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UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
101 MARIETTA ST., N.W., SUITE 3100
ATLANTA, GEORGIA 30303

AUG 13 1979

In Reply Refer To:
RII:JPO
50-413, 50-414
50-488, 50-489
50-490, 50-491
50-492, 50-493

Duke Power Company
Attn: L. C. Dail, Vice President
Design Engineering
P. O. Box 33189
Charlotte, North Carolina 28242

Gentlemen:

The enclosed Bulletin 79-21 is forwarded to you for information. No written response is required. If you desire additional information regarding this matter, please contact this office.

Sincerely,

for J... P. O'Reilly
Director

Enclosures:
1. IE Bulletin No. 79-21 w/encl.
2. List of IE Bulletins Issued in the Last 6 Months

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Duke Power Company

cc w/encl:
D. G. Beam, Project Manager
Post Office Box 223
Clover, South Carolina 29710

J. T. Moore, Project Manager
Post Office Box 422
Gaffney, South Carolina 29340

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Accession No: 7908090193
SSINS No: 6820

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

August 13, 1979

IE Bulletin No. 79-21

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, Enclosure 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. Enclosure 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-acciden 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 Enclosure 1.

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.

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

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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.

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

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.

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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 conceiva[bly] 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 le[vel] 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 effects, 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

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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 ≤ 1000 psia

Reference Leg Calibration Temperature ≥ 90°F

Height of Reference Leg ≤ 1.1x Level Span

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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 Minimum 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 ≤ 1000 psia

Reference Leg Calibration Temperature ≥ 90°F

Height of Reference Leg ≤ 1.1 x Level Span

Pressure ≥ 50 psia

Pressure ≤ 200 psi + Calibration Pressure

Boiling in the Reference Leg is not assumed.

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LISTING OF IE BULLETINS
ISSUED IN LAST SIX MONTHS

Bulletin No.	Subject	Date Issued	Issued To
79-20	Packaging Low-Level Radioactive Waste for Transport and Burial	8/10/79	Materials Licensees who did not receive Bulletin No. 79-19
79-19	Packaging Low-Level Radioactive Waste for Transport and Burial	8/10/79	All Power and Research Reactors with OLs, fuel facilities except uranium mills, and certain materials licensees
79-18	Audibility Problems Encountered on Evacuation	8/7/79	All Power Reactor Facilities with an Operating License
79-17	Pipe Cracks in Stagnant Borated Water Systems at PWR Plants	7/26/79	All PWR's with operating license
79-16	Vital Area Access Controls	7/26/79	All Holders of and applicants for Power Reactor Operating Licenses who anticipate loading fue prior to 1981
79-15	Deep Draft Pump Deficiencies	7/11/79	All Power Reactor Licensees with a CP and/or OL
79-14	Seismic Analyses for As-Built Safety-Related Piping System	6/2/79	All Power Reactor facilities with an OL or a CP
79-13	Cracking In Feedwater System Piping	6/25/79	All PWRs with an OL for action. All BWRs with a CP for information.
79-02 (Rev. 1)	Pipe Support Base Plate Designs Using Concrete Expansion Anchor Bolts	6/21/79	All Power Reactor Facilities with an OL or a CP
79-12	Short Period Scrams at BWR Facilities	5/31/79	All GE BWR Facilities with an OL

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LISTING OF IE BULLETINS
ISSUED IN LAST SIX MONTHS

Bulletin No.	Subject	Date Issued	Issued To
79-11	Faulty Overcurrent Trip Device in Circuit Breakers for Engineered Safety Systems	5/22/79	All Power Reactor Facilities with an OL or a CP
79-10	Requalification Training Program Statistics	5/11/79	All Power Reactor Facilities with an OL
79-09	Failures of GE Type AK-2 Circuit Breaker in Safety Related Systems	4/17/79	All Power Reactor Facilities with an OL or CP
79-08	Events Relevant to BWR Reactors Identified During Three Mile Island Incident	4/14/79	All BWR Power Reactor Facilities with an OL
79-07	Seismic Stress Analysis of Safety-Related Piping	4/14/79	All Power Reactor Facilities with an OL or CP
79-05C&06C	Nuclear Incident at Three Mile Island - Supplement	7/26/79	To all PWR Power Reactor Facilities with an OL
79-06B	Review of Operational Errors and System Misalignments Identified During the Three Mile Island Incident	4/14/79	All Combustion Engineering Designed Pressurized Water Power Reactor Facilities with an Operating License
79-06A (Rev 1)	Review of Operational Errors and System Misalignments Identified During the Three Mile Island Incident	4/18/79	All Pressurized Water Power Reactor Facilities of Westinghouse Design with an OL
79-06A	Review of Operational Errors and System Misalignments Identified During the Three Mile Island Incident	4/14/79	All Pressurized Water Power Reactor Facilities of Westinghouse Design with an OL

LISTING OF IE BULLETINS
ISSUED IN LAST SIX MONTHS

Bulletin No.	Subject	Date Issued	Issued To
79-06	Review of Operational Errors and System Mis-alignments Identified During the Three Mile Island Incident	4/11/79	All Pressurized Water Power Reactors with an OL except B&W facilities
79-05B	Nuclear Incident at Three Mile Island	5/21/79	All B&W Power Reactor Facilities with an OL
79-05A	Nuclear Incident at Three Mile Island	4/5/79	All B&W Power Reactor Facilities with an OL
79-05	Nuclear Incident at Three Mile Island	4/2/79	All Power Reactor Facilities with an OL and CP
79-04	Incorrect Weights for Swing Check Valves Manufactured by Velan Engineering Corporation	3/30/79	All Power Reactor Facilities with an OL or CP
78-12B	Atypical Weld Material in Reactor Pressure Vessel Welds	3/19/79	All Power Reactor Facilities with an OL or CP
79-03	Longitudinal Welds Defects In ASME SA-312 Type 304 Stainless Steel Pipe Spools Manufactured by Youngstown Welding and Engineering Co.	3/12/79	All Power Reactor Facilities with an OL or CP
79-01A	Environmental Qualification of Class 1E Equipment (Deficiencies in the Environmental Qualification of ASCO Solenoid Valves)	6/6/79	All Power Reactor Facilities with an OL or CP

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