UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON, DC 20555-0001

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# NRC INFORMATION NOTICE 2002-10, SUPPLEMENT 1: DIABLO CANYON MANUAL REACTOR TRIP AND STEAM GENERATOR WATER LEVEL SETPOINT UNCERTAINTIES

## ADDRESSEES

All holders of operating licenses for nuclear power reactors, except those who have permanently ceased operations and have certified that fuel has been permanently removed from the reactor.

## PURPOSE

The U.S. Nuclear Regulatory Commission (NRC) is issuing this supplement to give addressees further information about the manual reactor trip of Diablo Canyon Unit 2 which resulted from a failure of the main feedwater regulating valve, non-conservative steam generator setpoints and contributing causes, and other licensee actions relating to these events. This supplement provides information that became available after the issuance of the original information notice (IN). The NRC expects that recipients will review the information for applicability to their facilities and consider taking actions, as appropriate. However, this supplement does not contain any NRC requirements and does not require any specific action or written response.

## BACKGROUND

Diablo Canyon Nuclear Power Plant reported a manual reactor trip of Unit 2 which resulted from a loss of main feedwater to a steam generator (LER 2-2002-002-00) and that the narrow-range steam generator water level instrumentation did not respond as expected to initiate an automatic reactor trip and emergency feedwater actuation on low-low water level in the steam generator (LER 1-2002-001-00). The NRC issued IN 2002-10 on March 7, 2002, to inform licensees of this event. Following the issuance of the original IN, the NRC staff conducted a Special Inspection at Diablo Canyon, as well as a public meeting with Westinghouse. In addition, Diablo Canyon has completed its Licensee Event Reports (LERs), Westinghouse has issued five Nuclear Safety Advisory Letters (NSALs) relating to this phenomenon or the presence of the void content of the two phase mixture above the mid-deck plate, and other facilities have generated reports under Title 10, Section 50.72, of the *Code of Federal Regulations* (10 CFR 50.72).

## DESCRIPTION OF CIRCUMSTANCES

On February 9, 2002, with Unit 2 at full power, main feedwater regulating valve (MFRV) FW-2-FCV-540 failed in the closed position, resulting in a rapid decrease in the water level of Steam Generator 2-4. The indicated narrow-range water level decreased to 7.5 percent and leveled out. Operators tripped the Unit 2 reactor within approximately 1 minute after the main feedwater regulating valve closed. On February 14, 2002, after the Unit 2 restart but while still investigating the event, the licensee identified a potentially unanalyzed condition involving the narrow-range steam generator water level instrumentation. The licensee determined that during the plant transient, the actual water level in steam generator 2-4 fell below the 7.2-percent narrow-range trip setpoint for engineered safety feature and reactor trip actuations before operators manually tripped the reactor. The steam generator vendor attributed this water-level discrepancy to a previously unaccounted for differential pressure created by steam flow past the mid-deck plate in the moisture separator section of the steam generator. This differential pressure phenomenon caused the steam generator narrow-range instruments to indicate a higher-than-actual water level. Thus, the steam generator narrow-range water level low-low setpoint was non-conservative during the loss of normal feedwater transient.

**Physical Phenomenon and System Description**

Steam generators designed by Westinghouse incorporate two-stage moisture separation. The first stage uses centrifugal separators, and the second stage uses chevron-type separators. A mid-deck divider plate separates the two stages. The steam generator water level instrumentation uses differential pressure instruments with two ranges, a wide-range non-safety-related instrument and three or four (depending on plant) narrow-range safety-related instruments. The wide-range instrument spans essentially the entire length of the downcomer region, while the narrow-range instruments span only the upper 25 percent of the wide-range to cover the normal operating band. The upper taps for all four instruments are located above the mid-deck plate, while the lower taps are all located below this plate.

In the event at Diablo Canyon, the holes in the mid-deck, which were designed to allow moisture removed from the second-stage separators to flow back into the downcomers, acted as orifices which restricted steam flow and allowed pressure differences with water levels below the mid-deck region. At higher steam flow rates with a decreasing steam generator water level, steam exiting the first stage separators along with the moisture being separated was enough to build up pressure below the plate that was not acting above the plate. Since the upper steam generator water level instrument taps were connected above the plate, a pressure difference acted on the four instruments and provided a bias that caused the instruments to indicate a higher-than-actual level. For the limiting safety setting of the low-low steam generator water level setpoint, this bias acts in the non-conservative direction. The magnitude of the bias drops as the steam flow decreases.

**Post Trip Analysis**

Following this event, the NRC completed an onsite special team inspection at Diablo Canyon Nuclear Power Plant. The inspection examined the events surrounding the Unit 2 reactor trip on February 9, 2002, as they relate to safety and compliance with the Commission’s rules and regulations and the conditions of the Diablo Canyon license. The inspection consisted of examining procedures and records, and interviewing station personnel and staff members, as well as the reactor plant contractor. The NRC’s Special Inspection Team also developed a detailed sequence of events and organizational response time line which is summarized in the “Overview and Sequence of Events” Attachment 2 to this IN.

This event provided an unusual challenge to the licensee in that it involved an unrecognized phenomenon. Many plant events involve equipment behaving in an unexpected manner, but the failure mechanisms are usually well-understood. However, a well-structured corrective action process should still be effective under these circumstances by being sufficiently rigorous to recognize conditions that are adverse to quality and then treating them according to their safety significance. From a review of the post trip review, the NRC's Special Inspection Team concluded that the licensee's process was narrowly focused on finding, understanding, and correcting the cause of the trip. While the station's post-trip analysis procedure contained steps to review plant behavior before, during, and after the event, this was effectively not performed. The cause of the event was readily apparent without the need to analyze plant parameters. However, by not performing a methodical review of the plant's behavior and comparing it to the behavior expected under those conditions, the licensee failed to recognize that an automatic plant trip and ESF actuation of auxiliary feedwater did not occur when required. Had this been recognized, the licensee would probably have delayed restarting the plant until after the cause and implications were understood.

The licensee's review of the anomalous steam generator water level attempted to explain why wide-range indication did not track with narrow-range indication, which was thought to have indicated accurately. The NRC's Special Inspection Team concluded that the licensee's theory and supporting data were not compared with other available but conflicting indications. The licensee calculated that the event would have resulted in loss of approximately 75 percent of the initial water mass in the affected steam generator, and should have caused the wide-range level to be 20 percent of the actual level. The licensee did not note that the bottom of the narrow-range corresponds to approximately 75 percent wide-range level, so the narrow-range instruments should have been expected to be reading off scale low. Also, when auxiliary feedwater actuated, narrow-range level instruments did not show increasing level until after some delay, confirming that actual level was well below the narrow-range.

In addressing the wide-range instrument question, it was clear that the licensee was not fully satisfied that the issue was well-understood. However, rather than clarify the issue immediately, the licensee used a station administrative process that required resolution of the issue within 30 days, and declared the problem to be an issue needing validation to determine impact on operability. The NRC's Special Inspection Team concluded that this process was not integrated with the station's operability determination process, and could permit an issue that was thought to relate to an operability question to be studied for 30 days before addressing the operability question. Although this issue was resolved in 4 days, this approach was considered to be contrary to Generic Letter 91-18, which provided guidance on the need to perform prompt operability assessments.

The following paragraphs present examples of corrective actions from other licensees:

**Callaway**

Callaway reported (EN 38740) that based on an assessment of its steam generator narrow-range low-low level trip setpoints, the existing low-low level trip at 14.8 percent did not account for the uncertainties associated with the differential pressure created by the steam flow past the mid-deck plate in the moisture separator section of the steam generator. A plant power reduction was commenced at 4:58 p.m. on February 28, 2002, to decrease reactor power to below 30 percent where engineering calculations indicate that the steam generator mid-deck plate differential pressure condition will no longer result in a non-conservative setpoint.

**Salem**

Similar to Callaway, the low-low level trip at 9 percent did not account for the uncertainties and Salem reduced power to 38 percent.

**Sequoyah**

Sequoyah personnel also performed an assessment and determined that the existing 10.7 percent low-low level trip setpoint did not account for the uncertainties associated with the differential pressure created by the steam flow past the mid-deck plate in the moisture separator section of the steam generator. As a conservative measure, after Westinghouse identified this issue via NSAL 02-3, Sequoyah actuated the environmental allowance monitor (EAM) on Friday, February 15, 2002. This feature changed the steam generator low-low level reactor trip setpoint from 10.7 percent to 15 percent. Since the known steam generator channel uncertainty was 8.3 percent with a narrow-range span uncertainty of 5.3 percent, Sequoyah determined that operating with the EAMs continuously actuated would allow continued operation.

Callaway, Salem, and Sequoyah have since recalibrated the low-low water level setpoints for the steam generator with the additional margin to account for this newly identified error. The licensees completed this instrument recalibration before increasing the plants' power level to full reactor power.

**Conclusion**

The event described in this IN highlights the potential impact of steam generator water level setpoint errors. These errors could delay the expected automatic reactor trip and emergency feedwater actuation. The IN identifies additional accident analyses and systems associated with the mid-deck plate phenomenon and highlights the importance of a thorough post-trip analysis prior to restart. The IN also provides some of the corrective actions taken because of this event and provides information sources for further investigation.

This information notice does not require any specific action or written response. If you have any questions about the information in this notice, please contact any of the technical contacts listed below or the appropriate project manager from the NRC's Office of Nuclear Reactor Regulation (NRR).

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Attachments:
1. List of References
2. Overview and Sequence of Events
3. List of Recently Issued NRC Information Notices

**REFERENCES**

LER 1-2002-001-00, “Technical Specification Violation Due to Non-Conservative Steam Generator Narrow-Range Water Level Instrumentation,” Diablo Canyon Nuclear Power Plant, April 15, 2002.

LER 2-2002-002-00, “Unit 2 Manual Reactor Trip Due to Loss of Main Feedwater to a Steam Generator,” Diablo Canyon Nuclear Power Plant, April 10, 2002.

NRC Special Team Inspection Reports 50-275/02-07 and 50-323/02-07 for Diablo Canyon Nuclear Power Plant, April 8, 2002.

NRC Information Notice 2002-10, “Non-Conservative Water Level Setpoints on Steam Generators,” March 7, 2002.

Westinghouse Nuclear Safety Advisory Letter NSAL-02-3, “Steam Generator Mid-Deck Plate Pressure Loss Issue,” February 15, 2002.

Westinghouse Nuclear Safety Advisory Letter NSAL-02-3, Rev. 1, “Steam Generator Mid-Deck Plate Pressure Loss Issue,” April 8, 2002.

Westinghouse Nuclear Safety Advisory Letter NSAL-02-4, “Maximum Reliable Indicated Steam Generator Water Level,” February 19, 2002.

Westinghouse Nuclear Safety Advisory Letter NSAL-02-5, “Steam Generator Water Level Control Uncertainty Issue,” February 19, 2002.

Westinghouse Steam Generator Water Level Uncertainty Issues Handout from NRC Public Meeting in Rockville, Maryland, March 20, 2002.

Event Report 38697, “Technical Specification Required Shutdown of Both Units Because Steam Generator Water Level Instrument Channels are Inoperable Due to a Non-Conservative Water Level Low-Low Setpoint,” Diablo Canyon Nuclear Power Plant, February 14, 2002.

Event Report 38713, “Safe Shutdown Capability Impacted by Non-Conservative SG NR Setpoint,” Sequoyah Nuclear Power Plant, February 20, 2002.

Event Report 38740, “Safe Shutdown Capability Impacted by Non-Conservative SG NR Setpoint,” Callaway Nuclear Power Plant, February 28, 2002.

Event Report 38702, “Discovery of a Non-Conservative Low-Low Level Trip Setpoint Due to a Differential Pressure Phenomenon that Causes Steam Generator Narrow-Range Level Channels to Read Higher Than Actual Water Level at High Steam Flows,” Salem, February 15, 2002.

**Overview and Sequence of Events**

This section discusses applicable events and actions before, during, and following the failure of steam generator feedwater regulating valve number 4.

In 1989, Diablo Canyon revised the steam generator narrow-range low-low level trip setpoint in accordance with License Amendment 34 for Unit 1 and Amendment 33 for Unit 2. The nuclear steam supply system (NSSS) vendor (Westinghouse) provided the analysis to reduce the low-low trip setpoint in topical report, WCAP-11784, “Calculation of Steam Generator Level Low and Low-Low Trip Setpoints With Use of a Rosemount 1154 Transmitter.” The licensee reduced the setpoint from 15 to 7.2 percent narrow-range on May 10, 1989, for Unit 1, and on April 12, 1989, for Unit 2. However, the licensee did not factor the mid-deck plate differential pressure into the narrow-range low-low level trip setpoint at this time because the mid-deck phenomenon was unknown.

In 1998, Westinghouse recognized the mid-deck phenomenon while evaluating the design of new (replacement) steam generators using computer modeling tools that were not available during the design review for the original steam generators. Westinghouse began accounting for this bias in the setpoint calculation during design work for replacement steam generators. Westinghouse began assessing the potential impact of the mid-deck plate for original model steam generators in late 1999. Westinghouse was planning to issue Nuclear Safety Advisory Letters about the mid-deck phenomenon at about the time of the Diablo Canyon manual reactor trip.

On February 9, 2002, the Main Feed Regulating Valve (MFRV) FW-2-FCV-540 failed closed, stopping the feedwater flow to steam generator 2-4. After a failed attempt to reopen the MFRV, operators initiated a manual reactor trip. The cause of the MFRV failure was excess current in the coil of an Asco model L206-381-6F solenoid valve (SV), which in turn caused the failure of a power fuse and forced the MFRV to close.

During discussions with the resident inspectors after the event, the operations manager and shift supervision expressed skepticism that the steam generator level dropped as low as observed by the steam generator wide-range instrument during the trip. The shift technical advisor confirmed that the wide-range level indication reached 10 percent. Diablo Canyon’s Engineering Services reported that steam generator structural integrity was not affected by low wide-range level. Engineering Services preliminarily concluded that dynamic processes contributed to inaccurate wide-range level indication. Later that night, during a conference call with the NRC staff to discuss the licensee’s plans for the restart of Unit 2, the licensee reviewed its corrective actions for the feedwater regulating valve and other failed components. The NRC staff expressed concern that wide-range indicated level was abnormally low for this transient. The licensee explained its theory that the actual level was higher because of the difference between the transient conditions (hot, dynamic) and the calibration conditions (cold, static). The licensee believed that the steam generator narrow-range level response was normal, and the wide-range level indication was overly conservative but did not impact operator response to such an indication. The NRC decided to conduct follow up activities on level anomalies. The Plant Staff Review Committee reviewed the results of the trip event response team investigation and readiness for restart. The steam generator wide-range water level anomaly issue was discussed and determined not to be a restart issue. The issue was classified as an issue needing validation to determine impact on operability (INVDIO). The Station Director granted permission to restart the plant.

Unit 2 was restarted the next day (February 10, 2002). The licensee continued its investigation of the Unit 2 steam generator response during the transient. Diablo Canyon personnel sent plant trip data to Westinghouse for review. The licensee began to focus on steam generator narrow-range indication as a potential concern. During a conference call between the licensee and Westinghouse on February 14, 2002, Westinghouse informed the licensee about a new process measurement error term related to mid-deck plate differential pressure that had not been included in the existing setpoint analysis. Operators in both units declared all channels of narrow-range level instrumentation inoperable and entered Technical Specification 3.0.3. Operations issued a shift order to manually trip the reactor on loss of feedwater flow. Operators in both units began reducing power to less than 60-percent thermal power to restore the narrow-range instruments to an operable condition and exit Technical Specification 3.0.3. On the basis of information received from Westinghouse, the licensee promptly completed an operability assessment which concluded that the existing trip setpoint at a 7.2 percent narrow-range level remained operable at or below 60-percent reactor power. Operators stopped power reductions at 60-percent power. This action had to be taken because the failure to correct this condition prior to restart resulted in Unit 2 changing Modes (Mode 3 to 2 to 1) with the reactor trip system and engineered safety system steam generator water level low-low instrumentation inoperable and subsequent operation of Units 1 and 2 (Mode 1) in a condition prohibited by Technical Specification 3.3.1. This Technical Specification requires that the reactor trip system instrumentation for steam generator level low-low be operable for Modes 1 and 2 and the engineered safety feature actuation instrumentation steam generator water level-low-low be operable in Modes 1, 2, and 3.

On February 15, 2002, the licensee implemented setpoint changes on both units to raise the steam generator low-low setpoint to 15 percent. After implementation, operators increased power to 100 percent in both units. Westinghouse issued NSAL 02-3, “Steam Generator Mid-Deck Plate Pressure Loss Issue” on February 19, 2002. That NSAL warned plants with Westinghouse-designed steam generators that the error source has not been accounted for and has potentially adverse effects on steam generator level low-low uncertainty calculations as a bias in the indicated high direction. Westinghouse further warned that for plants for which Westinghouse maintains the calculation of record, this pressure drop effect may require a maximum decrease of approximately 9 percent (in percent narrow-range span) in the safety analysis limit (SAL) for establishing the low-low steam generator water level reactor trip for the loss of normal feedwater (LONF) transient or the loss of offsite power (LOOP) transient to compensate for this bias. NSAL 02-3 added additional transients to consider, such as the steamline break mass and energy release, and for plants with feed line check valves inside containment, the feedline break transient, to compensate for this described bias. Revision 1 to the NSAL 02-3 also provided updated information regarding the steam generator water level mid-deck plate pressure loss issue. Specifically, Westinghouse revised the NSAL to address the impact of this issue on the feedwater line break analysis (when feedwater check valves were located inside containment), the ATWS mitigation system actuation circuitry system, and steamline break mass and energy release calculations. Westinghouse subsequently issued NSAL 02-4, “Maximum Reliable Indicated Steam Generator Water Level,” and NSAL 02-5, “Steam Generator Water Level Control System Uncertainty Issue” on February 19, 2002. Revision 1 to NSAL 02-5 was written to clarify the need to calculate the steam generator water level uncertainties at normal operating level, including the impact, if any, of the mid-deck plate pressure differential and to compare the uncertainties used in the initial condition of the safety analyses to determine if they remain bounding. NSAL 02-5, Rev. 1, also discusses the potential impact on safety analyses performed at reactor power levels other than 100 percent and the impact of steam generator water level uncertainty on LOCA mass and energy release. These letters covered other effects of the same physical phenomenon as Nuclear Safety Advisory Letter 02-3 and the void content of the two phase mixture above the mid-deck plate (NSAL 02-4). Westinghouse also held a workshop with industry representatives on February 28, 2002 and a public meeting with the NRC staff on March 20, 2002.

In its LER, Diablo Canyon indicated that it will submit a license amendment request to revise the Technical Specifications to account for the mid-deck plate differential pressure in the steam generator narrow-range low-low level protection setpoints.