Turkey Point 4	FPL	50-251	OL	II	W	09-23-80	81-13(06-09-81)	Closed 3

See notes at end of table.

TABLE B.1 (contd)

Facility	Utility	Docket	Facility Status	NRC Region	NSSS	Utility Response Date	Inspection Report and Date	Closeout Status and Criterion
Yankee-Rowe 1	YAECO	50-029	OL	I	W	09-19-80	81-04(04-09-81)	Closed 3
*Zion 1	CECO	50-295	OL	III	W	09-22-80	80-20(11-28-80) 84-03(05-21-84)	Open
*Zion 2	CECO	50-304	OL	III	W	09-22-80	80-21(11-28-80) 84-03(05-21-84)	Open

Notes for Table B.1:

1. Facility status is based on Reference 1, Page B-6.

2. The following abbreviation applies to facility status: OL, operating license.

3. An asterisk (*) indicates that the facility is listed as a plant of potential concern in the Westinghouse letter (see Page A-8).

4. The following Westinghouse facilities were near to licensing for operation when the bulletin was issued, and are included in Table B.1:

   North Anna 2   McGuire 1   Sequoyah 1
   Diablo Canyon 1   Salem 2

TABLE B.2 LIST OF FACILITIES ISSUED IEB 80-18 FOR INFORMATION

Facility	Utility	Docket	Facility Status	NRC Region	NSSS	Utility Response Date	Inspection Report and Date
Beaver Valley 2	DLC	50-412	CP	I	W		
Bellefonte 1	TVA	50-438	CP	II	B&W		
Bellefonte 2	TVA	50-439	CP	II	B&W		
Braidwood 1	CECO	50-456	CP	III	W	09-16-83	81-13(12-01-81) 82-08(02-07-82)
Braidwood 2	CECO	50-457	CP	III	W	09-16-83	81-13(12-01-81) 82-08(02-07-82)
Byron 1	CECO	50-454	OL	III	W	09-16-83	82-22(11-19-82)
Byron 2	CECO	50-455	CP	III	W	09-16-83	82-16(11-19-82)
Callaway 1	UE	50-483	OL	III	W	09-15-80	82-02(04-05-82)
Catawba 1	DUPCO	50-413	OL	II	W		84-56(07-19-84)
Catawba 2	DUPCO	50-414	OL	II	W		84-26(07-19-84)
Comanche Peak 1	TUGCO	50-445	CP	IV	W		
Comanche Peak 2	TUGCO	50-446	CP	IV	W		
Diablo Canyon 2	PG&E	50-323	OL	V	W	10-16-80	
Farley 2	APCO	50-364	OL	II	W	09-23-80	82-09(04-27-82)
Harris 1	CP&L	50-400	CP	II	W		
McGuire 2	DUPCO	50-370	OL	II	W	09-22-80 09-29-80	
Millstone 3	NNECO	50-423	OL	I	W		
Palo Verde 1	APSCO	50-528	OL	V	C-E		84-22(07-06-84)
Palo Verde 2	APSCO	50-529	OL	V	C-E		84-20(07-06-84)
Palo Verde 3	APSCO	50-530	CP	V	C-E		84-13(07-06-84)
San Onofre 2	SCE	50-361	OL	V	C-E		
San Onofre 3	SCE	50-362	OL	V	C-E		82-23(11-05-82)
Seabrook 1	PSNH	50-443	CP	I	W		
Seabrook 2	PSNH	50-444	CP	I	W		
Sequoyah 2	TVA	50-328	OL	II	W	07-02-81	81-42(10-13-81)

See notes at end of table.

TABLE B.2 (contd)

Facility	Utility	Docket	Facility Status	NRC Region	NSSS	Utility Response Date	Inspection Report and Date
South Texas 1	HL&P	50-498	CP	IV	W		81-32(11-12-81)
South Texas 2	HL&P	50-499	CP	IV	W		81-32(11-12-81)
St. Lucie 2	FPL	50-389	OL	II	C-E		83-11(03-17-83)
Summer 1	SCE&G	50-395	OL	II	W	01-22-82 04-29-87	83-24(08-31-83)
TMI 2	Met-Ed	50-320	SDI	I	B&W	10-07-80	
Vogtle 1	GPC	50-424	CP	II	W		81-15(01-28-82)
Vogtle 2	GPC	50-425	CP	II	W		81-15(01-28-82)
WNP 1	WPPSS	50-460	CP	V	B&W		
WNP 3	WPPSS	50-508	CP	V	C-E		82-25(01-14-83)
Waterford 3	LP&L	50-382	OL	IV	C-E		
Watts Bar 1	TVA	50-390	CP	II	W		85-08(03-28-85)
Watts Bar 2	TVA	50-391	CP	II	W		85-08(03-28-85)
Wolf Creek 1	KG&E	50-482	OL	IV	W	09-15-80	84-01(02-14-84)

Notes for Table B.2:

1. Facility status is based on references 1, 2 and 3, Page B-6.

2. The following abbreviations apply to facility status:
   CP, Construction Permit; OL, Operating License;
   SDI, Shut Down Indefinitely.

REFERENCES

1. United States Nuclear Regulatory Commission, Licensed Operating Reactors, Status Summary Report, Data as of 09-30-86, NUREG-0020, Volume 10, Number 10, October 1986.

2. United States Nuclear Regulatory Commission, Nuclear Power Plants, Construction Status Report, Data as of 06-30-82, NUREG-0030, Volume 6, Number 2, October 1982.

3. United States Nuclear Regulatory Commission, Listing of Inactive Current Holders of Construction Permits, Letter dated May 29, 1985, to Richard A. Lofy (Parameter, Inc.) from Robert L. Baer (NRC/IE HQ).

4. United States Nuclear Regulatory Commission, Code of Federal Regulation, Energy, Title 10, Chapter 1, January 1, 1987, cited as 10CFR 0.735-1.

APPENDIX C

Proposed Followup Items

Region III

Zion 1,2

These are four-loop plants of potential concern as listed by Westinghouse (see Page A-8).

Inspection report 84-03/84-03 leaves the bulletin open pending the Westinghouse permanent solution.

APPENDIX D

Utility Manhours Expended on IEB 80-18

TABLE D.1

Facility	Review & Reporting	Corrective Action	Total	Closeout Status and Criterion
Beaver Valley 1	50	16	66	Closed 2
Calvert Cliffs 1,2	17	0	17	Closed 3
Cook 1,2	200	-	200	Closed 6
Davis-Besse 1	NR*	NR	136	Closed 6
Diablo Canyon 1,2	70	8	78	Closed 2
Fort Calhoun 1	NR	NR	5	Closed 6
Ginna	3	0	3	Closed 5
Indian Point 3	NR	NR	30	Closed 3
Kewaunee	10	0	10	Closed 6
Maine Yankee	15	0	15	Closed 3
McGuire 1,2	52	-	52	Closed 2
North Anna 1,2	120	50	170	Closed 2
Point Beach 1,2	8	0	8	Closed 3
Rancho Seco 1	200	0	200	Closed 4
Robinson 2	55	0	55	Closed 3
Sequoyah 1	55	15	70	Closed 2
St. Lucie 1	4	0	4	Closed 3
TMI 1	75	0	75	Closed 4
Trojan	NR	NR	320	Closed 2
Turkey Point 3,4	NR	NR	5	Closed 3
Zion 1,2	35-50	Several Thousand	35	Open

Total Hours Reported 1554 to 1569

*NR signifies "not reported"

APPENDIX E

Abbreviations

AEPSCO	American Electric Power Services Corporation
APCO	Alabama Power Company
AP&L	Arkansas Power and Light Company
APSCO	Arizona Public Service Company
BG&E	Baltimore Gas and Electric Company
BWST	Borated Water Storage Tank
CCP	Centrifugal Charging Pump
CECO	Commonwealth Edison Company
CFR	Code of Federal Regulations
ConEd	Consolidated Edison Company of New York, Inc.
CP	Construction Permit
CPC	Consumers Power Company
CP&L	Carolina Power and Light Company
CR	Contractor Report
CVCS	Chemical and Volume Control System
CYAPCO	Connecticut Yankee Atomic Power Company
DLC	Duquesne Light Company
DUPCO	Duke Power Company
ECCS	Emergency Core Cooling System
ESF	Engineered Safety Feature
FPC	Florida Power Corporation
FPL	Florida Power & Light Company
FSAR	Final Safety Analysis Report
GAO	Government Accounting Office
GPC	Georgia Power Company
GPUN	GPU Nuclear Corporation
HELB	High Energy Line Break
HL&P	Houston Lighting and Power Company
HPSI	High Pressure Safety Injection
IE	(See NRC/IE)
IEB	Inspection and Enforcement Bulletin (NRC)
IMECO	Indiana and Michigan Electric Company
IR	Inspection Report (NRC/IE)
KG&E	Kansas Gas and Electric Company
LOCA	Loss of Cooling Accident
LP&L	Louisiana Power and Light Company
LPSI	Low Pressure Safety Injection
LPTL	Low Power Testing License

Met-Ed	Metropolitan Edison Company
MYAPCO	Maine Yankee Atomic Power Company
NNECO	Northeast Nuclear Energy Company
NRC/IE	Nuclear Regulatory Commission/ Office of Inspection & Enforcement
NSP	Northern States Power Company
NU	Northeast Utilities
NYPA	New York Power Authority
OL	Operating License
OPPD	Omaha Public Power District
PASNY	Power Authority of the State of New York
PCT	Peak Clad Temperature
PGE	Portland General Electric Company
PG&E	Pacific Gas and Electric Company
PORV	Power-Operated Relief Valve
PSE&G	Public Service Electric and Gas Company
PSNH	Public Service Company of New Hampshire
PWR	Pressurized Water Reactor
R	Region (NRC)
RAS	Recirculation Actuation Signal
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RG&E	Rochester Gas and Electric Corporation
RWST	Refueling Water Storage Tank
SCE	Southern California Edison Company
SCE&G	South Carolina Electric and Gas Company
SI	Safety Injection
SIAS	Safety Injection Actuation Signal
SMUD	Sacramento Municipal Utility District
SNUPPS	Standardized Nuclear Unit Power Plant Systems
TECO	Toledo Edison Company
TMI	Three Mile Island
TUGCO	Texas Utilities Generating Company
TVA	Tennessee Valley Authority
UHI	Upper Head Injection
VCT	Volume Control Tank
UE	Union Electric Company
VEPCO	Virginia Electric and Power Company
W	Westinghouse Electric Corporation
WEPCO	Wisconsin Electric Power Company
WNP	Washington Nuclear Project
WPPSS	Washington Public Power Supply System
WPS	Wisconsin Public Service Corporation
YAECO	Yankee Atomic Electric Company

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12. SUPPLEMENTARY NOTES

13. ABSTRACT (200 words or less)

On May 8, 1980, Westinghouse reported to the NRC/IE per Title 10 CFR Part 21 that one or more centrifugal charging pumps could be damaged by low flow at certain plants before satisfactory termination of safety injection after a secondary side high energy line rupture. The plants of concern had been notified, a calculational method of evaluation and plant-specific reviews had been recommended, and interim modifications had been proposed by Westinghouse. The NRC/IE issued IE Bulletin 80-18 on July 24, 1980 because of this potential safety-related problem. Licensees and near-term licensees of pressurized water reactors (PWRs) were required to take specific actions and submit written responses. Utilities with PWRs under construction were issued the bulletin for information, in preparation for the licensing process. Evaluation of utility responses and NRC/Region inspection reports shows that the bulletin can be closed out per specific criteria for 44 (96%) of the 46 facilities to which it was issued for action. A followup item for the remaining dual plant is proposed for use by the NRC to ensure satisfactory completion of required and corrective actions.

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CLOSEOUT OF IE BULLETIN 80-18: MAINTENANCE OF ADEQUATE MINIMUM FLOW THRU CENTRIFUGAL CHARGING PUMPS FOLLOWING SECONDARY SIDE HIGH ENERGY LINE RUPTURE

JANUARY 1988