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SUBJECT: LER 93-009-01:on 930217,existence of noncondensible gases in leg of RPV could effect RPV Level 3 isolation occurance. Caused by design deficiency.Criteria developed for operability of RPV logic for RHR isolation.W/930517 ltr.

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U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: **NUCLEAR PLANT WNP-2, OPERATING LICENSE NPF-21 LICENSEE EVENT REPORT NO. 93-009-01**

Transmitted herewith is Licensee Event Report No. 93-009-01 for the WNP-2 Plant. This report is submitted in response to the report requirements of 10CFR50.73 and discusses the items of reportability, corrective action taken, and action taken to preclude recurrence.

Sincerely,

J. V. Parrish (Mail Drop 1023)
Assistant Managing Director, Operations

JVP/DAS/jd
Enclosure

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LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Washington Nuclear Plant - Unit 2	0 5 0 0 0 3 9 7	1 OF 15

TITLE (4)
EXISTENCE OF NONCONDENSIBLE GASES IN THE REFERENCE LEG OF REACTOR PRESSURE VESSEL INSTRUMENTATION

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAMES	DOCKET NUMBERS(S)
										0 5 0 0 0
0 2	1 7	9 3	9 3	0 0 9	0 1	0 5	1 7	9 3		0 5 0 0 0

OPERATING MODE (9)	1	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)			
POWER LEVEL (10)	0 9 6	[ ] 20.402(b)	[ ] 20.405(C)	[ ] 50.73(a)(2)(iv)	[ ] 77.71(b)
		[ ] 20.405(a)(1)(i)	[ ] 50.36(c)(1)	[X] 50.73(a)(2)(v)	[ ] 73.73(c)
		[ ] 20.405(a)(1)(ii)	[ ] 50.36(c)(2)	[ ] 50.73(a)(2)(vii)	[ ] OTHER (Specify in Abstract below and in Text, NRC Form 366A)
		[ ] 20.405(a)(1)(iii)	[ ] 50.73(a)(2)(i)	[ ] 50.73(a)(2)(viii)(A)	
		[ ] 20.405(a)(1)(iv)	[ ] 50.73(a)(2)(ii)	[ ] 50.73(a)(2)(viii)(B)	
		[ ] 20.405(a)(1)(v)	[ ] 50.73(a)(2)(iii)	[ ] 50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER
D. A. Swank, Compliance Engineer	AREA CODE 5 0 9 3 7 7 - 4 1 4 7

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED (14)		EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
[ ] YES (If yes, complete EXPECTED SUBMISSION DATE)	[X] NO				

ABSTRACT (16)

On February 17, 1993, after a period of engineering review and evaluation, plant management concluded that the existence of noncondensible gases in the reference leg of Reactor Pressure Vessel (RPV) Instrumentation could have affected the point at which a RPV Level 3 isolation occurred. This actuation is required in response to a postulated RHR pipe crack and would have delayed a Nuclear Steam Supply Shutoff System (NS^4) Groups 5 and 6 isolation.

As an immediate corrective action credit was taken for an existing flood watch to assure a leak would be detected and any flood could be terminated. Further corrective actions include procedure changes and training to provide plant operators with information on how to recognize the noncondensible gas problem and to describe actions to be taken if the conditions are observed. Boiling Water Reactor Owners Group (BWROG) activities associated with this problem are being tracked to provide a short and long-term solution to this issue.

Testing completed by the Electric Power Research Institute (EPRI) for the BWROG in April 1993, indicates that the impact of noncondensible gases could, under postulated accident conditions, effect the RPV Level 3 isolation actuation setpoints to a greater extent than previously known. Observations during the WNP-2 shutdown for refueling on May 1, 1993, did not reveal significant degassing. Observed level indication deviations on May 1, 1993, did not meet the unacceptable notching criteria of six inches offset for two minutes or longer duration.

The root cause of this event was a design deficiency. The RPV level instrumentation designer did not recognize the impact of noncondensible gases in the system.

The event posed no threat to the health and safety of either the public or plant personnel.

Plant Conditions

Power Level - 95%
Plant Mode - 1

Event Description

On February 17, 1993, after a period of engineering review and evaluation, plant management concluded that the existence of noncondensible gases in the reference leg of Reactor Pressure Vessel (RPV) Instrumentation could have affected the point at which the RPV Level 3 isolation actuated. This trip is used to mitigate a postulated moderate energy line crack during shutdown cooling. During shutdown cooling a RPV Level 3 trip results in a Nuclear Steam Supply Shutoff System (NS^4) Groups 5 and 6 isolation. This isolation closes the Residual Heat Removal (RHR) Shutdown Cooling Outboard and Inboard Supply Valves (RHR-V-8 and RHR-V-9) plus other RHR system valves (please see the attached figure). The purpose of this isolation function is to prevent draining the RPV water to areas outside containment in the unlikely event of a pipe crack in the RHR piping. Four Level Indicating Switches (MS-LIS-24A, B, C, and D) are used to detect a loss of reactor water level during this shutdown cooling condition. Each of these instruments is associated with a separate condensing chamber (MS-CU-4A, D, C, and B, respectively) and reference leg instrument tubing. The logic associated with Group 5 and 6 isolation is a two-out-of-two arrangement. One trip system (Division) of the logic operates the outboard valves and the other the inboard valves. A RPV low water level condition detected by MS-LIS-24A (associated with MS-CU-4A) and MS-LIS-24B (associated with MS-CU-4D) would result in an outboard isolation (closure of RHR-V-8, plus other valves). Likewise, a RPV low water level condition detected by MS-LIS-24C (associated with MS-CU-4C) and MS-LIS-24D (associated with MS-CU-4B) would result in an inboard isolation (closure of RHR-V-9 plus other valves).

The relationship between condensing chambers, level indicating switches, and Narrow Range level transmitters used for input to the Transient Data Acquisition System (TDAS) computer and the plant computer is provided below.

Condensing Chamber #	Indicating Switch #	Level Transmitter #	TDAS Point	Plant Computer
MS-CU-4A	MS-LIS-24A	RFW-DPT-4A	A or B	Yes
MS-CU-4B	MS-LIS-24D	RFW-DPT-4B	A or B	Yes
MS-CU-4C	MS-LIS-24C	RFW-DPT-4C	N/A	Yes
MS-CU-4D	MS-LIS-24B	N/A	N/A	N/A

Previously, on January 21, 1993, during depressurization after 140 days at power, RPV level notching and degassing were observed as part of preplanned data gathering for the Boiling Water Reactor Owners Group (BWROG). The notching and degassing are both believed to have been initiated by the build up of noncondensible gases in the level instrumentation reference legs during the approximately 140 day run prior to the scram. During operation, these gases can move into the reference legs by several methods. These methods include, 1) leakage at the instrument racks, which allows the water with the dissolved gases to migrate down the reference legs from the condensing chambers, 2) thermal mixing within the reference leg, 3) gas diffusion down the reference legs and 4) gases introduced by previous surveillance test activities when test equipment is connected to the instrument lines.

The term notching is used to describe a step increase in level indication that lasts for a noticeable period of time, followed by the indicated level returning to the previous values. It is believed to be a result of noncondensible gases, which accumulate in the reference legs during plant operation, coming out of solution as pressure decreases and forming relatively large bubbles. These bubbles eventually propagate to the vertical sections of the reference legs, resulting in temporary water head perturbations on the associated reference leg. This notching is generally repetitive and may reflect the instrument piping geometry. An unacceptable notch is defined as a level deviation of six inches or larger for a duration of two minutes or longer.

The term degassing is used to describe erratic noise like pulses that can result in an increased level biased error. Like notching, degassing is a result of noncondensible gases coming out of solution as pressure decreases and forming many small bubbles that propagate to vertical sections of piping, resulting in momentary water head perturbations. The quantity of noncondensible gasses released during depressurization by this mechanism, possibly in combination with the notching phenomenon, can displace water from the reference leg as the gases expand. This reference leg water head decrease results in a higher than actual reactor water level indication. At the relatively low reactor pressure and temperature where this phenomenon occurs, it can take a substantial period of time (two or more hours) for the reference leg to refill through condensation within the condensing chamber.

Following the reactor scram on January 21, 1993, notching on Narrow Range Channels B and C (associated with condensing chambers MS-CU-4B and MS-CU-4C respectively) was observed by high resolution recording provided by the computer system. This data was being collected to support BWROG activities associated with this subject. No notching was detected on Narrow Range Channel A and MS-LIS-24B was not monitored. There is no Narrow Range Level transmitter associated with condensing chamber MS-CU-4D. The first notch occurred on Narrow Range Channel C about 7.7 hours after the scram when reactor pressure was about 120 psig. The notch gave a false high level indication of about four inches lasting for about one minute. Narrow Range Channel C notching indications reappeared at ten to fifteen minute intervals until they were masked by significant degassing at about 10.7 hours after the scram. Notching appeared on Narrow Range Channel B about 8.6 hours after the scram with pressure about 50 psig. Narrow Range Channel B showed repeated double notches of about seven inches in height with the first notch having a duration of about one minute followed by a second notch of about 0.8 minute duration. The Narrow Range Channel B notching continued throughout the monitored period.

Some degassing was also observed on Narrow Range Channel B starting about eight hours after the scram with pressure at about 50 psig. The degassing for Narrow Range Channel B did not produce an appreciable bias; the indicated increase was less than about two inches. Narrow Range Channel C showed increased degassing of about 10 inches at about nine hours after the scram when pressure was 35 psig. At about 10.7 hours after the scram, coincident with the initiation of shutdown cooling and a pressure drop from 20 to 10 psig, significant degassing occurred on Narrow Range Channel C. The degassing resulted in a peak level offset of about 32 inches within four minutes of the pressure reduction. Within about 2 hours Narrow Range Channel C fully recovered to the expected level; probably in response to the condensing chamber refilling the reference leg.

This observed behavior of the RPV level instrumentation could have delayed the initiation of a Level 3 trip isolation in the event of a moderate energy pipe crack in the RHR piping during the shutdown cooling mode of operation. The inboard isolation function is initiated only if level indicating switches MS-LIS-24C and MS-LIS-24D are both actuated. This trip could have been delayed until the level was 32 inches below Level 3 based on the observed degassing behavior of Narrow Range Channel C. The response of the redundant outboard logic as a result of degassing was indeterminate. No high resolution recording data is available for the instrumentation associated with condensing Chamber MS-CU-4D and associated level indicating switch MS-LIS-24B. No degassing was observed associated with MS-CU-4A and its connected MS-LIS-24A. However, the two-out-of-two logic requires both Narrow Range Channels to trip for isolation.

This condition could have impacted the flooding analysis and the capability to remove decay heat from the reactor.

Mock up testing completed by EPRI for the BWROG in April 1993, indicates that the impact of high concentrations of noncondensible gases could, under postulated accident conditions, effect the RPV Level 3 isolation actuation setpoints to a greater extent than previously known. The test results, including those for a mock up of the WNP-2 Narrow Range Channel C level transmitter instrument piping configuration, were provided to the Staff for their review provided as proprietary information by the BWROG on May 5, 1993.

During the scheduled plant shutdown for refueling on May 1, 1993, additional nonlicensed personnel were stationed to monitor the subject instruments. Significant degassing was not observed. Level indication changes of less than six inches for durations of less than one minute were observed on Narrow Range Channel C. Narrow Range Channel C disturbances of one inch or more were recorded at the listed reactor pressures: 1" for less than 10 seconds at 47 psig; 2) 6.5 inches for 55 seconds at 12.5 psig; 3) 6 inches for 55 seconds at 7.5 psig; 4) 5.5 inches for 55 seconds at 6 psig; 5) 5.5 inches for 55 seconds at 3.5 psig; and 6) 2 inches for 2 minutes at 3.8 psig. No significant level indication disturbances were observed on Narrow Range Channels A and B or the upset range indicators. The reduced magnitude of reactor level indication changes observed during this shutdown may be the result of the limited time at power, 70 days, during which noncondensible gases could accumulate in the instrument reference legs.

Additionally, unlike during the January depressurization, MS-LIS-24B was observed locally during the May shutdown. As predicted based on the instrumentation reference leg configurations, MS-LIS-24B level indication disturbances were observed. Approximately 13 level indication disturbances, starting at 42 psig RPV pressure, were observed. These disturbances were less than six inches in magnitude and less than ten seconds in duration.

Immediate Corrective Action

Credit was taken for an existing flood watch using surveillance cameras or an hourly flood tour to assure a leak is detected and any flood could be terminated by operator action. These compensatory measures were initiated as compensatory action for problems identified with the ECCS pump room penetration seals as explained in LER 92-034.

Further Evaluation, Root Cause and Corrective Action

A. Further Evaluation

1. This event is being reported per the requirements of 10CFR 50.73(a)(2)(v) as, "....an event or condition that alone could have prevented the fulfillment of the safety function of....systems that are needed to....remove residual heat." A four hour nonemergency report in accordance with 10CFR 50.72(b)(2)(iii) was made at 1656 hours on February 17, 1993. An update of the four hour nonemergency report was made at 0830 on April 16, 1993, based on the EPRI test data for the WNP-2 configuration.

2. A further evaluation was performed for each of the instruments attached to the condensing chambers. The instrumentation can be divided into three functional categories: those that provide signals for automatic actuations or process variable information for plant operator use at pressures above 450 psig; those that provide process variable information for plant operator use at pressures below 450 psig; and those that provide signals for automatic actuations at pressures below 450 psig. The following is a discussion of each category:

   a. Functions that use instrumentation to provide signals for automatic actuation or process variable information for operator action at pressures above 450 psig include:

      (1) High Pressure Core Spray (HPCS) high level trip,

      (2) RPV low water level scram,

      (3) Reactor Core Isolation Cooling (RCIC) high level trip,

      (4) HPCS RPV low water level initiation,

      (5) Containment isolation initiation,

      (6) Post accident RPV level and pressure monitoring,

      (7) Main Steam Isolation Valve (MSIV) high pressure scram interlock,

      (8) RPV high pressure scram,

      (9) Reactor Feedwater RPV high water level trip,

      (10) RPV low water level ATWS initiation,

      (11) ECCS RPV low water level initiation,

      (12) Automatic Depressurization System (ADS) initiation on low RPV level,

      (13) Safety Relief Valve (SRV) actuation,

      (14) ECCS injection valve interlock on low RPV pressure, and

      (15) ATWS initiation on RPV high pressure.

Since these functions all occur at high pressure the evaluation concluded there would be no impact from notching or degassing.

b. Functions that use instrumentation to provide information for plant operator use at pressures below 450 psig include:

(1) Post accident water level monitoring (main control room and remote shutdown panels),

(2) Post accident RPV pressure monitoring (main control room and remote shutdown panels), and

(3) Post accident fuel zone, shutdown range, and upset range RPV water level monitoring.

For these functions plant operators have been provided with guidance in determining RPV vessel water level following depressurization when degassing may have resulted in inaccurate water level displays.

In addition, in this category the Main Steam Isolation Valve Leakage Control System (MSLC) function provides a permissive for Operator initiated MSLC following a LOCA at a pressure which will protect the low pressure piping. This variable actuates at 61 psig which is within the range at which WNP-2 and other plants have experienced the effects of instrument line degassing. The worst case result of degassing is a premature initiation of MSLC at approximately 79 psig. This has no safety significance since WNP-2 has preliminary calculations which indicate that the MSLC piping will not fail from an inadvertent opening of the system inlet valves with the MSIVs closed and leaking at maximum Technical Specification rates.

c. There is one function that uses instrumentation signals for automatic actuation at RPV pressure below 450 psig; RHR Isolation during shutdown cooling.

The RHR shutdown cooling isolation valve control logic has a reactor pressure interlock to prevent opening these valves during high reactor pressure conditions and exposing the low pressure piping to high pressure. For moderate energy pipe crack events (RHR piping), during shutdown cooling operation when the reactor is depressurized, these valves are automatically closed by reactor low water level signals (Level 3 trip). Under Mode 3 conditions, the RPV Level 3 trip instrumentation could experience degassing and not be available to actuate RHR valve isolation at the correct setpoint. The Completed Corrective Actions below describe how the plant will be operated within the bounds of the Technical Specification under these conditions.

Further evaluation has been completed assuming that the Reactor Vessel Level 3 trip signals are not available. In this case, the crack will continue to flood the RHR pump room without automatic isolation (refer to FSAR Question Response 211.031). This condition will continue until the pump room flood monitors actuate and alert plant Operators of flood conditions. The accident analysis assumes that Operator action can occur from the Control Room 20 minutes after the condition is known. This event will result in loss of approximately 63 inches of reactor water level with loss being terminated by Operator action at a vessel level of approximately -22 inches; this is 138 inches above the Top Of Active Fuel.

The above shutdown cooling scenario assumes that the pump room flood monitor actuates. Since these monitors are not single failure proof, it is necessary to postulate loss of this signal. Flooding can be mitigated by either the hourly tour or surveillance cameras. This will result in a scenario where operators will begin RPV makeup if water level drops within the EOP control band, thus preventing core uncovery. The crack will continue to flood the pump room until it is identified by the tour (the most conservative option) and operator action is taken to isolate the crack. The flood tour or camera monitoring will continue when in the shutdown cooling mode of RHR operation with the plant in Hot Shutdown and any of the four MS-LIS-24A/B/C/D instruments inoperable. During shutdown cooling, the flood tour or continuous monitoring need only be continued until the four narrow range instruments are restored to an operable status, or until the plant is in Cold Shutdown and the instruments are not required to be operable.

Automatic isolation of the shutdown cooling mode of RHR operation was also evaluated for Hot Shutdown conditions. This condition was evaluated for the unlikely event that multiple RPV Level switches exhibit notching and are placed in the tripped condition, resulting in a shutdown cooling isolation. This isolation is the design basis safety condition for a small crack in the RHR shutdown cooling line. The shutdown cooling mode of RHR operation is not credited for any of the design basis accidents, nor is it required to achieve or maintain Cold Shutdown conditions.

The shutdown cooling mode of operation is prevented above 135 psig by hardware interlocks which preclude opening the shutdown cooling suction valves. Administrative requirements, implemented through procedural controls, prevent operation of RHR in the shutdown cooling mode above 48 psig reactor pressure. Finally, procedural guidance recommends that a reactor pressure of less than 20 psig be reached prior to initiation of shutdown cooling. Therefore, the region of concern for automatic isolation of shutdown cooling is 20 psig ($\approx$ 255°F), and no greater than 48 psig ($\approx$280°F), to 0 psig reactor pressure. WNP-2 has not recorded notching or significant degassing above 135 psig.

The preferred response to a shutdown cooling isolation is to restore shutdown cooling by clearing the cause of the isolation. Returning the level indicating switches to an operable status will be accomplished as described in the Completed Corrective Actions section. If shutdown cooling cannot be restored, either the Condensate system if the condenser is available, or the Reactor Core Isolation Cooling (RCIC) system if the condenser is not available, can be utilized. RCIC use requires a reactor pressure of 75 psig, which will be achieved by reactor heatup to $\approx$320°F.

Isolation of shutdown cooling in Hot Shutdown can be divided into two regions, between 200°F and about 212°F, and above 212°F. Below 212°F, reactor heatup will raise temperature until boiling occurs. Then, reactor pressure and temperature will rise in a saturated state. Both the Condensate and RCIC systems are capable of providing the necessary core cooling at the pressures and temperatures where notching has been detected at WNP-2, 120 psig and below. The Condensate system can be used up to 330 psig, or higher if Reactor Feedwater system is placed in service. The RCIC system can be used from 75 psig and is capable of performing the decay heat removal function even during a station blackout event.

The design basis response to a loss of shutdown cooling is to use the alternate shutdown cooling mode of RHR. This requires use of one of the RHR pumps to pump water from the suppression pool to the reactor vessel. Then from the Control Room either steam or, if necessary, water is relieved through the main steam relief valves back to the suppression pool. Ultimate heat sink capability is provided by the safety-related Standby Service Water System. Although this method uses safety-related equipment, it is not the preferred response method.

Backfilling the instrument reference legs at a pressure higher than 135 psig was evaluated. Backfilling at a higher reactor pressure would reduce the probability of notching occurring. Narrow range reactor level is measured using differential pressure (level indicating switches) instrumentation. There are individual reference legs for channel MS-LIS-24 A, B, C and D instruments, but the variable legs of the narrow range instruments share a common line. A and B share one sensing line, C and D share another. Any perturbation on the reference leg can be transmitted through the differential instrument to the variable leg, causing a transient on the opposite channel instrument. These level indicating switches also input trip signals to the Reactor Protection System (RPS). Any effort to backfill requires isolation of the instruments associated with the reference leg. This requires placing the logic associated with these instruments in a tripped condition. Although this is the safe condition, a single spurious signal could then cause an automatic actuation of a safety system. This spurious trip of an instrument would not be expected to come from a notch since a notch results in an indicated level which is higher than the actual level, and the isolation occurs on low level.

Backfilling requires the introduction of relative cold water ($\approx 65°F$) through the reference leg piping and through the instrument line nozzles at the reactor vessel. These nozzles are designed for a limited number of thermal cycles. The number of available cycles is a function of the temperature gradient experienced. The temperature gradient at 20 psig is $\approx 190°F$, the gradient at 1000 psig is $\approx 480°F$, and the gradient at 135 psig is $\approx 295°F$. Based on the requirement to isolate instrumentation, the potential for spurious actuation of instrumentation of both Divisions, and the thermal cycle experienced, it was determined that backfilling of the reference legs will not be performed unless necessary to return the instrumentation to operability, and then at the lowest possible reactor pressure and temperature.

3. There were no structures, components or systems that were inoperable prior to the start of this event which contributed to the event.

B. Root Cause

The root cause of this event was a design deficiency. The RPV instrumentation designer did not recognize the impact of noncondensible gases in the system. Very low leakage criteria were not a requirement of the original design.

C. Completed Corrective Action

The following corrective actions have been completed in response to this issue.

1. Criteria have been developed for operability of RPV Water Level Isolation Logic for RHR Isolation.

   a. The Narrow Range Level Channels (A through D) are operable in Modes 1 and 2 as the phenomenon observed can only exist during depressurization in Mode 3, Hot Shutdown.

   b. For all reactor depressurizations to below 450 psig, other than during emergency reactor depressurization or immediate depressurization to protect the health and safety of the public or plant equipment, personnel will be stationed to observe the indicated reactor level on Narrow Range Channels A, B, and C, and locally at MS-LIS-24B. Notches, or degassing, that exceeds six inches in height for a duration of two minutes or longer are unacceptable. The trip channel associated with that signal will be declared inoperable for Mode 3 operation and Technical Specification (3/4 3.3.2) requirements complied with.

   c. If a trip channel has been declared inoperable, the function of that channel may be recovered and the channel returned to operable status. Recovery of the channel requires the indication to return to an indicated level value that is consistent with the other operable water level channels and maintain that recovered water level indication for at least five minutes. This can be accomplished either through normal refill of the reference leg from the condensing chamber, or by plant personnel backfilling the reference leg using approved procedures.

2. Under the guidance provided above, MS-LIS-24A/B/C/D are operable in Mode 3. This position is based upon the recognition that the notching observed in these channels is well defined, of short duration, produces a small error in RPV level indication and is much different from what would be expected for actual loss of RPV inventory that would put the plant at risk. This position is also based on the monitoring actions described in this LER which ensure that should unacceptable notching occur, the channel will be declared inoperable and will not be declared operable again until evidence exists, as described in 1.c above, to show the channel is operable.

3. RPV water level instrumentation will be closely monitored during depressurization when RPV pressure is below 450 psig until Cold Shutdown is reached. This observation will be enhanced by using a real time plot of available Transient Data Acquisition System (TDAS) computer and/or plant computer channels, when available, and local observation of MS-LIS-24B associated with condensing chamber MS-CU-4D. In addition, the plant process computer will be configured to display available RPV Narrow Range data. In Mode 3, if reference leg water level notching or significant degassing is observed and the trip channel is declared inoperable per the criteria in Paragraph 1 above, the Technical Specification Action Statement for MS-LIS-24A/B/C/D (3/4 3.3.2) will be entered.

4. In Mode 3, if entry into the shutdown cooling mode has occurred and significant degassing or notching is observed, the plant will either continue to Cold Shutdown within the applicable Technical Specification actions or will remain in Hot Shutdown and restore the inoperable channel to an operable condition per the requirements of Paragraph 1 above, prior to continuing plant shutdown to Cold Shutdown.

5. If any channel(s) is inoperable, the existing hourly flood tour (or continuous camera surveillance) will continue in the associated RHR pump room whenever RHR is in the shutdown cooling mode and the plant is in Mode 3, and will remain in effect until the plant is in Mode 4 or the channel is returned to an operable status.

6. Plant Procedures PPM 3.2.1, Normal Shutdown to Cold Shutdown, PPM 3.2.2, Normal Shutdown to Hot Shutdown, and PPM 3.3.1, Reactor Scram have been revised to accomplish the following: a) provide the operators instructions for enhanced monitoring of the narrow range RPV data to facilitate the identification of notching and degassing; b) provide the operators with guidance on how to recognize these conditions; and c) describe the actions to take if they are observed.

7. Operator training on the recognition of degassing and notching and on the determination of RPV water level during depressurization has been completed.

8. A plant walkdown has been completed of instruments that are connected to RPV level condensing chamber reference legs A, B, C, and D to visually identify leaks. Leakage was measured over a four day period ending on March 1, 1993. Evidence of past leakage was present on valves and fittings associated with all condensing chambers. At the time of the inspection no visible leakage was observed on equipment associated with MS-CU-4A. Those areas that had the potential for significant leakage were bagged in an effort to quantify the leak. Measured leakage from equipment associated with MS-CU-4B was 41 milliliters over 98 hours. Leakage from devices associated with MS-CU-4C was 10 milliliters over 117 hours and leakage from MS-CU-4D was 15 milliliters over 99 hours.

9. Procedures have been developed to support backfill of the instrument reference legs in Modes 3 and 4.

10. An evaluation was performed to identify maintenance activities which can be implemented in the near term to minimize the RPV level indication inaccuracies. Maintenance Work Requests have been written to reduce leakage from these instruments and the associated piping. This work will be completed during the current refueling outage eight.

The actions described above will be performed for each reactor depressurization below 450 psig until plant modifications are made.

D. Further Corrective Action

1. A walkdown will be performed to identify areas in the reference leg piping where line slope could cause noncondensible gases to accumulate. This will be completed by the end of refuel outage number eight (approximately June 15, 1993).

2. The Supply System will continue to participate in BWROG activities associated with this subject. Recommendations made by the BWROG are expected to provide both short term maintenance and long-term resolution of this issue.

Safety Significance

Previous plant operation without knowledge of the impact of noncondensible gasses in the instrument lines could have resulted in nonconservative biased indication of the instrumentation. However, information was available to assist plant operators in responding to plant emergencies with inaccurate instrumentation. With regard to the shutdown cooling isolation, the existence of the safety related room flood monitor provides a backup to the Level 3 trip. Even with the failure of this second device the analysis shows the plant can be safely shutdown. Therefore, the Supply System concludes the specific event described in this LER involving a Level 3 trip is not safety significant. However, the general issue of noncondensible gases in the reference leg of RPV instrumentation is safety significant.

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION			
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TITLE (4) EXISTENCE OF NONCONDENSIBLE GASES IN THE REFERENCE LEG OF REACTOR PRESSURE VESSEL INSTRUMENTATION

EIIS Information

Text Reference	EIIS Reference System	EIIS Reference Component
Reactor Pressure Vessel (RPV)	--	VSL
Nuclear Steam Supply Shutoff System (NS[4])	BD	--
Residual Heat Removal System (RHR)	BO	--
RHR Shutdown Cooling Supply Valves (RHR-V-8 & RHR-V-9)	BO	V
Level Indicating Switches (MS-LIS-24A, B, C and D)	SB	LIS
Condensing Chambers (MS-CU-4A, B, C and D)	SB	COND
High Pressure Core Spray System (HPCS)	BG	--
Reactor Core Isolation Cooling System (RCIC)	BN	--
Safety Relief Valves (SRV)	SB	V
Main Steam Leakage Control (MSCC)	SB	--
Transient Data Acquisition System (TDAS)		--
Reactor Feedwater (RFW)	SJ	--
Condensate	SD	--
Reactor Protection System	JC	--

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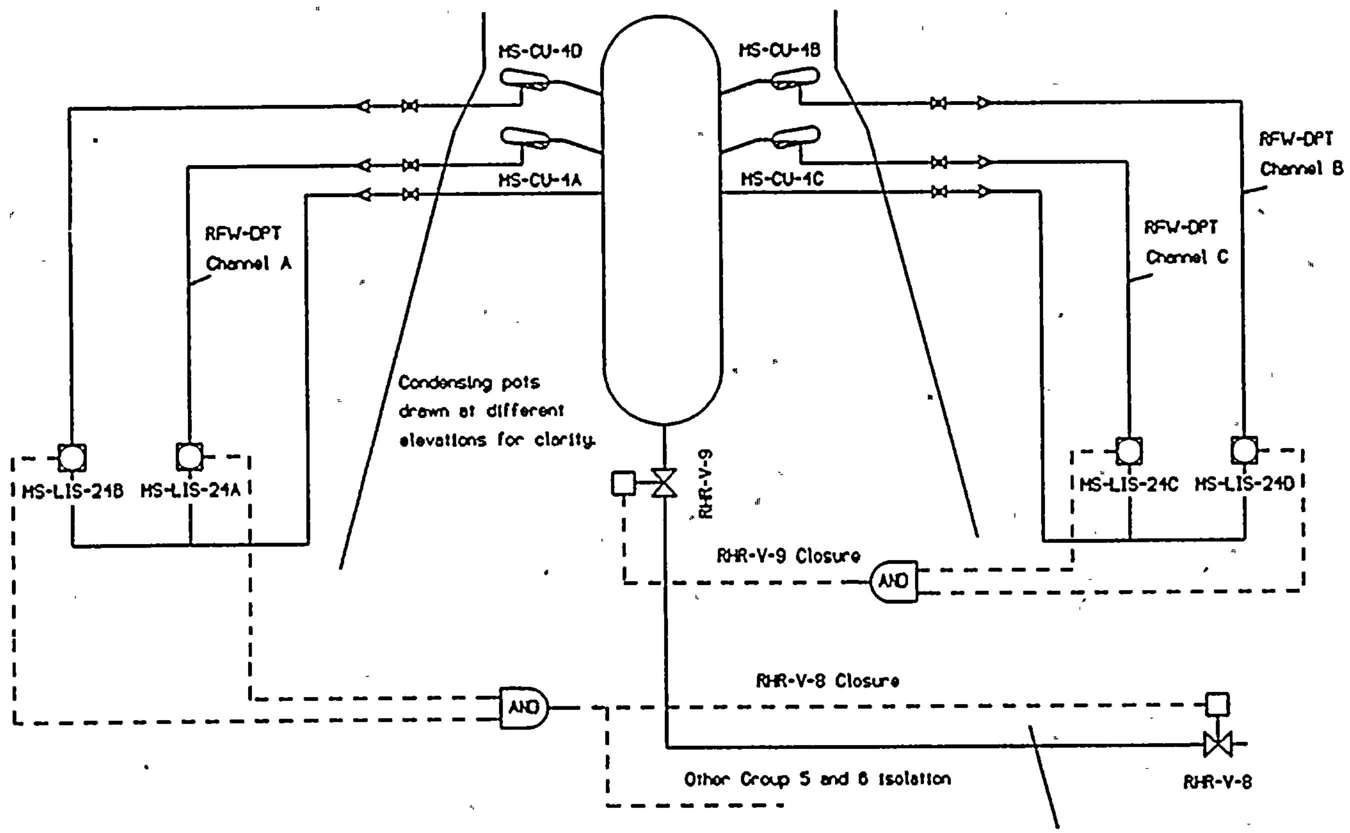
TITLE (4)
EXISTENCE OF NONCONDENSIBLE GASES IN THE REFERENCE LEG OF REACTOR PRESSURE VESSEL INSTRUMENTATION



RPV Narrow Range Level