Central File

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

APR 6 1994

Mr. Thomas J. Saporito, Jr.
Post Office Box 7603
Jupiter, FL 33468

Dear Mr. Saporito:

I am responding to your letter of February 22, 1994, to Chairman Selin regarding water level instrumentation. Although you are entitled to your opinion of the Chairman's performance on the television program aired in late February, the subject of this letter is the NRC's handling of the technical issue that you raised in your letter, particularly the issue of BWR water level instrumentation. As you stated, Paul Blanch raised these concerns and he deserves credit for bringing this matter to our attention. Reactor vessel water level instrumentation is critical to the safety and operation of boiling water reactors (BWRs) and the NRC has taken actions to address this issue.

On July 22, 1992, the staff requested activation of the BWR Owners Group (BWROG) Regulatory Response Group, and held a public meeting with the group on July 29, 1992, to discuss this issue. The staff issued Information Notice 92-54 (Enclosure 1) on July 24, 1992, to alert licensees to the potential for level instrumentation inaccuracies caused by rapid depressurization. On August 19, 1992, the staff issued Generic Letter (GL) 92-04 (Enclosure 2), in which it requested that each BWR licensee (1) evaluate the impact of potential level indication errors on its facility, (2) notify the staff of any corrective actions taken, and (3) provide its plans and schedule for corrective actions. In response to the GL, all BWR licensees implemented short-term compensatory measures including operator training and procedures.

The BWROG proposed a program to resolve the concerns discussed in GL 92-04, which included a full scale testing program. The NRC staff determined that continued plant operation was acceptable until hardware modifications could be implemented, and agreed to allow the BWROG to proceed with their test program. The basis for continued operation as discussed in the GL was as follows: (1) the level instrumentation is expected to initiate safety systems before a significant depressurization of the reactor and, therefore, before significant errors occur; (2) emergency procedures 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.

New information was obtained during early 1993 that led to the issuance of NRC Information Notice 93-27 (Enclosure 3) on April 8, 1993, and NRC Bulletin (NRCB) 93-03 (Enclosure 4) on May 26, 1993. This new information included (1) report of an event at the WNP-2 plant involving significant level errors during normal plant depressurization (a 32 inch level error occurred and recovered over a two hour period); (2) data from a test program conducted by the BWROG that confirmed the potential for significant errors in the level instrumentation; and (3) a report submitted by the BWROG on May 20, 1993,

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(Enclosure 5), that discussed the possible effect of level errors on safety system response during accident scenarios initiated from shutdown conditions, such as draindown events. In NRCB 93-03, the staff requested that each affected BWR licensee take additional short-term compensatory actions to address level errors during normal depressurization conditions and that they implement hardware modifications during the next cold shutdown after July 31, 1993.

Each affected BWR licensee has completed these short-term compensatory actions and has committed to implement hardware modifications to the level instrumentation. To date, 21 of the 36 affected BWR units have completed implementation of hardware modifications. Eight units are currently shutdown and will complete modifications prior to restart. By June, 1994, only 4 units will be operating without hardware modifications installed, and all are scheduled to shutdown by 10/94 and will implement modifications prior to restart.

The NRC considers this issue to be very important and is continuing to take actions, including requests for timely implementation of hardware modifications, to ensure that it is addressed adequately. Thank you for informing us of your concerns about this issue.

Sincerely, Original Signed By
WILLIAM T. RUSSELL
William T. Russell, Director
Office of Nuclear Reactor Regulation

Enclosures:
As stated
cc: Paul M. Blanch

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ENCLOSURE 1

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

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

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 conclusions 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.

<u>Basis for Compliance Determination</u>

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.

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

April 8, 1993

NRC INFORMATION NOTICE 93-27: LEVEL INSTRUMENTATION INACCURACIES OBSERVED DURING NORMAL PLANT 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 inaccuracies in reactor vessel level indication that occurred during a normal depressurization of the reactor coolant system at the Washington Nuclear Plant Unit 2 (WNP-2) and to the fact that errors in level indication may result in a failure to automatically isolate the residual heat removal (RHR) system under certain conditions. 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.

Background

As discussed in NRC Information Notice 92-54, "Level Instrumentation Inaccuracies Caused by Rapid Depressurization," and Generic Letter 92-04, "Resolution of the Issues Related to Reactor Vessel Water Level Instrumentation in BWRs Pursuant to 10 CFR 50.54(f)," noncondensible gas may become dissolved in the reference leg of water level instrumentation and lead to false indications of high level after a rapid depressurization event. Reactor vessel level indication signals are important because these signals are used for actuating automatic safety systems and for guidance to operators during and after an event. While Information Notice 92-54 dealt with potential consequences of rapid system depressurization, this information notice discusses level indication errors that may occur during normal plant cooldown and depressurization.

Description of Circumstances

On January 21, 1993, during a plant cooldown following a reactor scram at WNP-2, "notching" of the level indication was observed on at least two of four channels of the reactor vessel narrow range level instrumentation. "Notching" is a momentary increase in indicated water level. This increase occurs when a gas bubble moves through a vertical portion of the reference leg and causes a temporary decrease in the static head in the reference leg. The notching at

WNP-2 was first observed on channel "C" at a pressure of approximately 827 kPa [120 psig]. Channel "B" experienced notching starting at approximately 350 kPa [50 psig]. At these pressures, the level error was on the order of 10 to 18 centimeters [4 to 7 inches] and persisted for approximately one minute.

Beginning at a pressure of approximately 240 kPa [35 psig], the level indication from channel "C" became erratic and, as the plant continued to depressurize, an 81-centimeter [32-inch] level indication error occurred. This depressurization was coincident with the initiation of the shutdown cooling system. The 81-centimeter [32-inch] level error was sustained and was gradually recovered over a period of two hours. The licensee postulated that this large error in level indication was caused by gas released in the reference leg displacing approximately 40 percent of the water volume. The licensee also postulated that the slow recovery of correct level indication was a result of the time needed for steam to condense in the condensate chamber and refill the reference leg. The licensee inspected the "C" reference leg and discovered leakage through reference leg fittings. This leakage may have been a contributing factor for an increased accumulation of dissolved noncondensible gas in that reference leg.

The licensee determined that the type of errors observed in level indication during this event could result in a failure to automatically isolate a leak in the RHR system during shutdown cooling. The design basis for WNP-2 includes a postulated leak in the RHR system piping outside containment while the plant is in the shutdown cooling mode. For this event, the shutdown cooling suction valves are assumed to automatically isolate on a low reactor vessel water level signal to mitigate the consequences of the event. For the January 21, 1993 plant cooldown, the licensee concluded that, with the observed errors in level indication, the shutdown cooling suction valves may not have automatically isolated the RHR system on low reactor vessel water level as designed. The licensee has implemented compensatory measures for future plant cooldowns to ensure that a leak that occurs in the RHR system during shutdown cooling operation would be isolated promptly. These measures include touring the associated RHR pump room hourly during shutdown cooling and backfilling the water level instrument reference legs after entry into mode 3 (hot shutdown). The licensee is also evaluating measures to minimize leakage from the "C" reference leg.

Discussion

The event described above is different than events previously reported because of the large magnitude and sustained duration (as opposed to momentary notching) of the level error that occurred during normal plant cooldown. A large sustained level error is of concern because of the potential for complicating long-term operator actions. In addition, the scenario of a postulated leak in the RHR system evaluated by WNP-2 suggests that some safety systems may not automatically actuate should an event occur while the reactor is in a reduced pressure condition. Generic Letter 92-04 requested, in part, that licensees determine the impact of potential level indication errors on automatic safety system response during licensing basis transients and accidents. The information in this notice indicates that sustained level instrument inaccuracies can occur during a normal reactor depressurization. Therefore, events occurring during low pressure conditions may also be complicated by level indication errors.

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

Brian Grimes

Brian K. Grimes, Director
Division of Operational Events Assessment
Office of Nuclear Reactor Regulation

Technical contact: Amy Cubbage, NRR
(301) 504-2875

Attachment:
List of Recently Issued NRC Information Notices

LIST OF RECENTLY ISSUED
NRC INFORMATION NOTICES

Information Notice No.	Subject	Date of Issuance	Issued to
93-26	Grease Solidification Causes Molded Case Circuit Breaker Failure to Close	04/07/93	All holders of OLs or CPs for nuclear power reactors.
93-25	Electrical Penetration Assembly Degradation	04/01/93	All holders of OLs or CPs for nuclear power reactors.
93-24	Distribution of Revision 7 of NUREG-1021, "Operator Licensing Examiner Standards"	03/31/93	All holders of operator and senior operator licenses at nuclear power reactors.
93-23	Weschler Instruments Model 252 Switchboard Meters	03/31/93	All holders of OLs or CPs for nuclear power reactors.
93-22	Tripping of Klockner-Moeller Molded-Case Circuit Breakers due to Support Level Failure	03/26/93	All holders of OLs or CPs for nuclear power reactors.
93-21	Summary of NRC Staff Observations Compiled during Engineering Audits or Inspections of Licensee Erosion/Corrosion Programs	03/25/93	All holders of OLs or CPs for light water nuclear power reactors.
93-20	Thermal Fatigue Cracking of Feedwater Piping to Steam Generators	03/24/93	All holders of OLs or CPs for PWRs supplied by Westinghouse or Combustion Engineering.
93-19	Slab Hopper Bulging	03/17/92	All nuclear fuel cycle licensees.
93-18	Portable Moisture-Density Gauge User Responsibilities during Field Operations	03/10/93	All U.S. Nuclear Regulatory Commission licensees that possess moisture-density gauges.

OL = Operating License
CP = Construction Permit

ENCLOSURE 4

OMB No.: 3150-0012
NRCB 93-03

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

May 28, 1993

NRC BULLETIN 93-03: RESOLUTION OF ISSUES RELATED TO REACTOR VESSEL WATER LEVEL INSTRUMENTATION IN BWRs

Addressees

All holders of operating licenses or construction permits for boiling water reactors (BWRs) with the exception of Millstone, Unit 1, and Big Rock Point.

Purpose

The U.S. Nuclear Regulatory Commission (NRC) is issuing this bulletin to (1) notify addressees about new information concerning level indication errors that may occur during plant depressurization, (2) request that all addressees take certain action(s), and (3) require that all addressees report to the NRC if and to what extent the requested actions will be taken and notify the NRC when actions associated with this bulletin are complete.

Background

As discussed in NRC Information Notice 92-54, "Level Instrumentation Inaccuracies Caused by Rapid Depressurization," and Generic Letter 92-04, "Resolution of the Issues Related to Reactor Vessel Water Level Instrumentation in BWRs Pursuant to 10 CFR 50.54(f)," the staff is concerned that noncondensible gases may become dissolved in the reference leg of BWR water level instrumentation and lead to a false high level indication after a rapid depressurization event. Generic Letter 92-04 requested that addressees determine the impact of potential level indication errors after a rapid depressurization event on how the plants are operated. Generic Letter 92-04 also requested that addressees take short term compensatory measures to mitigate the consequences of potential level indication errors after a rapid depressurization event and provide the staff with plans for long term corrective actions, including any proposed hardware modifications. The generic letter requested that addressees implement the long term corrective actions during the first refueling outage commencing after November 19, 1992.

The industry, through the BWR Owners Group (BWROG), requested a delay in the implementation of the long term corrective actions until a de-gas test program could be completed. The test program was intended to gather data to support the design of any necessary hardware modifications. On December 2, 1992, the staff agreed to extend the deadline for the submission of addressee plans for the long term actions to July 1993, with implementation at the earliest opportunity.

Description of Circumstances

During a normal plant cooldown on January 21, 1993, operators at the Washington Public Power System, Unit 2 (WNP-2), observed a sustained level indication error of 0.81 meters [32 inches] that gradually recovered over a period of approximately 2 hours. The licensee determined that errors of this type could result in failure to automatically isolate a leak in the residual heat removal (RHR) system during shutdown cooling operation. On April 8, 1993, the staff issued Information Notice 93-27, "Level Instrumentation Inaccuracies Observed During Normal Plant Depressurization," to discuss level indication errors that may occur during normal plant depressurization.

Discussion

Following the event reported by the licensee at WNP-2, the NRC staff requested the BWROG to evaluate the effect of level indication errors on events, such as reactor pressure vessel (RPV) drain-down, initiated from low-pressure conditions. Several paths have the potential to drain the RPV. Operator misalignment of one or more valves can establish a flow path resulting in a drain-down of the RPV. Several events of this type have occurred at operating BWRs. Automatic isolation signals based on low RPV level are normally credited for terminating these events. However, automatic isolation of the RHR system, and other systems, will not occur if there are large level errors in multiple instruments.

In response to the staff request, the BWROG submitted a report, "Supplementary Information Regarding RPV Water Level Errors due to Noncondensible Gas in Cold Reference Legs of BWRs," to the NRC on May 20, 1993. The BWROG determined that the most limiting drain-down event is an RPV drain-down to the suppression pool through the low-pressure coolant injection suction flow path. The BWROG report indicated that, for this event, the core could reach 1100 °C [2000 °F] in as little as 16 minutes if there is no makeup to the coolant system.

On the basis of the assessment of the NRC staff and the information provided by BWROG, the staff concluded that additional compensatory measures are needed for normal cooldown evolutions. Although the interim procedures currently in place are appropriate for events initiated from full power, they are not adequate for providing protection against events initiated during cooldown when automatic safety systems may be defeated by level instrumentation inaccuracies. In addition, BWROG has completed a reference leg de-gas test program. Although the data are still preliminary, initial results of the test program show that large errors in the indications from the level instrumentation are possible. This information and the event at WNP-2 confirm that the noncondensible gas problem is real and not theoretical, and that he problem applies even to slow depressurizations. Therefore, for longer term operation this problem needs to be addressed promptly with hardware modifications and immediately with compensatory measures for cooldown conditions.

Millstone, Unit 1, is exempt from this bulletin because Northeast Utilities, the licensee, has already implemented a hardware modification to prevent the buildup of noncondensible gases in the RPV level instrumentation reference legs. Big Rock Point is exempt from this bulletin because the RPV level instrumentation system installed at that facility is not susceptible to the de-gas problem described in this bulletin.

Requested Actions

1. Short Term Compensatory Actions

(a) Within 15 days of the date of this bulletin, each licensee is requested to implement the following measures to ensure that potential level errors caused by reference leg de-gassing will not result in improper system response or improper operator actions during transients and accident scenarios initiated from reduced pressure conditions (Mode 3):

(1) Establish enhanced monitoring of all RPV level instruments to provide early detection of level anomalies associated with de-gassing from the reference legs.

(2) Develop enhanced procedures or additional restrictions and controls for valve alignments and maintenance that have a potential to drain the RPV during Mode 3.

(3) Alert operators to potentially confusing or misleading level indication that may occur during accidents or transients initiating from Mode 3. For example, a drain-down event could lead to automatic initiation of high-pressure emergency core cooling systems (ECCS) without automatic system isolation or low-pressure ECCS actuation.

Facilities that are in cold shutdown during this 15 day period are requested to complete the above actions within 15 days of the date of this bulletin or prior to startup, whichever is later.

(b) By July 30, 1993, each licensee is requested to complete augmented operator training on loss of RPV inventory scenarios during Mode 3, including RPV drain-down events and cracks or breaks in piping.

Facilities that are in cold shutdown as of July 30, 1993, are requested to complete this action prior to startup from that shutdown.

All of the short term actions described above shall remain in effect until the hardware modifications described below have been implemented.

2. Hardware Modifications

Each licensee is requested to implement hardware modifications necessary to ensure the level instrumentation system design is of high functional reliability for long-term operation. This includes level instrumentation performance during and after transient and accident scenarios initiated from both high pressure and reduced pressure conditions. The hardware modifications discussed here are the same as the modifications requested in Generic Letter 92-04. Since the level instrumentation plays an important role in plant safety and is required for both normal and accident conditions, the staff requests that these modifications be implemented at the next cold shutdown beginning after July 30, 1993. If a facility is in cold shutdown on July 30, 1993, each licensee is requested to implement these modifications prior to starting up from that outage.

Reporting Requirements

Written reports are required as follows:

(1) Addressees choosing not to take the requested short term actions must submit a report within 15 days of the date of this bulletin containing a description of the proposed alternative course of action, the schedule for completing it, and a justification for any deviations from the requested actions.

(2) By July 30, 1993, all addressees must submit a report providing:

(a) the description of the short term compensatory actions taken, and

(b) a description of the hardware modifications to be implemented at the next cold shutdown after July 30, 1993. If an addressee chooses not to take the requested actions specified in the Hardware Modifications section, the report shall contain a description of the proposed alternative course of action, the schedule for completing it, and a justification for any deviations from the requested actions.

(3) Within 30 days of completion of the requested hardware modifications, a report confirming completion and describing the modification implemented.

Address the required written reports to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555, under oath or affirmation under the provisions of Section 182a, Atomic Energy Act of 1954, as amended, and 10 CFR 50.54(f). In addition, submit a copy to the appropriate regional administrator.

Backfit Discussion

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

(1) General Design Criterion (GDC) 13, "Instrumentation and control," of Appendix A to 10 CFR Part 50 which states: "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 normal cooldown or accident conditions.

(2) GDC 21, "Protection system reliability and testability," which states: "The protection system shall be designed for high functional reliability...commensurate with the safety function to be performed." The instrumentation may not be reliable during and following normal depressurization and rapid depressurization.

(3) GDC 22, "Protection system independence," which states: "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." Degassing may cause a loss of the reactor vessel water level indication function during and following normal depressurization and 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, 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 states: "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 during and following normal depressurization and rapid depressurization.

The hardware modifications discussed here are the same as the modifications requested in Generic Letter 92-04 and, therefore, the modifications are not considered to be additional backfits beyond those requested in Generic Letter 92-04. The short term compensatory actions requested by this bulletin are considered necessary to ensure that the addressees are in compliance with existing NRC rules and regulations. Therefore, this bulletin is being issued as a compliance backfit under the terms of 10 CFR 50.109(a)(4).

A notice of opportunity for public comment on this bulletin was not published in the Federal Register because of the urgent nature of the short term compensatory actions requested by this bulletin and because the hardware modifications requested are the same as those previously requested in Generic Letter 92-04.

Paperwork Reduction Act Statement

This bulletin contains information collection requirements that are subject to the Paperwork Reduction Act of 1980 (44 U.S.C. 3501 et seq.). These requirements are covered by Office of Management and Budget clearance number 3150-0012, which expires June 30, 1994. The estimated average number of burden hours is 200 hours per licensee response, including the time for reviewing instructions, searching existing data sources, gathering and maintaining the data needed, and completing and reviewing the collection of information. Send comments regarding this burden estimate or any other aspect of this collection of information, including suggestions for further reducing reporting burden, to the Information and Records Management Branch (MNBB-7714), U.S. Nuclear Regulatory Commission, Washington, D.C. 20555; and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-3019, (3150-0012), Office of Management and Budget, Washington, D.C. 20503.

Compliance with the following request for information is purely voluntary. The information would assist NRC in evaluating the cost of complying with this bulletin:

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

(2) the licensee staff 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

(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

If you have any questions about this matter, please contact the technical contact or the lead project manager listed below or the appropriate Office of Nuclear Reactor Regulation project manager.

JG Partlow

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

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(301) 504-2875

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Attachment:
List of Recently Issued NRC Bulletins

LIST OF RECENTLY ISSUED
NRC BULLETINS

Bulletin No.	Subject	Date of Issuance	Issued to
93-02	Debris Plugging of Emergency Core Cooling Suction Strainers	05/11/93	All holders of OLs for nuclear power reactors.
93-01	Release of Patients After Brachytherapy Treatment with Remote	04/20/93	Brachytherapy Licensees Authorized to Use After-loading Devices
90-01, Supp. 1	Loss of Fill-Oil in Transmitters Manu-factured by Rosemount	12/22/92	All holders of OLs or CPs for nuclear power reactors.
92-03	Release of Patients after Brachytherapy	12/08/92	For Action - Brachytherapy Licensees Authorized to use the Omnitron Model 2000 High Dose Rate (HDR) Afterloading Brachytherapy Unit For Information - None
92-01, Supp. 1	Failure of Thermo-Lag 330 Fire Barrier System to Perform its Specified Fire Endurance Function	08/28/92	For Action - All holders of operating licenses for nuclear power reactors. For Information - All holders of construction permits for nuclear power reactors.
92-02	Safety Concerns Relating to "End of Life" of Aging Theratronics Tele-therapy Units	08/24/92	For Action - All Teletherapy Licensees For Information - None
92-01	Failure of Thermo-Lag 330 Fire Barrier System to Maintain Cabling in Wide Cable Trays and Small Conduits Free from Fire Damage	06/24/92	All holders of OLs or CPs for nuclear power reactors.

OL = Operating License
CP = Construction Permit

ENCLOSURE 5

BWR OWNERS' GROUP

Cynthia L Tully, Chairperson
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BWROG-93057
May 20, 1993

Nuclear Regulatory Commission
Nuclear Reactor Regulations
Washington, DC 20555

Attention: W. T. Russell, Associate Director
Inspection and Technical Assessment

Subject: BWR OWNERS' GROUP TRANSMITTAL OF SUPPLEMENTARY INFORMATION REGARDING RPV WATER LEVEL

Enclosed is Supplementary Information regarding the safety assessment of the slow depressurization effect on RPV water level instrumentation. This material addresses the concerns outlined in IN 93-27 and other scenarios in which the Residual Heat Removal (RHR) system shutdown cooling isolation may be needed or which have the potential to drain the vessel.

The comments/positions provided in this letter have been endorsed by a substantial number of the members of the BWROG; however, it should not be interpreted as a commitment by any individual member to a specific course of action. Each member must formally endorse the BWROG position for that position to become that member's position.

Please call George Stramback (GE) at (408) 925-1913 or me if you have any questions.

Very truly yours,

G. J. Beck, Chairman
BWROG Water Level Instrumentation Committee
Tel: (215) 640-6450

EXEC6T/GJB/PJK/rt
Enclosure

cc: AC Thadani, NRC
CL Tully, BWROG Chairperson
LA England, BWROG Vice Chairman
BWROG Primary Representatives
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NRC Document Management Branch
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Supplementary Information Regarding RPV Water Level Errors due to Noncondensable Gas in Cold Reference Legs of BWRs

Reviewed by: [signature]
G.B. Stramback
Safety Evaluations Programs Manager

Approved by: R. C. Mitchell
R.C. Mitchell
Safety and Communications Projects Manager

General Electric Company
San Jose, California
May 1993

~~9306280145~~ 28pp.

IMPORTANT NOTICE REGARDING
CONTENTS OF THIS REPORT

Please Read Carefully

The only undertakings of General Electric Company (GE) respecting information in this document are contained in the contract between the Boiling Water Reactor Owners' Group (BWROG) and GE, as identified in the respective participating utilities' BWROG Standing Purchase Order for the performance of the work described herein, and nothing in this document shall be construed as changing those individual contracts. The use of this information except as defined by said contracts, or for any purpose other than that for which it is intended, is not authorized; and with respect to any other unauthorized use, neither GE or any of the contributors to this document makes any representation or warranty, and assumes no liability as to the completeness, accuracy, or usefulness of the information contained in this document.

Executive Summary

This report was prepared as a result of an event which occurred at WNP-2. The event highlighted the importance of considering the potential for reactor water level instrumentation errors resulting from degassing of cold reference leg instrumentation following routine depressurization of the RPV. Although the principal concern is with the ability to terminate a reactor draindown or mitigate an RHR piping failure by automatic isolation of the RHR shutdown cooling line, other trip settings also are potentially affected.

A highly conservative analysis with degraded instrumentation shows that sufficient time is available for operators to respond to a draindown event. The analysis concludes that with proper operator awareness of the potential for RPV level indication errors, degassing can be recognized and actions taken such that there is no undue safety significance during hot shutdown. Based on this evaluation, it is concluded that a substantial safety hazard does not exist.

Table of Contents

Supplementary Information
Regarding RPV Water Level Errors
due to
Noncondensable Gas in Reference Legs of BWRs

1.0 Background

1.1 Previous Notching Studies

The issue of notching of the reactor water level instrumentation signals due to the degassing of noncondensable gas in BWR cold reference legs has been widely reviewed in response to Nuclear Regulatory Commission (NRC) concerns expressed in Generic Letter 92-04[1]. Initial review by the BWR Owners' Group[2] (BWROG) has concluded that the issue of noncondensable gas does not pose a substantial safety hazard on operating BWRs following a rapid depressurization from normal operating conditions.

However, during an investigation of an event at Washington Nuclear Project, Unit 2, (WNP-2), January 21, 1993, Washington Public Power Supply System (WPPSS) personnel identified a safety function of the water level instrumentation not previously addressed within the scope of the BWROG report and not previously discussed with the NRC. The WPPSS concern was that the auto closure function of shutdown cooling (SDC) isolation valves may be lost due to erroneous indication resulting from degassing of noncondensable gas in Reactor Pressure Vessel (RPV) water level cold reference legs. This concern occurs at low reactor pressures while all previous evaluations addressed effects resulting from rapid depressurization from near rated conditions.

Since this event, the NRC has issued Information Notice 93-27, "Level Instrumentation Inaccuracies Observed during Normal Plant Depressurization"[11].

1.2 WNP-2 Event Summary[3]

WNP-2 was in a hot shutdown condition following a reactor scram. The plant was being depressurized in accordance with normal plant procedures. Although some minor notching was recorded as the depressurization proceeded, when SDC was initiated a more significant increase occurred on the narrow range channel C of water level instrumentation. With the initiation of SDC, RPV pressure reduced from about 20 psig to 10 psig and Channel C experienced significant degassing which resulted in a transient increase in indicated RPV level of about 32 inches on narrow range instrumentation. The level indication recovered to within 6" in approximately 25 minutes and fully recovered within 2 hours.

During an evaluation of the safety significance of this event at WNP-2, WPPSS personnel identified that this response on Channel C could have delayed closure of the SDC isolatio ves. These valves receive an isolation signal based on RPV low water level (Level 3) to prevent a loss of RPV inventory in the event of a moderate energy break in a Residual Heat Removal (RHR) line while the system operates in shutdown cooling. WPPSS personnel concluded that the event was not safety significant due to instrumentation in the WNP-2 design which would have alerted the operators of RHR room flooding with sufficient time for operator action to isolate the RHR system. In addition WNP-2 is conducting hourly flood tours of RHR pump rooms.

1.3 Purpose of Evaluation

The purpose of this evaluation is to provide the BWROG with supplementary information on the safety significance of degassing of RPV water level cold reference leg instrumentation following routine slow depressurization. While the evaluation was initiated based on the WNP-2 event, it also considers other scenarios in which the RHR shutdown cooling isolation function may be needed or for which the potential to drain the RPV exists.

2.0 Evaluation of Safety Significance

GE-NE has conducted a comprehensive review of the safety significance of water level instrument errors resulting from cold reference leg degassing in BWR/3, BWR/4, BWR/5 and BWR/6 product lines during SDC operation. The importance of this concern is highlighted by recent plant observations and conservative BWROG testing which show that, depending on plant specific configurations such as instrument line size and routing, a significant amount of degassing can occur as the RPV is depressurized below 50 psig. Many BWRs would be expected to be affected to a smaller extent than portrayed in this evaluation. Nevertheless, sufficient degassing can affect the automatic instrument trip functions for all instruments sharing a common cold reference leg.

BWR/1 and BWR/2 plants do not rely upon cold reference leg instrumentation for safety system initiation and are therefore not specifically addressed by this evaluation. However, the effect of degassing on any long term actions required based upon post-accident monitoring instrumentation in BWR/1 and BWR/2 plants is addressed by existing Emergency Procedure Guidelines (EPGs)[6] and previous issued BWROG communications to plant operators[9,10].

2.1 Applicable Events for Consideration

2.1.1 Applicability of Initial Operating Conditions

During OPCON 1 (Power Operation) the reactor mode switch is in the RUN position which requires reactor pressure to be greater than about 750 psig (in most plants); this operating condition was considered as the initial condition in the previous BWROG Safety Assessment[2].

In OPCON 2 (Startup/Hot standby), the reactor is critical and RPV pressure could be at any value up to normal operating pressure. If the reactor has been recently started up it is likely that noncondensable gasses will not have had sufficient time to accumulate in the RPV water level cold reference legs. In addition, any potential events initiating from OPCON 2 are considered to be bounded by the OPCON 3 evaluation.

During OPCON 3 (Hot Shutdown), events that cause a rapid reduction in the RPV water level may be the most significantly affected by the occurrence of RPV water level errors. This operating condition also is selected for further evaluation because it corresponds to the condition in which the most significant degassing has been observed to occur.

OPCON 3 is normally of short duration in comparison to other operating conditions. At a reactor cooldown rate of 100°F/hr the transition from 135 psig to cold shutdown (OPCON 4) could be as short as 2.5 hours; at a slow reactor cooldown rate of 25°F/hr the transition could take 8 hours or more. Because a rapid reduction in RPV level can occur at any time during OPCON 3, the safety significance of degassing is reduced significantly by the limited time frame under which a rapid reduction in RPV level could occur.

In OPCON 4 (Cold Shutdown) or OPCON 5 (Refueling), the plant is depressurized and any degassing would have already occurred. Any events which occur during OPCON 4 or OPCON 5 would be bounded by consideration of the OPCON 3 evaluation. Therefore it is not necessary to consider water level errors due to degassing during these conditions any further in this evaluation.

2.1.2 RPV Draindown Events

Valve mispositioning events which cause reductions in RPV water level have occurred during cold shutdown or refueling operating conditions and in all cases they have been isolated by RPV level instrumentation logic. GE Service Information Letter (SIL) 388[4] discussed the applications of valve interlocks to limit the occurrence of draindown events. The Institute for Nuclear Power Operations (INPO) Significant Operating Experience Report (SOER) 87-2[5] discussed seven draindown events occurring between 1983 and 1986.

In 1990, the BWROG investigated draindown events occurring during cold shutdown (OPCON 4) conditions. A summary of the results relevant to this evaluation are included in Attachment A. These results are approximately applicable to the OPCON 3 condition, but are non-conservative because they do not account for the effect of RPV pressure on the rate of draindown. It is evident from Attachment A that the level 3 RHR isolation function is an important safety feature for avoiding the low RPV water level safety limit for draindown events. Due to the non-conservatism, this importance is heightened during OPCON 3.

2.1.3 Piping Failures

At pressures below about 150 psig, only piping leaks are normally required to be considered in the licensing basis. Nevertheless, piping failures can occur in any pressurized line during OPCON 3 and have been considered in this evaluation. Such piping failures are considered to have an extremely low probability of occurrence due to the low system pressure and margins inherent in reactor system piping design.

Piping breaks, instrument line breaks or a recirculation pump seal failure inside primary containment can initiate high drywell pressure trips or low-low reactor water level (Level 2) trips which provide protective functions for initiation of some Emergency Core Cooling (ECC) systems.

Piping failures outside of containment are in general protected by isolations derived from the RPV level instrumentation. As indicated in Attachment B, these include the RHR shutdown cooling lines (Level 3) and Reactor Water Cleanup (RWCU) lines (Level 2 or Level 3 on some plants).

2.2 Instrumentation Considerations

2.2.1 RPV Level Instrumentation Configurations and the Effect of Degassing

The configurations of RPV level instrumentation are indicated in section 4.0 of the BWROG report[2]. This section discusses the effect of degassing on the configuration and trip logic for the various BWR product lines. Figure 1, based on a BWR/6 design, shows the various narrow range and wide range instruments used to provide protective trips. Fuel zone, shutdown range and post accident monitoring channels used for operator awareness also are shown. All BWRs are similar except most BWR/3 and BWR/4 plants contain two cold reference legs rather than four.

The effect of degassing and the resultant water level signal bias on these trip functions is to delay the actuation of safety functions during postulated RPV water level reductions. If an extended response occurs and actual water level remains below the reactor vessel instrument tap for the variable leg of an instrument, the trip functions associated with that instrument will not occur regardless of the actual water level response in the RPV. Therefore, an important consideration in this evaluation is recognition of the margin between the trip function setpoint and the RPV variable leg tap in comparison with the size of the postulated RPV water level response and its duration.

2.2.2 Typical SDC Isolation and other Water Level Setpoints

Water level setpoints were reviewed for different BWR product lines to assess the margin available to accommodate responses in indicated RPV water level. The results of this review are provided in Table 1.

As shown in Table 1, BWR/4, BWR/5 and BWR/6 plants typically have between about 20 and 23 inches of margin between the level 3 trip and the narrow range variable leg instrument tap. Although a smaller margin exists between the Level 1 trip and the wide range instrument tap, the Level 3 trips are the first line of defense against a reduction in RPV water level during OPCON 3 and are the focus of this evaluation.

All BWRs use the safety grade narrow range instruments (see Figure 1) to provide a low water level (Level 3) isolation signal in addition to other reactor protection signals. This function provides protection against a draindown event. Remote manual and high system pressure automatic isolation also are provided in all BWRs.

It also can be seen from Table 1 that a small margin exists of about 8 inches between the Level 2 trip and the instrument tap for BWR/3 plants. BWR/3 plants use the Level 2 signal for initiation of Core Spray systems, actuation of the Automatic Depressurization System (ADS) and closure of Main Steamline Isolation valves (MSIVs). A much larger margin exists to the Level 3 trip.

2.3 Assumptions for Analysis

Based on the consideration of potential events and concerns associated with degassing, conservative assumptions were selected to make the results of this evaluation applicable generically to BWR product lines.

2.3.1 Initial Conditions

Initial RPV water level is assumed to be controlled by the plant operator in the normal control band as indicated on narrow range instrumentation (between the Level 4 and Level 7 alarms). RPV pressure is between 0 psig and 135 psig and decreasing consistent with a normal plant cooldown.

The cooldown and depressurization are assumed to be accomplished by the RHR system in shutdown cooling mode. The MSIVs are assumed to be closed. Initiation of SDC may occur at any pressure between zero and the RHR shutdown cooling interlock which ranges between 75 psig and 135 psig. Therefore this assumption is realistic.

The reactor is assumed to have been shutdown for about 3 hours and core power is in the decay heat range. This assumption is realistic based on expected conditions following a reactor scram. For a routine shutdown with manual control rod insertion, decay heat would normally be less than that considered in this evaluation.

2.3.2 Effects of Degassing

Degassing of the cold reference legs following a routine depressurization is assumed to have occurred and affected both wide range and narrow range level instruments sharing a common reference leg. It is assumed that this degassing causes a sustained instrumentation bias with a magnitude in excess of the difference between the Level 3 trip setpoint and the narrow range tap (BWR/4, BWR/5 and BWR/6) or the Level 2 trip setpoint and the wide range tap (BWR/3).

For purpose of evaluation a bias of 30" was arbitrarily assumed based on the typical values shown in Table 1. This magnitude of bias was selected to be sufficient to initiate Level 7 alarms and Level 8 high pressure makeup system trips. It is further assumed that the magnitude of the bias is less than an amount which would allow these setpoints to clear if level subsequently lowers.

This assumption is conservative because the response of RPV water level indications has typically been observed to be smaller than this amount of bias and of short duration. The longest observed response occurred during the WNP-2 event (25 minutes). Preliminary results of the BWROG test program based on a slow depressurization of the WNP-2 reference leg C configuration are summarized in Table 2. These data support the assumption that an extreme bias does not occur.

All water level instrument channels are assumed to be affected and result in a simultaneous bias of the water level instrumentation. This assumption is extremely conservative relative to plant experience, but has been made to emphasize the inherent safety of the existing instrumentation. It is unlikely that all cold reference legs experience the same degree of bias due to degassing because of differences in line routing and amount of leakage. Nevertheless, the simultaneous loss of all channels is not eliminated from consideration.

2.3.3 RPV Instrumentation Availability

For BWR/4,5 and 6 plants, Level 2 trips are assumed to be available; Level 1 and Level 3 trips are assumed to be unavailable due to the magnitude of the bias. Level 2 trips are operable because the amount of bias due to degassing also is assumed to be less than the difference between the level 2 trip and its instrument tap or about 112".

For BWR/3 plants, Level 3 trips are assumed to be available; Level 2 trips are assumed to be unavailable. Level 3 trips are available assuming the amount of bias due to degassing is less than the difference between the Level 3 trip and its instrument tap or about 66". The data from the BWROG tests shown in Table 2 support this assumption.

2.3.4 Initiating Event

A draindown of the RPV is assumed to occur due to inadvertent opening of the Low Pressure Coolant Injection (LPCI) suction path. This occurrence is unlikely because most BWRs have installed interlocks or administrative controls to reduce the probability of this event occurring. Such interlocks and controls are discussed in SIL 388[4] and INPO SOER 87-2[5]. Because the draindown event can result in a rapid RPV inventory loss, the draindown was considered the most limiting event from the standpoint of operator response.

For BWR/3 plants an additional potential initiating event is a piping break inside containment. This event is considered to be more limiting than a draindown event due to the unavailability of the Level 2 trips used for ECC system initiation even though reduced system pressure makes such events extremely unlikely during OPCON 3. Nevertheless, the ECC systems would be automatically initiated by high drywell pressure in response to breaks inside containment and no operator actions are required. This is addressed by the previous BWROG evaluation[7]. Therefore piping breaks for BWR/3 plants do not need to be evaluated further.

2.4 Safety Significance

2.4.1 Operator Actions

The following discussion is based on a review of the Emergency Procedure Guidelines[9] (EPGs). Available RPV level indication normally consists of narrow range normal indication, wide range or post-accident monitoring instrumentation from a cold reference leg instrument. In addition, fuel zone and shutdown range instruments which use other reference legs are available for additional guidance in following the EPG steps.

When degassing occurs the initial RPV water level response is an assumed 30" increase in indicated water level. Following receipt of the high water level alarms and trips (Level 7 and Level 8), no EPG entry condition is satisfied. Nevertheless, operators would be expected to initiate investigation into the cause of the alarms and trips.

Following initiation of the draindown event, when water level lowers below the indicated RPV scram point (Level 3), the RPV Level Control guideline (RC/L) is entered. RC/L requires manual initiation of the emergency diesel generators, closure of containment isolation valves and initiation of ECC equipment as needed. RPV level is required to be maintained in the normal range or at least above the top of active fuel (TAF). In both cases the safety significance relies upon an operator assessment of RPV water level indication availability and the time available for operator action.

For BWR/3 plants, the RHR valves automatically isolate in response to the level 3 trips to terminate the draindown. Automatic initiation of ECC systems from the level 2 trip would not be necessary. Fuel Zone RPV level indication may be used to provide indication that water level can be restored by plant operators using normal makeup sources. If RPV level cannot be determined, RPV flooding (Contingency 4) is required by the EPGs. Such conditions are addressed in a utility's operator training program.

For BWR/4, BWR/5, and BWR/6 plants, EPGs require remote manual isolation of the RHR SDC line when it has been determined that the Level 3 signal has not occurred. Additional RPV level indication is available from the wide range instruments, post accident monitoring instrumentation or fuel zone instrumentation.

2.4.2 Evaluation of plant response

Because the draindown is automatically isolated in BWR/3 plants, the following discussion applies to the BWR/4, BWR/5 and BWR/6 product lines. As indicated in Attachment A, only a few minutes are available before reaching TAF following the most severe draindown event from an initial normal water level. Plant operators have been sensitized to the water level degassing phenomenon[7,10] and several alarms would alert the operator to take action.

Prior to any draindown event, plant operators would normally be controlling RPV water level as indicated by the narrow range instrumentation. The earliest indication to alert plant operators would either be the high water level alarms (level 7) or trip signals (level 8) when the degassing occurred. Regardless, as RPV level lowers, the low water level alarm (level 4) would occur unless an extreme bias has occurred.

Sufficient information is available to alert the operator to take action in accordance with the EPGs, but the closure of the LPCI suction valve may take up to one minute for a full stroke (see Attachment B). If isolation has not occurred, High Pressure Core Spray (HPCS) would initiate (for BWR/5 and BWR/6 plants) to begin to restore RPV level. However, even at the HPCS pump "run-out" flow, the maximum draindown flow rate, shown in Attachment A, can exceed the capacity of the HPCS system. Therefore, the draindown could proceed until RPV level has lowered to the top of the jet pumps (about 2/3 core height).

Without operator action (or HPCS operation in BWR/5 and BWR/6 plants) to provide makeup, RPV water level would continue to lower below the 2/3 core height due to steaming. Core heatup would eventually commence and result in fuel damage if reactor makeup is not restored. GE studies, conducted to support early revisions of EPG contingencies[7], indicate that without coolant makeup a core temperature above 2000°F can occur in about 15 minutes after reaching the 2/3 core height level.

Additional operator actions which could be taken as an alternate to isolation include initiation of ECC systems in accordance with EPGs or alternatively increasing makeup flow from the control rod drive (CRD) system in accordance with SIL 200[8].

2.4.3 Safety Significance

For BWR/3 plants, degassing of cold reference legs causes the unavailability of Level 2 trips used for ECC system initiation. Following a drywell piping failure safety is assured by diverse instrumentation which initiates the ECC equipment. For a draindown event, isolation is highly probable.

In BWR/4, BWR/5 and BWR/6 plants, degassing of cold reference legs may cause the unavailability of both Level 3 and Level 1 automatic trips. Although a RPV draindown event can lower RPV level from Level 3 to TAF in about 2 minutes, a total of more than 17 minutes are available for operator recognition of the draindown, isolation of the line or initiation of a core makeup system before core damage might occur. This time is sufficient to assure operator action, especially considering instructions already provided to plant operators[9,10]. Based on this available time for operator response, and the low probability of the event, the unavailability of the Level 3 and Level 1 trip logic does not have an undue safety significance for RPV draindown events.

Conclusions

This safety evaluation addresses the significance of the potential loss of RPV water level instrumentation resulting from degassing of the cold reference legs during hot shutdown and low reactor pressure.

The evaluation of plant response to a highly conservative degradation of the RPV water level instrumentation during a RPV draindown event shows that alternate water level indication is available for operator recognition of the need for action to restore core cooling. Furthermore, at least 17 minutes is available for the operator to initiate core cooling. Through several BWROG communications[2,9,10], BWR operators are sensitized to the potential for a degassing effect which further enhances their awareness of needed operator action.

Therefore, it can be concluded that with proper operator training and awareness of RPV level indication, degassing can be recognized and actions taken such that there is no undue safety significance during hot shutdown (OPCON 3). Based on the preceding evaluation, it is concluded that a substantial safety hazard does not exist.

1.0 References

1. Resolution of the Issues Related to Reactor Vessel Water Level Instrur   ation in BWRs Pursuant to 10CFR50.54(f), Generic Letter 92-04, October 16, 1992

2. BWR Reactor Vessel Water Level Instrumentation, GENE-770-15-0692, August 1992

3. Existence of Noncondensable Gases in the Reference Legs of Reactor Pressure Vessel Instrumentation, WPPSS LER 93-009, March 19, 1993

4. RHR Valve Misalignment during Shutdown Cooling Operation for BWRs 3,4/5 and 6, SIL 388, February 1983

5. Inadvertent Draining of Reactor Vessel to Suppression Pool at BWRs, INPO SOER 87-2, March 1987

6. BWR Owners' Group Emergency Procedure Guidelines, Revision 4, NEDO 31331, March 1987

7. Evaluation of EPG Contingency No. 3   'E memo Varzaly to Rogers, August 4 1981

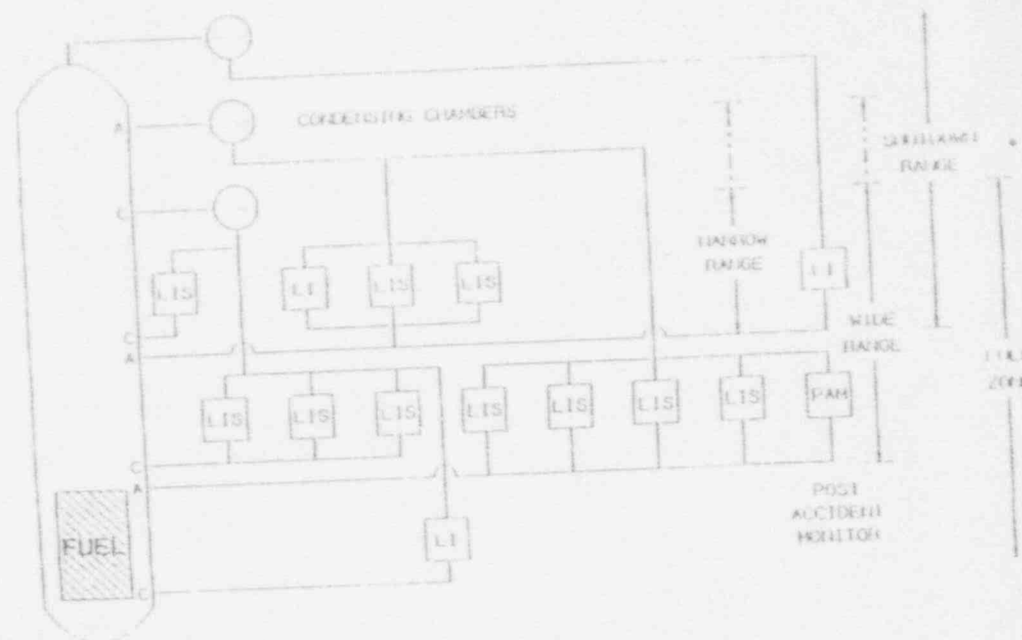
8. CRD Hydraulic Control System Return Line Modification, SIL 200, July 1979

9. Effect of Noncondensable Gases on BWR Cold Reference Leg RPV Water Level Instruments, BWROG-92096, October 16, 1992

10. Effect of Noncondensable Gases on BWR Cold Reference Leg RPV Water Level Instruments, OG92-937-62, August 19, 1992

11. Information Notice 93-27, "Lev   Instrumentation Inaccuracies Observed during Normal Plant Depressurization", April 8, 1993

Figure 1 - Typical BWR/6 RPV Water Level Indications



NOTE: Figure shows Channels "A" & "C" only. Channels "B" & "D" are similar

Table 1
<u>Typical RPV Water Level Trip Setpoints</u>

Trip Point and Typical Action	BWR/3	BWR/4	BWR/5	BWR/6
Trip Level 8 - High Pressure System Trip	+48.0"	+44.0"	+54.5"	+52.0"
Trip Level 7 - High Water Level Alarm	+44.0"	+29.0"	+40.5"	+38.8"
Trip Level 4 - Low Water Level Alarm	+19.0"	+17.0"	+31.5"	+30.8"
Trip Level 3 - Low Water Level Isolation & Scram	+9.0"	+2.0"	+13.0"	+8.9"
Instrument Zero	0	0	0	0
Narrow range inst. tap	n/a	- 21.0"	- 10.5"	- 11.6"
Trip Level 2 - High Pressure ECC initiation		-48.0"	-50.0"	-45.5
Trip Level 1 - Low Pressure ECC initiation, ADS, MSIV closure	-49.0"	-160.0"	-129.0"	-145.5
Wide range inst. tap	-57.0"	-172.0"	-161.5"	-162.6"

n/a = not applicable

Table 2
BWROG Test Program
Slow Depressurization Tests
of
WNP-2 Reference Leg C Configuration

mber	Reference Leg O_2 Concentration (ppm by Volume)	Approximate Max. Transient Error (inches)	Residual Error (inches)
90	715	29	7.1
91	327	17	7.1
93	[illegible]	< 5	0.8
94	179	< 5	-1.8

Note: Test results are unverified

Attachment A

OPCON 4 RPV Drain-down Analysis for BWR/4 and BWR/6 Plants

Introduction

Current BWR/4 and BWR/6 standard technical specifications (TS 3.5.2) require two ECC systems to be operable. The basis for the specification is mitigation of an undefined RPV drain-down event. Because there is no quantitative basis for this requirement, this analysis is conducted to provide functional requirements for makeup systems during OPCON 4 based on conservative assumptions.

Although this evaluation is based on BWR/4 and BWR/6 configurations, by adopting a wide range of assumptions for the analysis, the results also are considered to be applicable to BWR/3 and BWR/5 product lines.

Methodology

Drain-down or pump-down pathways with the potential to drain the RPV were reviewed to determine the expected worst case conditions for analysis. This evaluation is included in Attachment B. In this review, piping system breaks were not postulated because of the low system pressure during shutdown. Also, maintenance activity on piping systems which could result in leaks were not considered because it was judged that such pathways would be observed by maintenance personnel. Therefore, the evaluation in Attachment B focuses on potential valve misalignment which could result in limiting conditions for analysis. Based on this review two pump down and one drain down cases were selected for the analysis.

The analysis is based on first principles of fluid flow without credit for any makeup to the RPV. The evaluation estimates the time available before a drain down to the top of active fuel occurs, the required amount of RPV makeup flow to recover without isolation and the time before mitigating action is needed for various configurations of potential drain down and pump down paths. The RPV level response is evaluated following the inadvertent opening of a valve in a line providing diversion from SDC to the suppression pool. Steady state conditions of flow are determined. Therefore valve opening times are ignored. This is conservative from the standpoint of action time.

Fluid flow is based on Bernoulli's equation. The form used is:

$$Q^2 = 2g(L_{rpv}-L_{sp})/(K/A^2) \qquad (1)$$

Where: Q = Flow rate
g = Gravitational constant
L = RPV and Suppression pool levels
K = cumulative loss coefficient in the line
A = Effective drain path area

For the pump down analyses, a constant flow through the pump was assumed. The piping configuration was used as input to determine the flow split between the suppression pool and the return of the fluid to the RPV. For both the pump down and drain down analyses, the flow to the suppression pool is considered the required makeup flow.

During OPCON 4 technical specification safety limit 2.1.4 (NUREG 0123) requires that RPV level be maintained above the top of active irradiated fuel. Therefore the conservative acceptance criteria for these analyses is initiation of actions to prevent violating this safety limit.

Assumptions

1. The initial RPV level was assumed to be at the normal operating level specified in plant FSAR.

2. For the drain-down case, the LPCI suction valve is assumed to instantaneously open to its full open position. This assumption is considered conservative.

3. The flow rate is assumed to instantaneously achieve the steady state flow rate based on the initial RPV level. No time is considered which accounts for the acceleration of the flow from the zero flow condition.

4. Table A-1 summarizes the parameters assumed in the analysis based on typical BWR/4 and BWR/6 designs. Although individual plants will have values slightly different from these assumed values, it is not expected that plant specific analyses would show results significantly different than are evaluated in this study.

Results and conclusions

Table A-2 summarizes the results of various analyses of different events in the typical configurations described above. The "time" is the time, in minutes, from initiation of the draining to the top of active fuel.

Table A-1
Drain-Down Analysis
Evaluation Assumptions

	BWR/4	BWR/6
PLANT ELEVATION DATA* (ft)		
Fuel Pool Surface	682.5	207.8
Transfer Canal Bottom	659.5	184.5
RPV Flange	659.5	184.5
Normal Water Level	647.1	167.5
RPV Isolation (Level 3)	645.3	165.8
Top of Active Fuel	630.8	150.9
RHR Return to RPV	599.7	161.5
RPV to RHR outlet	589.5	134.0
RHR Suppression Pool Return	555.3	109.5
Normal Suppression Pool Level	557.5	111.9
RHR Pump Suction	541.0	93.0
RHR Heat Exchanger Outlet	541.0	104.6
* Reference plant elevations above sea level		
CROSS-SECTIONAL DATA		
RPV upper diameter (ft)	34.0	34.0
RPV lower diameter (ft)	18.5	18.4
Storage Pool Area (sq ft)	1360	1156
PUMP FLOW DATA		
RHR Design Pump Flow (gpm)	10000	7500
OPERATIONAL DATA		
RPV Water Temperature (°F)	140	140
RPV Pressure (psia)	14.7	14.7
Containment Pressure (psia)	14.7	14.7

The flows provided are the design pump rates, where applicable, and, for the drain down case, the gravity drain rate at the time RPV level is at Level 3. This point was arbitrarily selected to compare the drain-down rate at the time of shutdown cooling line isolation with makeup system capacity.

For the two pump down cases, the flow split between the return line and the pathway to the suppression pool was determined. It was found that, due to the RPV head and pressure drops in the line that all pumped fluid passed through the drain path and none returned to the RPV. Therefore pump-down times were calculated simply as a function of flow rate and initial inventory.

The rapid loss of RPV level which occurs in OPCON 4 LPCI suction drain-down is believed to be somewhat, but not excessively, conservative due to the assumptions made in the analysis. The period between the isolation actuation signal at RPV Level 3 and the top of active fuel is about 70 seconds for the BWR/4 evaluation and 100 seconds for the BWR/6 evaluation.

Because of this short period and the high flow rate (which exceeds the ECC system makeup capacity from a single pump), interlocks were recommended in SIL 388 to prevent opening of the SDC suction valves if the LPCI suction valve is not fully closed. Although most BWRs have interlocks which prevent opening the SDC suction valves if the LPCI suction valve is open, variations exist in the designs of these interlocks and inadvertent manual opening while the SDC pump is shutdown can still occur.

If the interlock fails or if inadvertent manual operation occurs, isolation of either isolation valve within 60 seconds beginning at RPV Level 3 is adequate to prevent exceeding the safety limit.

ATTACHMENT B
POTENTIAL RPV DRAINDOWN PATHS
DURING OPCON 4 & 5

Identification of Pathways

Plant technical specifications require that under certain conditions "operations that have a potential for draining the reactor vessel" be suspended. These operations are not defined, but are left to plant operators judgement. In order to identify the limiting pathways for analysis, potential reactor pressure vessel (RPV) drain-down paths existing were reviewed. Table B-1 summarizes the potential drain paths that were identified from a review of typical BWR/4 and BWR/6 isolation diagrams and P&IDs. In each case the potential pathway is identified and the isolation closure time provided, if isolation is available.

SIL 388, "AID-67 RHR Valve Misalignment During Shutdown Cooling Operation for BWRs 3/4/5 and 6", issued February 1983, discussed several RHR pathways and issues associated with operator error during reactor system shutdown cooling (SDC) to highlight the importance of operator training and operator compliance with plant operating procedures. These paths are included in the evaluation. In addition, the paths identified in INPO Significant Operating Experience Report, SOER-87-2, "Inadvertent Draining of Reactor Vessel to Suppression Pool at BWRs", dated March 19, 1987, were considered.

Other pathways that were identified either contain an inboard check valve to prevent the drain down or are in paths which exit the RPV above the top of active fuel and are therefore not a concern.

Discussion of Pathways

RPV Connections (Paths 1,2,3,4 and 5)

Pathways 1 and 2 require a piping break to occur and can generally be excluded from further consideration during OPCON 4 and 5 since the RPV is at low pressure and the potential for a break is diminished. If maintenance is performed on seismic restraints on unisolated portions of piping (or temporary shielding is attached to piping which exceeds the seismic loading), a drain path could develop following a seismic event during shutdown. Such a situation is judged to be remote and would normally be administratively controlled and therefore not a limiting concern.

During maintenance multiple lines could also be opened to create a drain path. However, such work would normally be administratively controlled and the small line size of path 1 (about 1") make these paths of minimal concern.

Table A-2
OPCON 4 RPV Drain-down
Analytical Results

	BWR/4		BWR/6	
DRAIN PATH	TIME (min)	FLOW (gpm)	TIME (min)	FLOW (gpm)
LPCI SUCTION*	1.3	27000	1.8	20600
RHR TEST LINE	3.0	10000	3.9	7500
RHR MIN FLOW	30.5	1000	39.4	750

* Drain down event

The bottom head drain path to Radwaste (Pathway 3) and the RPV sample lines
(Pathway 4) each require specific operator action to unisolate two series valves in the
line. Although this could occur due to a valving error, it is judged to be remote.
Furthermore, the line size (typically less than 2") limits the potential for a rapid loss
of RPV inventory.

A similar situation occurs with the CRD discharge path (Pathway 5), except that the
equivalent line size is much smaller and highly restricted. During CRD mechanism
removal a potential drain path exists. However, this activity is normally well
coordinated with the control room, is limited to one mechanism at a time, is normally
of short duration and has a small effective area for drainage. Consequently this is not
considered a potentially limiting path.

RWCU Blowdown (Pathway 6)

The potential pump-down of the RPV through the RWCU blowdown valve exists.
With either the bottom head drain or normal suction open, opening of the blowdown
valve to the main condenser (F034) or Radwaste (F035) results in a drain path.
Normally both valves can be open during OPCON 4. Redundant isolation valves
(F001 and F004) would close in 10 to 35 seconds when RPV level falls to RPV level
2 (Level 3 in some plants). The line path is long and contains significant restrictions
to flow through five series sets of heat exchanger tubes, a demineralizer and two
globe valves. The maximum flow rate through this pathway is equal to the RWCU
flow rate: about 240 gpm in a BWR 4 type plant, about 480 gpm for BWR/6.

RHR SDC Suction to Suppression Pool (Pathway 7)

The shutdown cooling mode of the RHR system is a manually operated mode. It is
placed into operation by the operator during OPCON 3 when the RPV pressure has
reached a predetermined level (about 135 psig for most BWR product lines), and the
RPV water level is above level 3. BWR field experience has shown that, during
shutdown cooling operation, operators, on occasion have inadvertently opened
non-shutdown cooling RHR valves which has resulted in a RPV drain-down event.
The correct procedures for valving the shutdown cooling mode are specified in the
RHR operating procedures presented in the GE Operation and Maintenance Instruction
(O&MI) manual, and if followed, such errors are normally avoided.

Selected valve interlocks are provided in the RHR system to minimize valve
misalignments and other operator errors during SDC. However, in order to provide
flexibility, a comprehensive valve interlock arrangement has not been provided for all
the RHR valves that could be actuated during shutdown cooling. A brief description
of some of the possible operator errors which could occur, and the possible events
following the error, are described in the discussions which follow.

a. Test Return Line (F024) or Torus Spray (F027)

RPV water can be pumped or drained to the suppression pool if the RHR test return line valve (F024) or the drywell spray valve (F027) is inadvertently opened during SDC operation. This flow condition, which is applicable to BWR 3 through 6, can cause significant loads on the containment penetration, piping, and pipe supports. The maximum flow rate through this pathway is equal to the design flow of a single RHR loop, or about 10,000 gpm. If the flow is split between the bypass path and the normal return path the drain-down rate could be less than rated.

b. Minimum Flow Bypass Line (F064)

Prior to initiating SDC, the RHR piping is warmed up by using RPV water. During this period the minimum flow bypass line valve (E12-F064) is normally closed. However, an operator can inadvertently open this valve, which would allow some flow to the suppression pool. At this stage, the flow through this line is probably below the low flow setpoint, and the valve will not automatically close. This flow condition, also applicable to BWR 3 through 6, could cause additional loads on piping and pipe supports. The flow rate through this line is limited by the piping. The maximum flow rate is about 750 gpm.

c. Suppression Pool Cooling Line (F026)

Pump-down of the RPV through the suppression pool cooling line is possible (Note: On BWR/3 and some BWR/4 and BWR/6 plants, this line is combined with the test return line). With the RHR system in operation in SDC mode, opening of the suppression pool cooling return valve (F026) results in a drain path. The line is fairly long and contains restrictions through a pump and a single heat exchanger, but fewer restrictions than the test return line. The maximum flow rate through the pathway is the same as for the test return line.

d. LPCI Suction Valve (F004)

On BWR/3 (with RHR) to BWR/6, with the SDC valves open a drain path is established if the normally open LPCI suppression pool suction valve (F004) is opened after establishing SDC. A valve interlock is included on most BWRs to minimize operator errors by preventing the SDC suction line valve (F006) from opening if the LPCI suction valve (F004) is not fully closed. However, an operator can inadvertently open F004 after SDC has been established. This path has very little restriction and is driven by the head of water in the RPV.

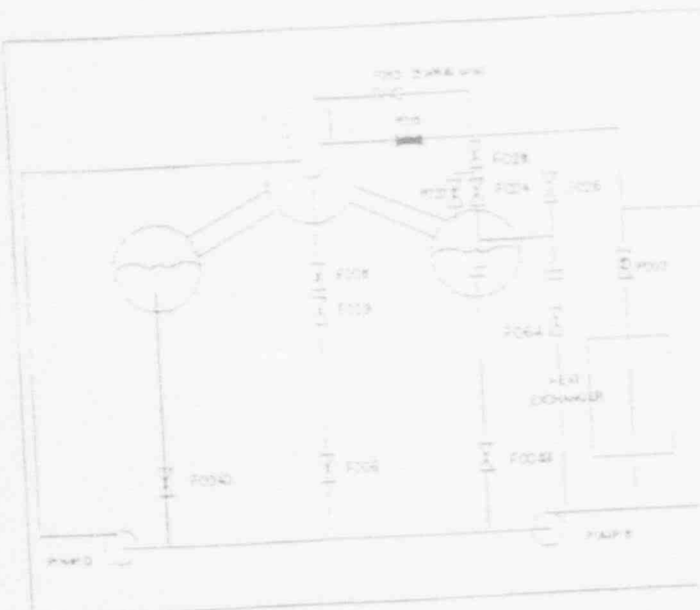


Figure 1 - Typical RHR System

TABLE B-1
POTENTIAL DRAIN-DOWN PATHWAYS

POTENTIAL PATHWAY	ISOLATION TIME	
	BWR/4	BWR/6
1. RPV INSTRUMENT LINE	NONE	NONE
2. RECIRC SUCTION OR RISER	NONE	NONE
3. RPV BOTTOM HEAD DRAIN W/NORM.	MANUAL	MANUAL
4. RPV SAMPLE LINES TO RADWASTE	15 SEC	UNKNOWN
5. CRD HYDRAULICS	UNKNOWN	N/A
6. RWCU BLOWDOWN	10 SEC	35 SEC
7. RHR SDC SUCTION TO SUPP POOL	52 SEC	40 SEC