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Closeout of IE Bulletin 79-21: Temperature Effects on Level Measurements

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PARAMETER, Inc.

Prepared for
U.S. Nuclear Regulatory Commission

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U.S. Nuclear Regulatory Commission
Washington, DC 20555
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ABSTRACT

On June 22, 1979, Westinghouse Electric Corporation reported to the NRC that elevated containment temperature could affect the reference leg water column and the indicated steam generator water level. IE Bulletin 79-21 was issued by the NRC on August 13, 1979 because of concern that the temperature effect could cause indication of erroneously high steam generator water levels, could delay or prevent protection signals and could cause incorrect information during post-accident monitoring. Because safety-related water level measuring systems used by Babcock & Wilcox and Combustion Engineering could be affected in the same way, the bulletin was issued for action to all utilities with operating pressurized water reactors (PWRs). The bulletin was issued for information to utilities with either PWRs under construction or operating boiling water reactors (BWRs). A related generic letter concerning BWRs was issued by the NRC in July 1979 for information only. Evaluation of licensees' responses and NRC/IE inspection reports shows that the bulletin can be closed out per specific criteria for all of the 41 facilities to which it was issued for action. It is concluded that utility responses were consistent because of guidance from the NSSS suppliers. Remaining areas of concern involve a possible need for manually deenergizing pressurizer heaters in B&W facilities, and further evaluation of boiling in the reference leg by Westinghouse.

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CLOSEOUT OF IE BULLETIN 79-21:
TEMPERATURE EFFECTS ON LEVEL MEASUREMENTS

INTRODUCTION

This report provides documentation for the closeout status of IE Bulletin 79-21 in accordance with the Statement of Work in Task Order 005 under Contract NRC 05-85-157-02. The documentation is based on the records obtained from the IE File and the NRC Document Control System.

IE Bulletin 79-21 was issued August 13, 1979 after Westinghouse had reported to the NRC that elevated containment temperature could affect the reference leg water column and the indicated steam generator water level. Because Babcock & Wilcox and Combustion Engineering PWR systems could be affected similarly, the bulletin was issued for action to all utilities with operating PWRs. The safety concerns were that the temperature effect could cause indication of erroneously high steam generator water levels, could delay or prevent protection signals and could cause incorrect information during post-accident monitoring.

For background information, IE Bulletin 79-21 is included in Appendix A. Evaluation of licensees' responses and NRC/IE inspection reports is documented in Appendix B as the basis for bulletin closeout. Check lists for evaluating utility responses are included for each NSSS supplier. Abbreviations used in this report and associated documents are presented in Appendix C.

SUMMARY

1. The bulletin has been closed out for the following two facilities because they have been shut down indefinitely (Criterion 1):

   Indian Point 1 TMI 2

2. The bulletin has been closed out for Haddam Neck on the basis of acceptable responses which indicate that no corrective action is required (Criterion 2).

3. The bulletin has been closed out for the following 40 facilities on the basis of favorable NRC/IE inspection reports and acceptable responses (Criterion 3):

Arkansas 1,2	Indian Point 2,3	Robinson 2
Beaver Valley 1	Kewaunee	Salem 1
Calvert Cliffs 1,2	Maine Yankee	San Onofre 1
Cook 1,2	Millstone 2	St. Lucie 1
Crystal River 3	North Anna 1	Surry 1,2
Davis-Besse 1	Oconee 1,2,3	TMI 1
Farley 1	Palisades	Trojan
Ft. Calhoun 1	Point Beach 1,2	Turkey Point 3,4
Ginna	Prairie Island 1,2	Yankee-Rowe 1
	Rancho Seco 1	Zion 1,2

4. The bulletin is not called open for any facility.

CONCLUSION

With a few exceptions, the utilities prepared consistent responses because of assistance provided by the three NSSS suppliers. This consistency is indicated by the check lists in tables B.3, B.4 and B.5.

REMAINING AREAS OF CONCERN

The pressurizer heaters at Crystal River 3, Davis-Besse 1, Rancho Seco 1 and TMI 1 may have to be deenergized manually in case level instrumentation is affected significantly by elevated containment temperature (See Note 2, Page B-12). The concern is whether this situation occurs at other facilities.

CRITERIA FOR CLOSEOUT OF THE BULLETIN

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

1. The facility has been shut down indefinitely (SDI).

2. A response for the facility complies with actions required by the bulletin and indicates that no corrective action was necessary.

3. A response for the facility complies with actions required by the bulletin, and an NRC/IE inspection report indicates that all corrective action was completed satisfactorily.

Note: Compliance with required bulletin actions has been evaluated by means of the check lists provided in tables B.3, B.4 and B.5.

APPENDIX A

Background Information

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

TEMPERATURE EFFECTS ON LEVEL MEASUREMENTS

Description of Circumstances:

On June 22, 1979, Westinghouse Electric Corporation reported, to NRC, a potential substantial safety hazard under 10 CFR 21.

The report, Attachment No. 1, addresses the effect of increased containment temperature on the reference leg water column and the resultant effect on the indicated steam generator water level. This effect would cause the indicated steam generator level to be higher than the actual level and could delay or prevent protection signals and could, also, provide erroneous information during post-accident monitoring. Attachment No. 1 addresses only a Westinghouse steam generator reference leg water column; however, safety related liquid level measuring systems utilized on other steam generators and reactor coolant systems could be affected in a similar manner.

Actions To Be Taken By Licensees:

For all pressurized water power reactor facilities with an operating license:*

1. Review the liquid level measuring systems within containment to determine if the signals are used to initiate safety actions or are used to provide post-accident monitoring information. Provide a description of systems that are so employed; a description of the type of reference leg shall be included, i.e., open column or sealed reference leg.

2. On those systems described in Item 1 above, evaluate the effect of post-accident ambient temperatures on the indicated water level to determine any change in indicated level relative to actual water level. This evaluation must include other sources of error including the effects of varying fluid pressure and flashing of reference leg to steam on the water level measurements. The results of this evaluation should be presented in a tabular form similar to Tables 1 and 2 of Attachment 1.

*Boiling water reactors have been requested by a July generic letter from the NRC to provide similar information.

3. Review all safety and control setpoints derived from level signals to verify that the setpoints will initiate the action required by the plant safety analyses throughout the range of ambient temperatures encountered by the instrumentation, including accident temperatures. Provide a listing of these setpoints.

If the above reviews and evaluations require a revision of setpoints to ensure safe operation, provide a description of the corrective action and the date the action was completed. If any corrective action is temporary, submit a description of the proposed final corrective action and a timetable for implementation.

4. Review and revise, as necessary, emergency procedures to include specific information obtained from the review and evaluation of Items 1, 2 and 3 to ensure that the operators are instructed on the potential for and magnitude of erroneous level signals. All tables, curves, or correction factors that would be applied to post-accident monitors should be readily available to the operator. If revisions to procedures are required, provide a completion date for the revisions and a completion date for operator training on the revisions.

A report of the above actions shall be submitted within 30 days of the receipt of this Bulletin.

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

For boiling water reactors with an operating license and all power reactors with a construction permit, this Bulletin is for information purposes and no written response is required.

Approved by GAO, B180225 (R0072); clearance expires 7/31/80. Approval was given under a blanket clearance specifically for identified generic problems.

Attachment:
Memo Westinghouse Electric Corp.
to Victor Stello dated June 22, 1979

ATT/ MENT TO IE BULLETIN 79-21

79-136-000

Westinghouse Electric Corporation Power Systems

June 22, 1979

NS-TMA-2104

Mr. Victor Stello
Director, Office of Inspection and Enforcement
U.S. Nuclear Regulatory Commission
East West Towers Building
4350 East West Highway
Bethesda, Maryland 20014

Dear Mr. Stello:

Subject: Steam Generator Water Level

This is to confirm my telephone conversation of June 21, 1979 with Mr. Norman C. Moseley, Director, Division of Reactor Operation and Inspection and Mr. Samuel E. Bryan, Assistant Director for Field Coordination. In that conversation, I reported that Westinghouse had informed its utility customers of corrections that should be applied to indicated steam generator water level and recommended that they incorporate those corrections in the steam generator low water level protection system setpoints and emergency operating procedures for operating plants as appropriate.

High energy line breaks inside containment can result in heatup of the steam generator level measurement reference leg. Increased reference leg water column temperature will result in a decrease of the water column density with a consequent apparent increase in the indicated steam generator water level (i.e., apparent level exceeding actual level). This potential level bias could result in delayed protection signals (reactor trip and auxiliary feedwater initiation) which are based on low-low steam generator water level. In the case of a feedline rupture, this adverse environment could be present and could delay or prevent the primary signal arising from declining steam generator water level (low-low steam generator level). The following is a list of backup signals available in those Westinghouse plants which take credit in their Final Safety Analysis Reports for steam generator water level trip with an adverse containment environment: overtemperature delta T; high pressurizer pressure; containment pressure and safety injection. For other high energy line breaks which could introduce a similar positive bias to the steam generator water level measurement, steam generator level does not provide the primary trip function and the potential bias would not interfere with needed protective system actuation.

Westinghouse has advised all customers with affected operating plants that the potential temperature-induced bias in indicated level can be compensated for by raising the steam generator low-low water level setpoint. For immediate action, Westinghouse has recommended a change in the allowable water level setpoint sufficient to accommodate the bias (up to 10% of level) which could result from containment temperatures up to 280°F. Containment analyses following a secondary high energy line break on typical plants have shown that a containment high pressure signal would be generated before the containment temperature reaches 280°F. Thus, postulation of all water-level measurement errors occurring simultaneously in the adverse direction results in the containment high pressure signal becoming the primary protective function following some feedline rupture events, i.e., for those cases in which the containment temperature exceeds 280°F before a steam generator low-low water level trip is actuated, the high containment pressure signal provides protection. The combination of the revised low-low water level setpoint and the high containment pressure signal will provide reactor trip and auxiliary feedwater initiation following a feedline rupture and will ensure that the feedline break criteria stated in the Safety Analysis Reports continue to be met. Some applicants may choose to use plant-specific containment analyses, possibly combined with changes in the containment high-pressure setpoint, to justify reducing the bias introduced due to reference leg heatup which must be accommodated in the steam generator low-low water level setpoint.

The potential steam generator level measurement bias also has implications for post-accident monitoring considerations. Since the post-accident environment for high energy line breaks can exceed 280°F, the level bias can exceed the 10% limit which must be considered for protection system actuation. A positive bias of up to 20% can be anticipated for an extreme environmental condition. The appropriate bias must be coupled with instrumentation and other process errors, to determine the required range of indicated level to be maintained during post-accident monitoring to ensure that the steam generator tubes are fully covered and the steam generator is not water solid. Westinghouse has provided all of its customers with operating plants with information to enable them to modify their emergency operating procedures to ensure that suitable steam generator level temperature bias allowance is made.

In a related area, it has been found that a bias in steam generator level may also be introduced by changes in steam generator pressure, due to changes in steam generator fluid densities. Westinghouse has quantified this effect for all of its customers with operating plants. Westinghouse has notified all customers with operating plants that such a bias will exist in the level indication of all steam generators and that the operator should be instructed to monitor steam generator pressure, as well as level, to ensure that the potential bias is reflected in his post-accident recovery actions.

Also, following depressurization of any steam generator, boiling could conceivably occur in the reference leg and cause a major bias for a short time period. Westinghouse has notified all customers with operating plants that the water level indication in the depressurized steam generators may be erroneous due to the potential boiling in the reference leg.

For plants under construction, customers have been advised of the above affects, and the options open to them for corrective action will be reviewed in a timely manner. The NRC will be advised of proposed resolutions for these plants.

The attached tables have been supplied to all customers. They have been informed that we are reporting this to you as a potential substantial safety hazard under 10CFR21 in operating plants and as a significant deficiency under 10CFR50.55(e) for plants under construction.

Should you have any questions on this material, please contact Mr. K. R. Jordan (412/373-4795).

Very truly yours,

Westinghouse Electric Corporation

T. M. Anderson

T. M. Anderson, Manager
Nuclear Safety

JPC:kk

cc: Mr. Norman C. Moseley
Director, DRO&I

Mr. Samuel E. Bryan
Asst. Director, DRO&I

TABLE 1

Correction to indicated steam generator water level for Reference Leg Heatup effects due to post-accident containment temperature (before reactor trip)

Maximum containment temperature reached before reactor trip, °F	Correction to S/G Level, % of Span
90°	0%
200°	4%
280°	10%
320°	13%
400°	20%

BASIS:

Level Calibration Pressure $\leq$ 1000 psia

Reference Leg Calibration Temperature $\geq$ 90°F

Height of Reference Leg $\leq$ 1.1x Level Span

TABLE 2

Corrections to allowable indicated steam generator water level for Reference Leg Heatup and Pressure changes following a high-energy line break, to assure that true level is between the level taps

Containment Temperature °F	Correction To Mininum Allowed Indicated Level, % of Span	Corrections to Maximum Allowed Indicated Level, % of Span
90°	+ 1	- 4
200°	+ 6	- 4
280°	+11	- 4
320°	+14	- 4
400°	+21	- 4

BASIS:

Level Calibration Pressure $\leq$ 1000 psia

Reference Leg Calibration Temperature $\geq$ 90°F

Height of Reference Leg $\leq$ 1.1 x Level Span

Pressure $\geq$ 50 psia

Pressure $\leq$ 200 psi + Calibration Pressure

Boiling in the Reference Leg is not assumed.

APPENDIX B

Documentation of Bulletin Closeout

TABLE B.1 BULLETIN CLOSEOUT STATUS

Facility	Utility	Docket	Facility Status	NRC Region	NSSS Supplier	Utility Response Date	Inspection Report and Date	Closeout Status and Criterion
Arkansas 1	AP&L	50-313	OL	IV	B&W	09-24-79	80-11(08-07-80)	Closed 3
Arkansas 2	AP&L	50-368	OL	IV	C-E	09-24-79	80-11(08-07-80)	Closed 3
Beaver Valley 1	DLC	50-334	OL	I	W	09-18-79 07-24-80 08-14-80	82-26(12-21-82)	Closed 3
Calvert Cliffs 1	BG&E	50-317	OL	I	C-E	09-13-79	82-03(02-24-82)	Closed 3
Calvert Cliffs 2	BG&E	50-318	OL	I	C-E	09-13-79	82-03(02-24-82)	Closed 3
Cook 1	IMECO	50-315	OL	III	W	09-26-79 11-05-79	81-01(03-18-81)	Closed 3
Cook 2	IMECO	50-316	OL	III	W	09-26-79 11-05-79	81-01(03-18-81)	Closed 3
Crystal River 3	FP	50-302	OL	II	B&W	09-17-79 06-06-80	81-07(06-12-81) 81-11(07-28-81)	Closed 3
Davis-Besse 1	TECO	50-346	OL	III	B&W	09-20-79 10-31-79	80-14(06-03-80)	Closed 3
Farley 1	APCO	50-348	OL	II	W	09-18-79 10-16-79 11-01-79	81-07(03-30-81)	Closed 3
Fort Calhoun 1	OPPD	50-285	OL	IV	C-E	09-12-79	83-37(01-10-84)	Closed 3
Ginna	RG&E	50-244	OL	I	W	09-14-79	79-15(11-01-79) 80-14(02-07-81)	Closed 3
Haddam Neck	CYAPCO	50-213	OL	I	W	09-17-79 10-09-79		Closed 2
Indian Point 1	ConEd	50-003	SDI	I	W	09-17-79		Closed 1
Indian Point 2	ConEd	50-247	OL	I	W	09-17-79	83-11(05-11-83)	Closed 3
Indian Point 3	PASNY	50-286	OL	I	W	09-05-79 11-02-79	81-10(09-14-81)	Closed 3
Kewaunee	WPS	50-305	OL	III	W	09-17-79 10-18-79 11-21-79	79-21(12-28-79)	Closed 3

See notes at end of table.

TABLE B.1 (contd)

Facility	Utility	Docket	Facility Status	NRC Region	NSSS Supplier	Utility Response Date	Inspection Report and Date	Closeout Status and Criterion
Maine Yankee	MYAPCO	50-309	OL	I	C-E	09-14-79 08-05-80	80-16(12-05-80)	Closed 3
Millstone 2	NNECO	50-336	OL	I	C-E	09-17-79	80-19(10-27-80)	Closed 3
North Anna 1	VEPCO	50-338	OL	II	W	09-14-79	80-20(05-16-80)	Closed 3
Oconee 1	DUPCO	50-269	OL	II	B&W	09-14-79 12-06-79	79-41(01-11-80) 80-05(02-20-80)	Closed 3
Oconee 2	DUPCO	50-270	OL	II	B&W	09-14-79 12-06-79	79-39(01-11-80) 80-04(02-20-80)	Closed 3
Oconee 3	DUPCO	50-287	OL	II	B&W	09-14-79 12-06-79	79-41(01-11-80) 80-04(02-20-80)	Closed 3
Palisades	CPC	50-255	OL	III	C-E	09-18-79	81-05(04-15-81)	Closed 3
Point Beach 1	WEPCO	50-266	OL	III	W	09-17-79	79-19(01-25-80)	Closed 3
Point Beach 2	WEPCO	50-301	OL	III	W	09-17-79	79-21(01-25-80)	Closed 3
Prairie Island 1	NSP	50-282	OL	III	W	09-14-79	79-30(01-25-80)	Closed 3
Prairie Island 2	NSP	50-306	OL	III	W	09-14-79	79-24(01-25-80)	Closed 3
Rancho Seco 1	SMUD	50-312	OL	V	B&W	09-14-79	79-20(11-15-79)	Closed 3
Robinson 2	CP&L	50-261	OL	II	W	09-14-79	81-15(05-26-81)	Closed 3
Salem 1	PSE&G	50-272	OL	I	W	10-05-79	80-32(01-20-81)	Closed 3
San Onofre 1	SCE	50-206	OL	V	W	09-14-79 09-28-79	79-17(01-18-80)	Closed 3
St. Lucie 1	FPL	50-335	OL	II	C-E	09-18-79	79-32(01-03-80)	Closed 3
Surry 1	VEPCO	50-280	OL	II	W	09-14-79	84-10(02-01-85)	Closed 3
Surry 2	VEPCO	50-281	OL	II	W	09-14-79	84-10(02-01-85)	Closed 3
TMI 1	Met-Ed	50-289	OL	I	B&W	12-03-79 01-09-80 10-04-82	83-06(03-29-83)	Closed 3
TMI 2	Met-Ed	50-320	SDI	I	B&W	08-29-80		Closed 1
Trojan	PGE	50-344	OL	V	W	09-14-79	79-22(12-13-79)	Closed 3
Turkey Point 3	FPL	50-250	OL	II	W	09-18-79	80-16(05-29-80) 84-18(07-12-84)	Closed 3

See notes at end of table.

TABLE B.1 (contd)

Facility	Utility	Docket	Facility Status	NRC Region	NSSS Supplier	Utility Response Date	Inspection Report and Date	Closeout Status and Criterion
Turkey Point 4	FPL	50-251	OL	II	W	09-18-79	80-14(05-29-80) 84-18(07-12-84)	Closed 3
Yankee-Rowe 1	YAECO	50-029	OL	I	W	10-09-79	81-21(01-18-82)	Closed 3
Zion 1	CECO	50-295	OL	III	W	09-21-79 12-14-79	79-18(09-12-79)	Closed 3
Zion 2	CECO	50-304	OL	III	W	09-21-79 12-14-79	79-17(09-12 79)	Closed 3

Notes for Table B.1:

1. Facility status is based on Reference 1, Page B-29.

2. The following abbreviations apply to facility status:

   OL, Operating License
   SDI, Shut Down Indefinitely

3. Only facilities (PWRs with OLs) to which the bulletin was issued for action are included.

4. Current Facilities Grouped per NSSS Supplier

   B&W: Arkansas 1; Crystal River 3; Davis-Besse 1; Oconee 1,2,3; Rancho Seco 1; TMI 1.

   C-E: Arkansas 2; Calvert Cliffs 1,2; Fort Calhoun 1; Maine Yankee; Millstone 2; Palisades; St. Lucie 1.

   W: Beaver Valley 1; Cook 1,2; Farley 1; Ginna; Haddam Neck; Indian Point 2,3; Kewaunee; North Anna 1; Point Beach 1,2; Prairie Island 1,2; Robinson 2; Salem 1; San Onofre 1; Surry 1,2; Trojan; Turkey Point 3,4; Yankee Rowe 1; Zion 1,2.

TABLE B.2 LIST OF FACILITIES ISSUED IEB 79-21 FOR INFORMATION

Facility	Utility	Docket	Facil-ity Status	NRC Region	NSSS Type and Supplier	Utility Response Date	Inspection Report Number (Date)
Bailly 1	NIPSCO	50-367	CD	III			
Beaver Valley 2	DLC	50-412	CP	I	PWR/W		
Bellefonte 1	TVA	50-438	CP	II	PWR/B&W		80-02(02-05-80)
Bellefonte 2	TVA	50-439	CP	II	PWR/B&W		80-02(02-05-80)
Big Rock Point 1	CPC	50-155	OL	III	BWR/GE		82-03(03-30-82)
Braidwood 1	CECO	50-456	CP	III	PWR/W	02-09-84	81-13(12-01-81)
Braidwood 2	CECO	50-457	CP	III	PWR/W	02-09-84	81-13(12-01-81)
Browns Ferry 1	TVA	50-259	OL	II	BWR/GE		
Browns Ferry 2	TVA	50-260	OL	II	BWR/GE		
Browns Ferry 3	TVA	50-296	OL	II	BWR/GE		
Brunswick 1	CP&L	50-325	OL	II	BWR/GE		80-37(11-19-80)
Brunswick 2	CP&L	50-324	OL	II	BWR/GE		80-40(11-19-80)
Byron 1	CECO	50-454	OL	III	PWR/W	02-09-84	83-07(03-22-83)
Byron 2	CECO	50-455	CP	III	PWR/W	02-09-84	83-03(03-22-83)
Callaway 1	UE	50-483	OL	III	PWR/W		82-13(10-27-82)
Callaway 2	UE	50-486	CD	III			
Catawba 1	DUPCO	50-413	OL	II	PWR/W		82-30(12-03-82)
Catawba 2	DUPCO	50-414	LPTL	II	PWR/W		82-28(12-03-82)
Cherokee 1	DUPCO	50-491	CD	II			
Cherokee 2	DUPCO	50-492	CD	II			
Cherokee 3	DUPCO	50-493	CD	II			
Clinton 1	IP	50-461	CP	III	BWR/GE		80-04(04-21-80)
Clinton 2	IP	50-462	CHI	III			
Comanche Peak 1	TUGCO	50-445	CP	IV	PWR/W		
Comanche Peak 2	TUGCO	50-446	CP	IV	PWR/W		
Cooper Station	NPPD	50-298	OL	IV	BWR/GE		
Diablo Canyon 1	PG&E	50-275	OL	V	PWR/W	01-22-80	80-02(03-04-80)
Diablo Canyon 2	PG&E	50-323	OL	V	PWR/W	01-22-80	80-01(03-04-80)

See notes at end of table.

TABLE B.2 (contd)

Facility	Utility	Docket	Facility Status	NRC Region	NSSS Type and Supplier	Utility Response Date	Inspection Report Number (Date)
Dresden 1	CECO	50-010	SDI	III			
Dresden 2	CECO	50-237	OL	III	BWR/GE		80-17(09-29-80)
Dresden 3	CECO	50-249	OL	III	BWR/GE		80-21(09-29-80)
Duane Arnold	IELPCO	50-331	OL	III	BWR/GE		
Farley 2	APCO	50-364	OL	II	PWR/W	09-18-79	80-11(05-21-80)
						10-16-79	80-41(11-03-80)
						11-01-79	81-10(03-30-81)
Fermi 2	DECO	50-341	OL	III	BWR/GE		
FitzPatrick	PASNY	50-333	OL	I	BWR/GE		
Forked River	JCP&L	50-363	CD	I			
Fort St. Vrain	PSCC	50-267	OL	IV			
Grand Gulf 1	MP&L	50-416	OL	II	BWR/GE		
Grand Gulf 2	MP&L	50-417	CHI	II			
Harris 1	CP&L	50-400	CP	II	PWR/W		84-14(05-03-84)
Harris 2	CP&L	50-401	CHI	II			
Harris 3	CP&L	50-402	CHI	II			
Harris 4	CP&L	50-403	CHI	II			
Hartsville A-1	TVA	50-518	CD	II			
Hartsville A-2	TVA	50-519	CD	II			
Hartsville B-1	TVA	50-520	CD	II			
Hartsville B-2	TVA	50-521	CD	II			
Hatch 1	GPC	50-321	OL	II	BWR/GE		81-31(12-03-81)
Hatch 2	GPC	50-366	OL	II	BWR/GE		81-31(12-03-81)
Hope Creek 1	PSE&G	50-354	CP	I	BWR/GE		82-01(02-11-82)
							85-14(05-02-85)
Hope Creek 2	PSE&G	50-355	CHI	I			82-01(02-11-82)
Humboldt Bay 3	PG&E	50-133	SDI	V			
Jamesport 1	LILCO	50-516	CD	I			
Jamesport 2	LILCO	50-517	CD	I			

See notes at end of table.

TABLE B.2 (contd)

Facility	Utility	Docket	Facility Status	NRC Region	NSSS Type and Supplier	Utility Response Date	Inspection Report Number (Date)
La Crosse	DPC	50-409	OL	III	BWR/Allis-Chalmers		
LaSalle 1	CECO	50-373	OL	III	BWR/GE		79-38(11-30-79)
LaSalle 2	CECO	50-374	OL	III	BWR/GE		
Limerick 1	PECO	50-352	OL	I	BWR/GE		82-03(02-08-82)
Limerick 2	PECO	50-353	CP	I	BWR/GE		82-02(02-08-82)
Marble Hill 1	PSI	50-546	CHI	III			82-05(04-29-82) 83-14(09-08-83)
Marble Hill 2	PSI	50-547	CHI	III			82-05(04-29-82) 83-14(09-08-83)
McGuire 1	DUPCO	50-369	OL	II	PWR/W	05-07-80	
McGuire 2	DUPCO	50-370	OL	II	PWR/W	05-07-80	
Midland 1	CPC	50-329	CHI	III		04-15-82 09-28-82	81-03(05-13-81)
Midland 2	CPC	50-330	CHI	III		04-15-82 09-28-82	81-03(05-13-81)
Millstone 1	NNECO	50-245	OL	I	BWR/GE		80-17(10-27-80)
Millstone 3	NNECO	50-423	OL	I	PWR/W		81-04(08-17-81) 82-02(03-11-82)
Monticello	NSP	50-263	OL	III	BWR/GE		84-04(04-26-84)
Nine Mile Point 1	NMP	50-220	OL	I	BWR/GE		
Nine Mile Point 2	NMP	50-410	CP	I	BWR/GE		81-14(02-08-82)
North Anna 2	VEPCO	50-339	OL	II	PWR/W	09-14-79 05-05-80	80-21(05-16-80)
North Anna 3	VEPCO	50-404	CD	II			
North Anna 4	VEPCO	50-405	CD	II			
Oyster Creek 1	JCP&L	50-219	OL	I	BWR/GE		
Palo Verde 1	APSCO	50-528	OL	V	PWR/C-E		83-08(03-30-83) 84-51(01-28-85) 85-18(06-28-85)

See notes at end of table.

TABLE B.2 (contd)

Facility	Utility	Docket	Facility Status	NRC Region	NSSS Type and Supplier	Utility Response Date	Inspection Report Number (Date)
Palo Verde 2	APSCO	50-529	LPTL	V	PWR/C-E		83-04(03-30-83)
							84-35(01-28-85)
							85-20(06-28-85)
Palo Verde 3	APSCO	50-530	CP	V	PWR/C-E		83-03(03-30-83)
							85-14(06-28-85)
Peach Bottom 2	PECO	50-277	OL	I	BWR/GE		
Peach Bottom 3	PECO	50-278	OL	I	BWR/GE		
Perkins 1	DUPCO	50-488	CD	II			
Perkins 2	DUPCO	50-489	CD	II			
Perkins 3	DUPCO	50-490	CD	II			
Perry 1	CEI	50-440	LPTL	III	BWR/GE		82-14(11-09-82)
Perry 2	CEI	50-441	CP	III	BWR/GE		82-13(11-09-82)
Phipps Bend 1	TVA	50-553	CD	II			
Phipps Bend 2	TVA	50-554	CD	II			
Pilgrim 1	BECO	50-293	OL	I	BWR/GE		
Quad Cities 1	CECO	50-254	OL	III	BWR/GE		84-14(09-14-84)
Quad Cities 2	CECO	50-265	OL	III	BWR/GE		84-12(09-14-84)
River Bend 1	GSU	50-458	OL	IV	BWR/GE		85-54(10-04-85)
River Bend 2	GSU	50-459	CD	IV			
Salem 2	PSE&G	50-311	OL	I	PWR/W	10-05-79	79-35(12-13-79)
							9-37(02-20-80)
							80-03(04-23-80)
San Onofre 2	SCE	50-361	OL	V	PWR/C-E		
San Onofre 3	SCE	50-362	OL	V	PWR/C-E		83-01(02-15-83)
Seabrook 1	PSNH	50-443	CP	I	PWR/W		85-20(09-12-85)
Seabrook 2	PSNH	50-444	CP	I	PWR/W		85-20(09-12-85)
Sequoyah 1	TVA	50-327	OL	II	PWR/W		79-36(08-02-79)
Sequoyah 2	TVA	50-328	OL	II	PWR/W		79-21(08-02-79)
							80-03(02-27-80)

See notes at end of table.

TABLE B.2 (contd)

Facility	Utility	Docket	Facility Status	NRC Region	NSSS Type and Supplier	Utility Response Date	Inspection Report Number (Date)
Shoreham	LILCO	50-322	LPTL	I	BWR/GE		81-01(02-23-81) 83-05(03-17-83) 83-35(12-12-83)
South Texas 1	HL&P	50-498	CP	IV	PWR/W		
South Texas 2	HL&P	50-499	CP	IV	PWR/W		
St. Lucie 2	FPL	50-389	OL	II	PWR/C-E		83-02(01-27-83)
Sterling	RG&E	50-485	CD	I			
Summer 1	SCE&G	50-395	OL	II	PWR/W		
Susquehanna 1	PP&L	50-387	OL	I	BWR/GE		
Susquehanna 2	PP&L	50-388	OL	I	BWR/GE		
Tyrone	NSP	50-484	CD	III			
Vermont Yankee 1	VYNP	50-271	OL	I	BWR/GE		
Vogtle 1	GP	50-424	CP	II	PWR/W		
Vogtle 2	GP	50-425	CP	II	PWR/W		
WNP 1	WPPSS	50-460	CP	V	PWR/B&W		86-01(07-07-86)
WNP 2	WPPSS	50-397	OL	V	BWR/GE		80-06(05-09-80)
WNP 3	WPPSS	50-508	CP	V	PWR/C-E		
WNP 4	WPPSS	50-513	CHI	V			
WNP 5	WPPSS	50-509	CHI	V			
Waterford 3	LP&L	50-382	OL	IV	PWR/C-E		
Watts Bar 1	TVA	50-390	CP	II	PWR/W		80-06(03-21-80) 85-08(03-28-85)
Watts Bar 2	TVA	50-391	CP	II	PWR/W		80-05(03-21-80) 85-08(03-28-85)
Wolf Creek 1	KG&E	50-482	OL	IV	PWR/W		84-01(02-14-84) 84-44(01-11-85)
Yellow Creek 1	TVA	50-566	CHI	II			
Yellow Creek 2	TVA	50-567	CHI	II			
Zimmer 1	CG&E	50-358	CD	III			

See notes at end of table.

Notes for Table B.2:

1. Facility status is based on References 1, 2 and 3, Page B-29.

2. The following abbreviations apply to facility status:

   CD, Cancelled
   CHI, Construction Halted Indefinitely
   CP, Construction Permit
   LPTL, Low Power Testing License
   OL, Operating License

3. The NRC generic letter of July 1979 was issued for information to all BWRs.

TABLE B.3 CHECK LIST OF BULLETIN ACTIONS FOR B&W FACILITIES

ACTION 1

Facility	Components Generating Level Signals: Initiation of Safety Actions	Components Generating Level Signals: Post-Accident Monitoring (PAM)
Arkansas 1	None	S/G, PZR
Crystal River 3	None	S/G, PZR
Davis-Besse 1	S/G	S/G, PZR
Oconee 1,2,3	None	S/G, PZR
Rancho Seco 1	None	S/G, PZR
TMI 1	None	S/G, PZR

Notes: 1. All of these facilities have delta pressure measurements for water level indication using uninsulated, open column reference legs.

2. Only level measuring systems affected by increased containment temperature are included in this table.

3. During PAM, level indication alone is not relied upon; system temperatures and pressures are used for this function.

4. All responses for B&W facilities comply with the requirements of Action 1.

5. For the requirements of Action 1, see Page A-1.

TABLE B.3 (contd)

ACTION 2

Facility(B&W)	Evaluation of Post-Accident Effects: Varying Fluid Pressure	Evaluation of Post-Accident Effects: Flashing in Reference Leg	Presentation of Level Corrections: Tables	Presentation of Level Corrections: Curves
Arkansas 1	Yes	Yes	Yes	No
Crystal River 3	Yes	Yes	Yes	No
Davis-Besse 1	Yes	Yes	Yes	No
Oconee 1,2,3	Yes	Yes	Yes	No
Rancho Seco	Yes	Yes	Yes	No
TMI 1	Yes	Yes	Yes	No

Notes: 1. "Careful consideration" of reference leg flashing is taken to be equivalent to "evaluation".

2. All responses for B&W facilities comply with the requirements of Action 2.

3. For the requirements of Action 2, see Page A-1.

TABLE B.3 (contd)

ACTION 3

Facility(B&W)	Functions of Setpoints Derived from Level Signals		Listing of Setpoints		Notes
	Initiation of Safety Actions	Level Control or PAM	Initiation of Safety Actions	Level Control or PAM	
Arkansas 1	No	Yes	No	Yes	1
Crystal River 3	No	Yes	No	Yes	1,2
Davis-Besse 1	Yes	Yes	No(Note 4)	Yes	2,3,4
Oconee 1,2,3	No	Yes	No	Yes	1
Rancho Seco 1	No	Yes	No	Yes	1,2
TMI 1	No	Yes	No	Yes	1,2

Notes: 1. No level signals affected by containment temperature are used for initiation of safety actions; therefore, no corrective action is needed to satisfy bulletin requirements.

2. "Although not related to RPS actions, the pressurizer level instrumentation is used to deenergize the pressurizer heaters and therefore this action may have to be taken manually in the event of elevated containment temperatures."

Refer to the response of 09-17-79 for Crystal River 3 (FP), 09-20-79 for Davis-Besse 1, 09-14-79 for Rancho Seco 1 (SMUD) and 01-09-80 for TMI 1 (Met-Ed). The quotation is taken from the response for Crystal River 3, but it applies to all four of these responses.

3. At Davis-Besse 1 (response of 09-20-79), in the case of a steam line break inside of containment, the unaffected steam generator would be used to remove decay heat from the Reactor Coolant system, and its start up level instrumentation would be affected. Control set points were reviewed with consideration of this accident condition.

Notes continued on next page.

Notes for Action 3, Table B.3 (contd)

4. For Davis-Besse 1 (response of 09-20-79), the S/G safety-related level signals apply only to start-up when containment temperature is normal.

5. During PAM, level indication alone is not relied upon; system temperatures and pressures are used for this function.

6. All responses for B&W facilities comply with the requirements of Action 3.

7. For the requirements of Action 3, see Page A-1.

TABLE B.3 (contd)

ACTION 4	
Facility(B&W)	Licensee Commitments for Procedures and Training
Arkansas 1	According to the AP&L response of September 24, 1979, correction factors and training for post-accident monitoring were to be made available to operators by November 15, 1979.
Crystal River 3	According to the FP response of June 6, 1980, the procedure revisions and operator training required for post-accident monitoring were to be completed before startup.
Davis Besse 1	According to the TECO response of October 31, 1979, revisions of emergency procedures and operator training for level controls were to be completed by November 30, 1979.
Oconee 1,2,3	According to the DUPCO response of December 6, 1979, all required procedural revisions and necessary operator training were to be completed no later than December 6, 1979.
Rancho Seco 1	According to the SMUD response of September 14, 1979, "the District does not feel that revisions to procedures are required as a result of this evaluation, however, plant operators will be informed on the possible level indicator error".
TMI 1	On October 4, 1982, the Licensee provided five reasons why tables, curves or correction factors had not been placed into the applicable emergency procedures, and concluded that the operators had been trained to cope safely with potentially erroneous level indications.

Note: 1. All responses for B&W facilities comply with the intent of Action 4 requirements.

2. For the requirements of Action 4, see Page A-2.

TABLE B.4 CHECK LIST OF BULLETIN ACTIONS FOR C-E FACILITIES

ACTION 1

Facility	Components Generating Level Signals: Initiation of Safety Actions	Components Generating Level Signals: Post-Accident Monitoring(PAM)	Description of Reference Legs
Arkansas 2	S/G	S/G, PZR	Uninsulated, open column
Calvert Cliffs 1,2	S/G	S/G, PZR	Uninsulated, open column, condensate pot
Fort Calhoun 1	S/G	S/G	Completely sealed column
Maine Yankee	S/G	PZR	Vented column
Millstone 2	S/G	S/G, PZR	Uninsulated, open column, condensate pot
Palisades	S/G	S/G, PZR	Wet column
St. Lucie 1	S/G	S/G, PZR	Open column, condensate pot

Notes: 1. Only level measuring systems affected by increased containment temperature are included in this table.

2. All responses for C-E facilities comply with the requirements of Action 1.

3. For the requirements of Action 1, see Page A-1.

TABLE B.4 (contd)

ACTION 2

Facility(C-E)	Evaluation of Post-Accident Effects: Varying Fluid Pressure	Evaluation of Post-Accident Effects: Flashing in Reference Leg	Presentation of Level Corrections: Tables	Presentation of Level Corrections: Curves
Arkansas 2	Yes	Yes	Yes	No
Calvert Cliffs 1,2	Yes	Yes	No	Yes
Fort Calhoun 1	Yes	Yes	No	Yes
Maine Yankee	Note 2	Note 2	Yes	No
Millstone 2	Yes	Yes	No	Yes
Palisades	Yes	Note 2	Yes	Yes
St. Lucie 1	Yes	Yes	Yes	Yes

Notes: 1. "Careful consideration" of reference leg flashing is taken to be equivalent to "evaluation". Compliance with this bulletin requirement is implied in the responses of 09-13-79 for Calvert Cliffs (BG&E), 09-14-79 for Maine Yankee (MYAPCO) and 09-18-79 for Palisades (CPC).

2. The closing inspection reports for Maine Yankee (80-16, 2-5-80) and Palisades (81-05, 4-15-81) indicate all items of the bulletin were addressed adequately.

3. All responses for C-E facilities comply with the requirements of Action 2.

4. For the requirements of Action 2, see Page A-1.

TABLE B.4 (contd)

ACTION 3

Facility(C-E)	Functions of Setpoints Derived from Level Signals: Initiation of Safety Actions	Functions of Setpoints Derived from Level Signals: Level Control or PAM	Listing of Setpoints: Initiation of Safety Actions	Listing of Setpoints: Level Control or PAM	Notes
Arkansas 2	Yes	Yes	Yes	Yes	1,2
Calvert Cliffs 1,2	Yes	Yes	Yes	Yes	2
Fort Calhoun 1	Yes	Yes	Yes	Yes	2
Maine Yankee	Yes	Yes	Yes	Yes	
Millstone 2	Yes	Yes	Yes	Yes	2
Palisades	Yes	Yes	Yes	Yes	2
St. Lucie 1	Yes	Yes	Yes	Yes	2

Notes: 1. Ambient temperatures effects on setpoints for Arkansas 2 were considered in Section 2.3.2.5 of CEN-98(A)-P, which was submitted to J. Stolz (NRR/DE) per the AP&L letter of 02-28-79.

2. According to the response, no revision of setpoints was required.

3. All responses for C-E facilities comply with the requirements of Action 3.

4. For the requirements of Action 3, see Page A-1.

TABLE B.4 (contd)

ACTION 4	
Facility(C-E)	Licensee Commitments for Procedures and Training
Arkansas 2	Per the AP&L response of September 24, 1979, new factors and instructions for post-accident monitoring were to be made available to operators by November 15, 1979.
Calvert Cliffs 1,2	Per the BG&E response of September 13, 1979, any revisions to procedures and operator training required to correct for the effect of increased containment temperature were to be completed by November 30, 1979.
Fort Calhoun 1	Per the OPPD response of September 12, 1979, revisions to procedures and operator training required for post-accident monitoring were to be completed by November 14, 1979.
Maine Yankee	Per the MYAPCO response of August 4, 1980, revisions to procedures and operator training required to correct for temperature effects were to be completed by September 1, 1980.
Millstone 2	Per the NNECO response of September 17, 1979, revisions to procedures and operator training for post-accident monitoring were to be completed by January 18,1980.
Palisades	Per the CPC response of September 18, 1979, procedures were to be revised by November 1, 1979; operators were to be trained during the next regularly scheduled 5th Shift Training Cycle.
St. Lucie 1	Per the FPL response of September 18, 1979, procedures were to be revised by October 31, 1979; operator training was to be completed by November 30, 1979.

Notes: 1. All responses for C-E facilities comply with the requirements of Action 4.

2. For the requirements of Action 4, see Page A-2.

TABLE B.5 CHECK LIST OF BULLETIN ACTIONS FOR WESTINGHOUSE FACILITIES

ACTION 1

Facility	Components Generating Level Signals: Initiation of Safety Actions	Components Generating Level Signals: Post-Accident Monitoring(PAM)	Description of Reference Legs
Beaver Valley 1	S/G, PZR	S/G, PZR	S/G: Insulated, open column PZR: Insulated, sealed column
Cook 1,2	S/G, PZR	S/G, PZR	S/G: Condensing pot, open column PZR: Sealed bellows, open column
Farley 1	S/G, PZR	S/G, PZR	S/G: Insulated, open column; PZR: Sealed column
Ginna	S/G, PZR	S/G, PZR	S/G: Open column PZR: Open column, Sealed column
Haddam Neck	S/G, PZR	S/G, PZR	Open column
Indian Point 2	S/G, PZR	S/G, PZR	S/G: Open column PZR: Open column, Sealed column
Indian Point 3	S/G, PZR	S/G, PZR	S/G: Open column: PZR: Sealed column
Kewaunee	S/G, PZR	S/G, PZR	S/G: Open column; PZR: Sealed column
North Anna 1	S/G	S/G	S/G: Open column; PZR: Sealed column
Point Beach 1,2	S/G, PZR	S/G, PZR	S/G: Condensing pot, Open column PZR: Sealed column
Prairie Island 1,2	S/G, PZR	S/G, PZR	S/G: Condensate pot, Open column PZR: Sealed column

See notes at end of Action 1 of Table B.5.

TABLE B.5 (contd)

ACTION 1 (contd)			
	Components Generating Level Signals		
Facility(W)	Initiation of Safety Actions	Post-Accident Monitoring(PAM)	Description of Reference Legs
Robinson 2	S/G	S/G, PZR	Open column
Salem 1	S/G, PZR	S/G, PZR	S/G: Open column; PZR: Sealed column
San Onofre 1	PZR	S/G, PZR	Condensate pots, open columns
Surry 1,2	S/G, PZR	S/G, PZR	S/G: Open column; PZR: Sealed column
Trojan	S/G, PZR	S/G, PZR	S/G: Condensing pot, open column PZR: Condensing pot, sealed column
Turkey Point 3,4	S/G, PZR	S/G, PZR	S/G: Condensing pot, open column PZR: Sealed columns
Yankee-Rowe 1	S/G, PZR	S/G, PZR	Open columns
Zion 1,2	S/G, PZR	S/G, PZR	S/G: Open column; PZR: Sealed column

Notes: 1. At Beaver Valley 1, insulation was added to the reference leg column per instructions by W. Refer to the DLC response of 08-14-80.

2. All responses for W facilities comply with the requirements of Action 1.

3. For the requirements of Action 1, see Page A-1.

TABLE B.5 (contd)

ACTION 2					
	Evaluation of Post-Accident Effects		Presentation of Level Corrections		
Facility(W)	Varying Fluid Pressure	Flashing in Reference Leg	Tables	Curves	Notes
Beaver Valley 1	Yes	Yes	No	Yes	
Cook 1,2,	Yes	Yes	Yes	No	
Farley 1	Yes	Yes	Yes	No	
Ginna	Yes	Yes	Yes	Yes	
Haddam Neck	Yes	Yes	Yes	No	
Indian Point 2	Yes	Yes	Yes	No	
Indian Point 3	Yes	Yes	Yes	No	
Kewaunee	Yes	Yes	Yes	Yes	
North Anna 1	No	No	No	No	3
Point Beach 1,2	Yes	Yes	Yes	No	
Prairie Island 1,2	Yes	Yes	Yes	Yes	
Robinson 2	Yes	Yes	Yes	No	
Salem 1	Yes	Yes	Yes	Yes	
San Onofre 1	Yes	Yes	Yes	No	
Surry 1,2	Yes	Yes	Yes	No	
Trojan	Yes	Yes	Yes	Yes	
Turkey Point 3,4	Yes	Yes	Yes	Yes	
Yankee-Rowe 1	No	No	No	No	1
Zion 1,2	Yes	Yes	Yes	No	

Notes: 1. The YAECO response of 10-09-79 for Yankee-Rowe 1 indicates that post-accident monitoring is not affected significantly.

2. "Careful consideration" of reference leg flashing is taken to be equivalent to "evaluation".

Notes continued on next page.

Notes for Action 2, Table B.5 (contd)

3. Inspection reports for North Anna 1 indicate adequate attention to bulletin action requirements (see IR 80-20).

4. Requirements of Action 2 are met for all W facilities.

5. For the requirements of Action 2, see Page A-1.

TABLE B.5 (contd)

ACTION 3

Facility(W)	Review of Safety and Control Setpoints
Beaver Valley 1	The only S/G safety setpoint was raised to 12% to provide for feedline rupture, after insulating the reference legs. The PZR trip setpoint was not revised, because it was not needed to control rupture of a high energy line inside containment. Refer to the DLC responses of 09-18-79, 07-24-80 and 08-14-80.
Cook 1,2	The only S/G safety setpoints were raised from 11% to 15% for unit 1 and from 17% to 21% for unit 2, in order to provide for reaching 200 F before arriving at the containment high pressure setpoint. Because PZR setpoints were not needed for safety functions, they were not changed; however, pressure limits were established to accommodate the level bias of concern. Refer to the IMECO response of 11-05-79.
Farley 1	The only S/G low-low level safety setpoint was raised from 15% to 17% and the allowable values were increased from 14% to 16%, after insulating the reference legs. Although the PZR high level trip ensures protection against RCS pressurization, no credit is taken for this trip in the safety analysis. Refer to the APCO response of 11-01-79.

See notes at end of Action 3 of Table B.5.

TABLE B.5 (contd)

ACTION 3 (contd)

Facility(W)	Review of Safety and Control Setpoints
Ginna	Of the safety and control setpoints listed in table 5 of the RG&E response of 09-14-79, only two of the five setpoints required revision. The reactor trip setpoint on low-low S/G level was to be changed from 15% to $\geq$ 13%. The reactor trip setpoint on PZR high water level was to be decreased from 91% to $\leq$ 88%.
Haddam Neck	According to the response of 09-17-79, "CYAPCO has reviewed all safety and control setpoints derived from level measuring devices and determined that no revisions are necessary...." The bases for this statement were presented.
Indian Point 2	In the response of 09-17-79, Con Ed provided reasons why no changes in safety and control setpoints for the S/Gs and the PZR were necessary.
Indian Point 3	In the response of 11-02-79, PASNY provided reasons why no changes in safety and control setpoints for the S/Gs were necessary.
Kewaunee	In the responses of 09-17-79, 10-18-79 and 11-21-79, WPS provided a review of S/G and PZR setpoints which indicated that only one setpoint required revision. The S/G narrow range low-low reactor trip setpoint was to be raised to 17%, in order to allow for uncertainties and to avoid violation of the Technical Specification limit of 5%.
North Anna 1	The only S/G safety setpoint was raised from 5% to 15%. No PZR setpoints needed to be revised, because they were not included in safety analysis. Refer to VEPCO response of 09-14-79.

See notes at end of Action 3 of Table B.5.

TABLE B.5 (contd)

ACTION 3 (contd) Facility(W)	Review of Safety and Control Setpoints
Point Beach 1,2	Safety and control setpoints are listed for the S/Gs and the PZR in tables 5 and 6, respectively, of WEPCO response of 09-17-79. A review was presented to show that only the S/G low-low level reactor trip and auxiliary feedwater system startup setpoints needed to be revised.
Prairie Island 1,2	In the response of 09-14-79, NSP provided reasons why no changes in safety and control setpoints for the S/Gs and the PZR were needed.
Robinson 2	According to the CP&L response of 09-14-79, a Technical Specification change request had been initiated to increase the S/G low-low level setpoint to 14%, with the Plant Operating Manual remaining at 15% This was the only instrument system used for initiating action required by the Safety Analysis.
Salem 1	The review of safety and control setpoints presented by PSE&G in the response of 10-05-79 indicated that only the low-low level setpoint of the S/Gs was affected. The setpoint was to be increased from 5% to 11%.
San Onofre 1	The review of safety and control setpoints presented by SCE in the response of 09-14-79 indicated that only the PZR high level trip had to be considered. Because heatup of the reference leg would result in a conservative action by the RPS for the high level trip, no change in setpoint was to be made.
Surry 1,2	According to the VEPCO resonse of 09-14-79, setpoints had to be changed only for S/G low-low level trip. The setpoints were increased to 15% for Unit 1 and 25% for Unit 2. Because no safety credit is taken for PZR trip setpoint, no change of this setpoint is needed.

See notes at end of Action 3 of Table B.5.

TABLE B.5 (contd)

ACTION 3 (contd)	
Facility(W)	Review of Safety and Control Setpoints
Trojan	According to the PGE response of 09-14-79, the only setpoint which needed to be changed was for low-low S/G level. The affected setpoint was increased from 5% to 15% as an interim measure. W was investigating design changes of the S/G level instrument to permit returning the setpoint to its previous value.
Turkey Point 3,4	According to the FPL response of 09-18-79, only the S/G setpoint for narrow range water level trip initiated action required by plant safety analysis. The affected setpoint was increased from 5% to 15%. In safety analysis for PZR level function, no credit is taken.
Yankee-Rowe 1	The review presented by YAECO in the response of 10-09-79 indicated that the bulletin concerns about safety and control setpoints do not apply to the Yankee-Rowe 1 design.
Zion 1,2	Per the CECO response of 09-21-79, the S/G setpoint for reactor trip and auxiliary feedwater actuation was raised from 10% to 15%.

Notes: 1. Setpoints were revised at all Westighouse facilities except Haddam Neck, Indian Point 2 & 3, Prairie Island 1 & 2 and Yankee-Rowe 1.

2. Setpoints were to be revised or were being evaluated at Robinson 2 and Salem 1.

3. All responses for W facilities comply with the requirements of Action 3.

4. For the requirements of Action 3, see Page A-1.

TABLE B.5 (contd)

ACTION 4	
Facility(W)	Licensee Commitments for Procedures and Training
Beaver Valley 1	Per the DLC response of July 24, 1980, revision of procedures and training of operators had been completed.
Cook 1,2	Per the IMECO response of November 5, 1979, the existing procedures did not need to be revised, and operators had "been informed of the potential for non-conservative bias in indicated water level due to increased containment temperature".
Farley 1	Per the APCO response of November 1,1979, procedures were to be revised and graphs or curves were to be provided by December 15, 1979; in addition, operator training was to be completed by January 15, 1980.
Ginna	Per the RG&E response of September 14, 1979, tables, curves and caution notes were to be incorporated in the appropriate procedure by 9/24/79 and operator training was to be completed by 10/31/79.
Haddam Neck	Per the CYAPCO response of September 17, 1979, all emergency operating procedures were to be revised to include caution notes and treatment of measurement errors by October 1, 1979; in addition, operator training was to be completed by the same date.
Indian Point 2	Per the Con Ed response of September 17, 1979, existing procedures and operator training were satisfactory for the steam generators, but procedural revisions and retraining for the pressurizer were to be completed by December 17, 1979.
Indian Point 3	Per the PASNY response of November 2, 1979, existing procedures and operator training were satisfactory for the steam generators, but any necessary procedural revisions and retraining for the pressurizer were to be completed by January 1, 1980.

See notes at end of Action 4 of Table B.5.

TABLE B.5 (contd)

ACTION 4 (contd) Facility(W)	Licensee Commitments for Procedures and Training
Kewaunee	Per the WPS response of October 18, 1979, "the effects of the containment environment and the system status on all instrumentation and instruction in the use of diverse instrumentation is (sic) included in the operator training program".
North Anna 1	Per the VEPCO response of September 14, 1979, precautions for steam generator operation had been incorporated in emergency procedures and were to be brought to the attention of operators, but further analysis of pressurizer effects was required.
Point Beach 1,2	Per the WEPCO response of September 17, 1979, emergency procedures and/or standing orders were to be revised by December 31, 1979, and operator training was to be completed one or two months later or after completion of NRC review.
Prairie Island 1,2	Per the NSP response of September 14, 1979, emergency procedures were to be revised and operators were to be trained after agreement was reached by NSP, W and NRC personnel.
Robinson 2	Per the CP&L response of September 14, 1979, revision of emergency procedures and training of operators were to be completed by December 1, 1979.
Salem 1	Per the PSE&G response of October 5, 1979, post-accident operating procedures were to be revised prior to return to service and were to include cautions for operators; in addition, operators were to be provided with correction curves and trained to use the revised procedures by November 5, 1979.
San Onofre 1	Per the SCE response of September 14, 1979, emergency procedures and station orders had been or were to be reviewed, and all required actions including operator training were to be completed by September 30, 1979.

See notes at end of Action 4 of Table B.5.

TABLE B.5 (contd)

ACTION 4 (contd) Facility(W)	Licensee Commitments for Procedures and Training
Surry 1,2	Per the VEPCO response of September 14, 1979, applicable emergency and abnormal procedures were being reviewed, and any required revisions were to be completed by November 15, 1979. The planned completion date for operator training was not given.
Trojan	Per the PGE response of September 14, 1979, new corrective curves had been added to the Control Room Operating Curves and Tables Reference Manual, appropriate control room indicators had been marked to caution the operators, and operators had been trained to use the revised methods.
Turkey Point 3,4	Per the FPL response of September 18, 1979, emergency procedures were being reviewed and would be revised if required by October 31, 1979; in addition operator training in using the revised procedures and correction factors was to be completed by November 30, 1979.
Yankee-Rowe 1	Per the YAECO response of October 9, 1979, reactor trip caused by large ruptures inside containment was based primarily on pressure indications. For certain small ruptures inside containment, the primary reactor protection was provided per level measurements which were essentialy correct. Similarly, backup protection was not affected significantly by errors in level measurements. Post-accident monitoring was not affected significantly by errors in level measurements. It was concluded by the utility that the bulletin concerns for reactor protection and post-accident monitoring did not apply to the Yankee-Rowe design.
Zion 1,2	Per the CECO responses of September 21 and December 14, 1979, some emergency procedures had been revised and brought to the attention of operators; additional revisions of procedures and training of operators were to be completed by January 1,1980.

Notes: 1. All responses for W facilities comply with the requirements of Action 4.

2. For the requirements of Action 4, see Page A-2.

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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 Regulations, Energy, Title 10, Chapter 1, January 1, 1986, cited as 10CFR 0.735-1.

APPENDIX C

Abbreviations

AEPSCO	American Electric Power Services Corporation
Allis	Allis-Chalmers Corporation
APCO	Alabama Power Company
AP&L	Arkansas Power and Light Company
APSCO	Arizona Public Service Company
BECO	Boston Edison Company
BG&E	Baltimore Gas and Electric Company
B&W	Babcock & Wilcox
BWR	Boiling Water Reactor
CD	Cancelled
C-E	Combustion Engineering, Inc.
CECO	Commonwealth Edison Company
CEI	Cleveland Electric Illuminating Company
CFR	Code of Federal Regulations
CG&E	Cincinnati Gas and Electric Company
CHI	Construction Halted Indefinitely
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
CYAPCO	Connecticut Yankee Atomic Power Company
DE	Division of Engineering (NRR)
DECO	Detroit Edison Company
DLC	Duquesne Light Company
D/P	Differential Pressure
DPC	Dairyland Power Cooperative
DUPCO	Duke Power Company
FP	Florida Power Corporation
FPL	Florida Power & Light Company
GAO	Government Accounting Office
GPC	Georgia Power Company
GSU	Gulf States Utilities Company
HL&P	Houston Lighting and Power Company
HQ	Headquarters
IE	(See NRC/IE)
IEB	Inspection and Enforcement Bulletin (NRC)
IELPCO	Iowa Electric Light and Power Company
IMECO	Indiana and Michigan Electric Company
IP	Illinois Power Company
IR	Inspection Report (NRC/IE)

JCP&L	Jersey Central Power and Light Company
KG&E	Kansas Gas and Electric Company
LER	Licensee Event Report
LILCO	Long Island Lighting Company
LP&L	Louisiana Power and Light Company
LPTL	Low Power Testing License
Met-Ed	Metropolitan Edison Company
MP&L	Mississippi Power and Light Company
MYAPCO	Maine Yankee Atomic Power Company
NIPSCO	Northern Indiana Public Service Company
NMP	Niagara Mohawk Power Company
NNECO	Northeast Nuclear Energy Company
NPPD	Nebraska Public Power District
NRC/IE	Nuclear Regulatory Commission/ Office of Inspection & Enforcement
NRR	Office of Nuclear Reactor Regulation (NRC)
NSP	Northern States Power Company
NSSS	Nuclear Steam Supply System
NU	Northeast Utilities
NWL	Normal Water Level
OL	Operating License
OPPD	Omaha Public Power District
PAM	Post-Accident Monitoring
PASNY	Power Authority of the State of New York
PECO	Philadelphia Electric Company
PGE	Portland General Electric Company
PG&E	Pacific Gas and Electric Company
PP&L	Pennsylvania Power and Light Company
PSCC	Public Service Company of Colorado
PSCO	Public Service Company of Oklahoma
PSE&G	Public Service Electric and Gas Company
PSI	Public Service Indiana
psia	Pounds per square inch absolute
PSNH	Public Service Company of New Hampshire
PWR	Pressurized Water Reactor
PZR	Pressurizer
R	Region (NRC)
RCS	Reactor Cooling System
RG&E	Rochester Gas and Electric Corporation
RPS	Reactor Protection System
SCE	Southern California Edison Company
SCE&G	South Carolina Electric and Gas Company
SDI	Shut Down Indefinitely
S/G	Steam Generator
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
UE	Union Electric Company

VEPCO	Virginia Electric and Power Company
VYNP	Vermont Yankee Nuclear Power Corporation
W	Westinghouse Electric Corporation
WEPCO	Wisconsin Electric Power Company
WNP	Washington Nuclear Project
WNSD	Westinghouse Nuclear Service Division
WPPSS	Washington Public Power Supply System
WPS	Wisconsin Public Service Corporation
YAECO	Yankee Atomic Electric Company

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On June 22, 1979, Westinghouse Electric Corporation reported to the NRC that elevated containment temperature could affect the reference leg water column and the indicated steam generator water level. IE Bulletin 79-21 was issued by the NRC on August 13, 1979 because of concern that the temperature effect could cause indication of erroneously high steam generator water levels, could delay or prevent protection signals and could cause incorrect information during post-accident monitoring. Because safety-related water level measuring systems used by Babcock & Wilcox and Combustion Engineering could be affected in the same way, the bulletin was issued for action to all utilities with operating pressurized water reactors (PWRs). The bulletin was issued for information to utilities with either PWRs under construction or operating boiling water reactors (BWRs). A related generic letter concerning BWRs was issued by the NRC in July 1979 for information only. Evaluation of licensees' responses and NRC/IE inspection reports shows that the bulletin can be closed out per specific criteria for all of the 41 facilities to which it was issued for action. It is concluded that utility responses were consistent because of guidance from the NSSS suppliers. Remaining areas of concern involve a possible need for manually deenergizing pressurizer heaters in B&W facilities, and further evaluation of boiling in the reference leg by Westinghouse.

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