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

August 20, 1992

Docket No: 52-001

MEMORANDUM FOR: Robert C. Pierson, Project Director
Standardization Project Directorate
Division of Advanced Reactors and
Special Projects

FROM: Robert C. Jones
Reactor Systems Branch
Division of Systems Technology

SUBJECT: ABWR-LEVEL INSTRUMENTATION INACCURACIES CAUSED BY RAPID DEPRESSURIZATION

Generic Letter No. 92-04 " Resolution of the issues related to reactor vessel water level instrumentation in BWRs pursuant to 10 CFR 50.54 (F)" was issued on August 19, 1992 to request information regarding the adequacy of and corrective actions for BWR water level instrumentation with respect to the effects of non-condensible gases on system operation. We are enclosing copies of the Generic Letter and the Information Notice for transmitting to GE for their immediate action in the ABWR design. We require a quick response from GE for any potential design modifications or changes in the Emergency Procedure Guidelines (EPG) of the ABWR.

Robert C. Jones, Chief
Reactor Systems Branch
Division of Systems Technology

Enclosure:
as stated

cc (w/o enclosure):
C. Poslusny
V. McCree
S Newberry
M. Chiramal
J. Stewart
A. Thadani

Contact: G. Thomas, SRXB/DST, 504-1814

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PDR

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Ltr. Encl.
ADD: OGC/HDSS 1 1
ACRS 10 10
D03 1/1

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

August 19, 1992

TO: ALL BOILING WATER REACTOR (BWR) LICENSEES OF OPERATING REACTORS

SUBJECT: RESOLUTION OF THE ISSUES RELATED TO REACTOR VESSEL WATER LEVEL INSTRUMENTATION IN BWRs PURSUANT TO 10 CFR 50.54(F) (GENERIC LETTER NO. 92-04)

Purpose

The U.S. Nuclear Regulatory Commission (NRC) is issuing this generic letter to request information regarding the adequacy of and corrective actions for Boiling Water Reactor (BWR) water level instrumentation with respect to the effects of noncondensible gases on system operation.

Background and Safety Considerations

As discussed in NRC Information Notice No. 92-54 "Level Instrumentation Inaccuracies Caused by Rapid Depressurization," the staff is concerned that noncondensible gases may become dissolved in the reference leg of BWR water level instrumentation and can lead to a false high level indication after a rapid depressurization event. The dissolved gases which accumulate over time during normal operation can rapidly come out of solution during depressurization and displace water from the reference leg. A reduced reference leg level will result in a false high level indication. This is important to safety because water level signals are used for actuating automatic safety systems and for guidance to operators during and after an event.

On July 29, 1992, the NRC staff held a public meeting with the Regulatory Response Group (RRG) of the Boiling Water Reactor Owners Group (BWROG) to discuss the effect of inaccuracies in the reactor vessel level instrumentation system in BWRs. During the meeting, the BWROG and its consultant, General Electric Company (GE), presented the results of analyses assessing the safety implications of the postulated error in level indication. The analyses consisted of two basic parts: (1) an assessment of the mechanism and potential magnitude of errors in the level instruments and (2) a review of the relevant licensing basis transients and accidents to determine the effect of this error on plant response, including post-accident operator actions.

The BWROG analyses indicated that significant errors in level indication can occur as a result of degassing the instrument reference leg if noncondensible gas is dissolved in the reference leg and if the reactor abruptly depressurizes below 450 psig.

The NRC staff reviewed the BWROG analyses and selected design basis accident scenarios which lead to a lowering of the reactor vessel water level and has concluded that automatic safety systems will be actuated at pressures well above 450 psig, even for postulated worst-case noncondensible gas concentrations in the reference legs. Therefore, the NRC is confident that all emergency cooling systems will initiate as they were designed to do. In addition, the BWROG discussed diverse signals which would also initiate ECCS for reactor water level lowering events. The NRC staff reviewed the backup systems and concluded that the ECCS would be initiated by diverse signals as analyzed by the BWROG.

After ECCS actuation, reactor water level indication is used by the operators for long term actions (i.e., maintaining adequate reactor water level and ensuring adequate core cooling). Operators would not utilize only reactor vessel level indications to determine accident mitigation actions but would also utilize other indications such as containment pressure, temperature, and humidity to determine accident mitigation strategies. Additionally, events characterized by gradual depressurization would lead to a reduced error in the indicated level. There are two or four reference leg columns in each plant, depending on plant design. The amount of noncondensible gases dissolved in each depends primarily upon system leakage and geometry. Because of this, a common mode, common magnitude level indication error is unlikely. Operators would therefore see a mismatch in indicated level alerting them to a level indication problem. Finally, emergency procedure guidelines (EPGs) state that when reactor vessel water level is indeterminate, operators should flood the reactor vessel using at least one pump guided by the unaffected diverse instrumentation (i.e., high containment pressure indication). Reactor operators are trained to deal with these situations should they occur.

Upon reviewing the information provided by the BWROG and the staff's assessment, the staff concluded that interim plant operation is acceptable. The bases for the staff's conclusion are as follows: 1) the level instrumentation is expected to initiate safety systems prior to a significant depressurization of the reactor; 2) emergency procedures which are currently in place in conjunction with operator training are expected to result in adequate operator actions; and 3) an abrupt depressurization event resulting in a common mode, common magnitude level indication error is unlikely.

For longer term operation however, the staff considers potential water level instrumentation inaccuracies an important issue because level indication has safety and control functions in all

modes of BWR operation. Furthermore, since the analyses provided are of a generic nature and the magnitude of possible errors depends strongly upon plant-specific factors such as system leakage and geometry, it is important that the analyses be reviewed promptly by all individual licensees.

Basis for Compliance Determination

The level errors that could result from the effects of noncondensible gas may prevent the level instrumentation systems in BWRs from satisfying the following regulations:

(1) General Design Criterion (GDC) 13, "Instrumentation and control," which requires that "Instrumentation shall be provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions as appropriate to assure adequate safety." Existing instrumentation may not accurately monitor reactor vessel water level under accident conditions.

(2) GDC 21, "Protection system reliability and testability," which requires that "The protection system shall be designed for high functional reliability...commensurate with the safety function to be performed." The instrumentation may not be reliable under rapid depressurization conditions.

(3) GDC 22, "Protection system independence," which requires that "The protection system shall be designed to assure that the effects of natural phenomena, and of normal operating, maintenance, testing, and postulated accident conditions...do not result in loss of the protection function." The natural phenomena of degassing may cause a loss of the reactor vessel water level indication function following a rapid depressurization.

(4) Section 50.55a(h) of Title 10 of the Code of Federal Regulations (10 CFR 50.55a(h)), which requires that protection systems, for those plants with construction permits issued after January 1, 1971, shall meet the requirements stated in editions of the Institute of Electrical and Electronics Engineers Standard "Criteria for Protection Systems for Nuclear Power Generating Stations" (IEEE-279). Section 4.20 of IEEE-279 requires that "The protection system shall be designed to provide the operator with accurate,

complete, and timely information pertinent to its own status and to generating station safety." The water level instrumentation for the reactor vessel may not be accurate after a rapid depressurization event.

Requested Actions

1. In light of potential errors resulting from the effects of noncondensible gas, each licensee should determine:

   a. The impact of potential level indication errors on automatic safety system response during all licensing basis transients and accidents;

   b. The impact of potential level indication errors on operator's short and long term actions during and after all licensing basis accidents and transients;

   c. The impact of potential level indication errors on operator actions prescribed in emergency operating procedures or other affected procedures not covered in (b).

2. Based upon the results of (1), above, each licensee should notify the NRC of short term actions taken, such as:

   a. Periodic monitoring of level instrumentation system leakage; and,

   b. Implementation of procedures and operator training to assure that potential level errors will not result in improper operator actions.

3. Each licensee should provide its plans and schedule for corrective actions, including any proposed hardware modifications necessary to ensure the level instrumentation system design is of high functional reliability for long term operation. Since this instrumentation plays an important role in plant safety and is required for both normal and accident conditions, the staff recommends that each utility implement its longer term actions to assure a level instrumentation system of high functional reliability at the first opportunity but prior to starting up after the next refueling outage commencing 3 months after the date of this letter.

Required Information

Because of the importance of plant-specific aspects of this issue and the potential magnitude of the errors, the staff requires, pursuant to 10 CFR 50.54(f) and Section 182 of the Atomic Energy Act, that you provide a response to this letter by September 27, 1992.

Merely committing to evaluate the safety significance as part of the individual plant examination (IPE) program is not an acceptable alternative to the actions described herein, since the licensee should resolve this issue as a matter of compliance.

Backfit Discussion

In accordance with NRC procedures, the actions requested herein are considered a backfit to assure that facilities are in compliance with existing regulatory requirements discussed above. Thus, a backfit analysis is not required by 10 CFR 50.109(a)(4)(i), and the staff performed a documented evaluation as discussed in 10 CFR 50.109(a)(6). The documented evaluation is provided in the preceding discussions.

Burden Information

This request is covered by Office of Management and Budget Clearance Number 3150-0011, which expires May 31, 1994. The estimated average number of burden hours is 200 person hours for each licensee response, including the time required to assess the questions, search data sources, gather and analyze the data, and prepare the required response. These estimated average burden hours pertain only to the identified response-related matters and do not include the time for actual implementation of the requested actions. Comments on the accuracy of this estimate and suggestions to reduce the burden may be directed to Ronald Minsk, Office of Information and Regulatory Affairs (3150-0011), NEOB-3019, Office of Management and Budget, Washington, D.C. 20503 and to the U.S. Nuclear Regulatory Commission, Information and Records Management Branch, Division of Information Support Services, Office of Information and Resources Management, Washington, D.C. 20555.

Although no specific request or requirement is intended, the following information would be helpful to the NRC in evaluating the cost of complying with this generic letter:

(1) the licensee staff time and costs to perform requested inspections, corrective actions, and associated testing;

(2) the licensee staff's time and costs to prepare the requested reports and documentation;

(3) the additional short-term costs incurred as a result of the inspection findings such as the costs of the corrective actions or the costs of down time; and

(4) an estimate of the additional long-term costs which will be incurred in the future as a result of implementing commitments such as the estimated costs of conducting future inspections or increased maintenance.

Please address your response to this generic letter to the U.S. Nuclear Regulatory Commission, Attn: Document Control Desk, Washington, D.C. 20555 pursuant to 10 CFR 50.4(a) of the NRC's regulations.

Sincerely,

James G. Partlow
Associate Director for Projects
Office of Nuclear Reactor Regulation

Enclosure:
List of recently issued generic letters.

Technical Contact: Timothy E. Collins, NRR
(301) 504-2897

ENCLOSURE

LIST OF RECENTLY ISSUED GENERIC LETTERS

Generic Letter No.	Subject	Date of Issuance	Issued To
90-02 SUPPLEMENT 1	ALTERNATIVE REQUIREMENTS FOR FUEL ASSEMBLIES IN THE DESIGN FEATURES SECTION OF TECHNICAL SPECIFICATIONS	07/31/92	ALL LWR LICENSEES AND APPLICANTS
87-02 SUPPLEMENT 1	SAFETY EVALUATION REPORT NO. 2 ON SQUG GENERIC IMPLEMENTATION PROCEDURE, REVISION 2.	05/22/92	ALL USI A-46 LICENSEES WHO ARE SQUG MEMBERS
92-03	COMPILATION OF THE CURRENT LICENSING BASIS: REQUEST FOR VOLUNTARY PARTICIPATION IN PILOT PROGRAM	03/19/92	ALL NUCLEAR POWER PLANT APPLICANTS AND LICENSEES
92-01 REVISION 1	REACTOR VESSEL STRUCTURAL INTEGRITY, 10CFR50.54(f)	3/06/92	ALL HOLDERS OF OP LICENSES OR CONST. PERMITS FOR NUCLEAR PWR PLANTS (EXCEPT YANKEE ATOMIC FOR YANKEE NUC PWR STA.)
92-02	RESOLUTION OF GENERIC ISSUE 79, UNANALYZED REACTOR VESSEL (PWR) THERMAL STRESS DURING NATURAL CONVECTION COOLDOWN	03/06/92	ALL HOLDERS OF OP LICENSES OF CONST. PERMITS FOR PWRs
92-01	REACTOR VESSEL STRUCTURAL INTEGRITY, 10CFR50.54(f)	NOT ISSUED Revision Listed Above	ALL HOLDERS OF OP LICENSES OR CONST. PERMITS FOR NUCLEAR PWR PLANTS (EXCEPT YANKEE ATOMIC FOR YANKEE NUC PWR STA.)
* 89-10 SUPPLEMENT 4	CONSIDERATION OF VALVE MISPOSITIONING IN BWRs	02/14/92	ALL LICENSEES OF OP NUC PWR PLANTS AND HOLDERS OF CONSTRUC. PERMITS FOR PWR PLANTS

* NOTE: 89-10 Supp. 4 -
Accession No. 9202070037 has been changed to 9202250311.

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON, D.C. 20555

July 24, 1992

NRC INFORMATION NOTICE NO. 92-54: LEVEL INSTRUMENTATION INACCURACIES CAUSED BY RAPID DEPRESSURIZATION

Addressees

All holders of operating licenses or construction permits for nuclear power reactors.

Purpose

The U.S. Nuclear Regulatory Commission (NRC) is issuing this information notice to alert addressees to potential inaccuracies in water level indication during and after rapid depressurization events. This problem may affect the indication of pressurizer level for pressurized water reactors (PWR) and reactor vessel level for boiling water reactors (BWR). It is expected that recipients will review the information for applicability to their facilities and consider actions, as appropriate, to avoid similar problems. However, suggestions contained in this information notice are not NRC requirements; therefore, no specific action or written response is required.

Description of Circumstances

On April 17, 1991, Northeast Utilities (NU) filed a licensee event report (LER) for Millstone Unit 3, documenting pressurizer level instrument inaccuracies. According to the LER the inaccuracies result from non-condensible gases collecting in the condensing pots of the instrument reference legs. The LER stated that pressurizer level would be used to make decisions concerning operator actions directed by the Emergency Operating Procedures (EOP).

During the previous operating cycle, NU monitored the accuracy of the pressurizer level instrumentation and observed a worst case error of 3.6% of full scale and also confirmed that non-condensible gases had accumulated inside the condensing pots. The root causes for the accumulation of non-condensible gases in the condensing pots were 1) the instrument lines sloped upward from the pressurizer to the condensing pots and, 2) a restricting orifice in each instrument line prevented the free flow of steam and non-condensible gases between the pressurizer and the condensing pots. NU corrected the problem by removing the condensing pot and changing the instrument line slope.

Westinghouse and Combustion Engineering performed further engineering evaluations and concluded that during a rapid depressurization of the Reactor

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Coolant System (RCS), during certain design basis accidents, the release of non-condensible gases could result in a level indication error of about +40 percent of full scale. The staff has evaluated the effects of this error and determined that the only unacceptable actions that could be taken by operators as a result of this error are to prematurely terminate safety injection (SI) or to fail to re-initiate SI if required. The staff further determined that the pressurizer level instrumentation is not used as the primary parameter evaluated by operators for safety injection termination and that PWR emergency operating procedures direct operators to consult other instrumentation and parameters (reactor vessel level monitoring system, RCS subcooling and a stable or increasing RCS pressure) prior to terminating SI.

For BWRs, reactor vessel level indication system (RVLIS) errors had also been identified in the past and the staff issued Generic Letter (GL) No. 84-23, "Reactor Vessel Water Level Instrumentation in BWRs" to address the concern. This GL was based on the BWR Owners Group (BWROG) report (SLI-8218 issued in November 1982), "Inadequate Core Cooling Detection in Boiling Water Reactors." However, these documents do not specifically address the non-condensible gas evolution concern associated with rapid depressurization. The staff has requested the BWROG to address this issue and GE is preparing a report on behalf of the BWROG.

On July 15, 1992, Northeast Utilities (NU) made a notification to the NRC under Section 50.72 of Title 10 of the Code of Federal Regulations (10 CFR 50.72) regarding inaccuracies in reactor vessel level indication at Millstone Unit 1. This notification indicated that the level instrumentation may not provide accurate indication following a rapid depressurization event as a consequence of the expulsion of water from the reference leg due to the release of non-condensible gases in the instrument reference leg. In a conference call with NU on July 21, 1992, the licensee stated that the Yarway level measurement instrumentation which provides the automatic actuation of safety systems at Millstone Unit 1 would not be affected by this phenomenon. However, the GE/MAC level instrumentation, which is used for indication, feedwater control, and containment spray pump interlocks, would be affected. Following a rapid depressurization event, the operator might receive inaccurate information from the GE/MAC instrumentation leading the operator to perform inappropriate manual actions. The licensee has estimated, based on a conservative analysis, that the upper bound of the error in the GE/MAC instrumentation could be as much as 15 to 20 feet. Millstone Unit 1 is currently in cold shutdown for service water repairs, and NU is reviewing the error analysis and a possible modification to the condensing pot arrangement in order to reduce inaccuracies in the level indication to an acceptable level before restart.

In a conference call on July 22, 1992, the staff informed the BWROG of the results of the Northeast Utilities' analyses and the licensee's planned actions. The BWROG indicated its position that the error would not exceed 4 inches if the reference leg configuration is installed in accordance with vendor recommendations.

The NRC has activated the BWR Regulatory Response Group and scheduled a meeting to discuss this issue at NRC headquarters in Rockville, Md., on July 29, 1992.

Discussion

Inaccuracies in level instrumentation in PWRs and BWRs could affect the performance of safety functions. GL 84-23, BWROG report SLI-8218, and vendor recommendations are intended to provide guidance to preclude the operators from taking improper actions during normal plant operation. The inaccuracies caused by rapid depressurization events in PWRs have limited safety significance because instrumentation other than that for pressurizer level is used by the operators to determine appropriate manual actions. For BWRs, however, large errors in level indication may have greater safety significance. An evaluation by the staff is continuing and when the evaluation is completed the staff will determine if additional regulatory actions will be necessary.

This information notice requires no specific action or written response. If you have any questions regarding the information in this notice, please contact one of the technical contacts listed below or the appropriate Office of Nuclear Reactor Regulation (NRR) project manager.

Charles E. Rossi

Charles E. Rossi, Director
Division of Operational Events Assessment
Office of Nuclear Reactor Regulation

Technical contacts: Hukam C. Garg, NRR
(301) 504-2929

Tim Collins, NRR
(301) 504-2897

Attachment:
List of Recently Issued NRC Information Notices

LIST OF RECENTLY ISSUED
NRC INFORMATION NOTICES

Information Notice No.	Subject	Date of Issuance	Issued to
92-53	Potential Failure of Emergency Diesel Generators due to Excessive Rate of Loading	07/29/92	All holders of OLs or CPs for nuclear power reactors.
91-52, Supp. 1	Nonconservative Errors in Overtemperature Delta-Temperature (OTΔT) Setpoint Caused by Improper Gain Settings	07/16/92	All holders of OLs or CPs for Westinghouse (W)-designed nuclear power reactors.
92-52	Barriers and Seals Between Mild and Harsh Environments	07/15/92	All holders of OLs or CPs for nuclear power reactors.
92-51	Misapplication and Inadequate Testing of Molded-Case Circuit Breakers	07/09/92	All holders of OLs or CPs for nuclear power reactors.
92-50	Cracking of Valves in the Condensate Return Lines of A BWR Emergency Condenser System	07/02/92	All holders of OLs or CPs for BWRs.
92-49	Recent Loss or Severe Degradation of Service Water Systems	07/02/92	All holders of OLs or CPs for nuclear power reactors.
92-48	Failure of Exide Batteries	07/02/92	All holders of OLs or CPs for nuclear power reactors.
92-47	Intentional bypassing of Automatic Actuation of Plant Protective Features	06/29/92	All holders of OLs or CPs for nuclear power reactors.

OL = Operating License
CP = Construction Permit

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Charles E. Rossi, Director
Division of Operational Events Assessment
Office of Nuclear Reactor Regulation

Technical contacts: Hukam C. Garg, NRR
(301) 504-2929

Tim Collins, NRR
(301) 504-2897

Attachment:
List of Recently Issued NRC Information Notices

Note: TechEd was not available.

*See previous concurrences.

SICB*	SICB*	SICB:DST*	SRXB*	SRXB:DST*
H. Garg:lm	E. Marinos	S. Newberry	T. Collins	R. Jones
7/24/92	7/24/92	7/24/92	7/24/92	7/24/92
DD:DST*	D:DST*	OGCB:DOEA*	D:DOEA	
G. Holahan	A. Thadani	C. Berlinger	C. Rossi	
7/24/92	7/24/92	7/24/92	7/24/92	

Name of Document: 92-54.IN