U. S. NUCLEAR REGULATORY COMMISSION
REGION I

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REPORT NO.: 92-17

LICENSEE: Boston Edison Company
800 Boylston Street
Boston, Massachusetts 02199

FACILITY: Pilgrim Nuclear Power Station

INSPECTION AT: Plymouth, Massachusetts

INSPECTION DATES: August 4-14 1992

INSPECTORS: S. Hansell, Operations Engineer, BWR, Operations Branch, DRS — 10/2/92 (Date)

W. H. Ruland for L. Kay, Reactor Engineer, ES Engineering Branch, DRS — 10/2/92 (Date)

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Inspection Summary: The inspectors reviewed licensee actions regarding reactor vessel level indication anomalies due to the buildup of non-condensable gases in reference legs. Specific areas included were operator training and performance for recognition of level error, modifications and results of post-modification testing, and future actions by the licensee including assessment of instrument operability.

Results: The licensee had taken interim corrective actions to compensate for possible level perturbations during a rapid depressurization. The data strongly suggests that the potential problem still exists, especially in the "B" ECCS reference leg. Contingency plans developed by the licensee appeared appropriate. The inspectors concluded that: (1) based on licensee information and independent calculations, the anticipated level error should be small enough not to interfere with automatic ECCS initiation; (2) operators are now sufficiently knowledgeable to recognize and react to postulated level errors; and (3) the EOPs continue to provide sufficient guidance, coupled with operator training, on operator actions when vessel level cannot be accurately determined.

1.0 BACKGROUND

In October 1984, the NRC issued Generic Letter (GL) 84-23, "Reactor Vessel Water Level Instrumentation in BWRs." The scope of the GL was to address inaccuracies in water level instrumentation due to reference leg flashing. Flashing is a condition in the reactor water level reference column wherein the temperature and pressure permit steam voids to form. The GL further presented improvements to provide increased assurance of level instrumentation accuracies. In response to GL 84-23, Boston Edison Company (BECo) had chosen to re-route the reference leg piping to reduce vertical drops within the drywell and replace the existing Yarway combination condensing chamber/temperature equalizing reference column with a cold reference leg located outside the drywell. These modifications reduced the potential for flashing of water in the reference legs.

Pilgrim has had several spurious Group 1 isolations during plant cooldown since 1990. These isolations occurred at pressures less than 100 psig due to spiking reactor vessel level instrumentation. After extensive root cause analysis and study over the past 2 years, the licensee has eliminated virtually all potential causes of the spiking except for non-condensable gases dissolved in the reference leg water.

Recent analysis performed by BECo's consultant and other plant experience has indicated that dissolved gases have been causing the water level perturbations. This raises a potential safety issue. If significant dissolved gases come out of solution in the reference leg during a rapid depressurization, such as during a large loss of coolant accident (LOCA) or actuation of the automatic depressurization system (ADS), the gases could displace or expel water from the reference leg causing level indication to be greater than actual level.

On July 22, 1992, the NRC met with the Regulatory Response Group (RRG) of the Boiling Water Reactor Owner's Group (BWROG) in Rockville, MD to decide a course of action to address the dissolved gases issue. While no immediate action was required, the NRC asked the BWROG to submit short and long-term actions in writing.

2.0 SCOPE

The inspectors reviewed BECo's actions and data related to the reactor vessel water level anomalies to determine: (1) whether operator training and procedures are adequate to compensate for anticipated level indication errors; (2) the adequacy of BECo's corrective actions to date; (3) BECo's future plans; (4) if Pilgrim is bounded by the GE conclusions given at the BWROG meeting; (5) if possible, the extent of the problem at Pilgrim; and 6) BECo's basis for operability of the potentially affected instruments.

3.0 PREVIOUS LICENSEE ACTIVITIES

3.1 Actions to April 1992

Group 1 containment isolations due to spurious water level indications first occurred in March 1990. These events were reported in Licensee Event Report (LER 90-03). Subsequent isolations and licensee review led to plant modifications and procedure changes (see Attachment 2, excerpt from Inspection Report 50-293/92-04). These spurious water level spikes were occurring in a water level measuring system that had been modified in 1987. The reference leg piping was moved outside the drywell to minimize flashing, Rosemount transmitters replaced the Barton differential pressure switches, and the emergency and plant information computer (EPIC) was added in stages to record plant transient data. As recorded in report 92-04, BECo thought the problem was "unsatisfactory thermodynamic performance" of the "B" reference leg condensing chamber (or pot). They also concluded that non-condensable gases were a potential contributor to reducing the chamber and drain line heat transfer capability.

3.2 Plant Testing and Further Modifications

The post-modification testing performed, after removal of the drain line insulations in April 1992, demonstrated to BECo at that time that the problem appeared resolved and the instruments connected to the "B" ECCS reference leg were operable. Recorded test data demonstrated no spiking on the "A" channels and less than one inch of spiking on the "B" side. The "B" side spiking occurred at approximately 400 psig. The licensee determined this modification did improve the thermal performance of the condensing pots on the "B" side. This determination was substantiated at the time by temperature data depicting an increased differential temperature between the top and bottom of the condensing pot and an overall reduction in the temperature of the pot. Additionally, monthly functional checks required by Technical Specifications including a perturbation test, daily instrument channel checks, and monitoring of condensing pot temperatures on a regular frequency were performed by Operations. Based on this data, the licensee declared these instruments operable. However, as noted in a special Operations Review Committee meeting held on April 11, 1992, the licensee concluded that, although no spikes were observed during the testing under controlled depressurized conditions, correction of the root cause could not be ensured. The licensee believed that the root cause may involve the buildup of non-condensable gases following long periods of operation at full power.

3.3 Actions Completed After April 1992

To further understand the vessel level, BECo contracted S. Levy, Inc. (SLI) to perform an independent review of the issue. This review included examination of the licensee's root cause analysis and corrective actions to address reactor vessel level perturbations. SLI's report was issued on July 15, 1992.

SLI postulated that the level spiking was caused by the evolution of non-condensable gases from the reference legs due to a plant depressurization. Steam entering the condensing pot carries with it a small concentration of non-condensable gases. These gases are predominantly hydrogen and oxygen from the radiolytic decomposition of water. These gases could accumulate in the condensing pot to appreciable partial pressures if the reflex flow back to the reactor is not successful in returning gas-laden liquid to the reactor. Thus, the gases diffuse into the liquid in the condensing pot. However, further diffusion of the gases down the reference leg would only occur over long periods of plant operation. SLI proposed that small leaks in the reference leg or thermal convection could carry gas-laden water down the reference leg to the instrument racks. This could create a reference leg filled with a saturated solution of gas in water.

Under a postulated rapid depressurization event, these gases would come out of solution and form bubbles. While these bubbles are traveling up the reference leg and encountering both horizontal and vertical piping runs, they form slugs of gas (i.e., a bubble filling the entire pipe cross-section) in the water. The presence of gas causes the hydrostatic pressure of the reference leg to fall which is interpreted by the instrumentation as a rise in level.

The non-condensable gas theory was confirmed by SLI using a formula that predicted the level indication behavior during cooldown. Based on the calculated velocity of a slug of gas traveling up a vertical portion of the reference leg, SLI predicted the length of the vertical pipe runs in the "B" ECCS leg. Those predictions were within 5% for two vertical sections and within 21% for a short third vertical section. The calculations and review were characterized as a "first cut" by SLI with additional analysis required to identify the root cause and other postulated contributors.

3.4 **Further Actions by the Licensee**

Based on the SLI report findings, the licensee determined additional analysis must be performed to confirm this theory. The licensee stated to the inspectors that current information does not completely support the theory. The licensee has developed a contingency plan for gathering, evaluating, and dispositioning information on this issue.

The licensee has contracted SLI to further investigate and analyze the effects and root cause of this phenomenon. On September 18, 1992, SLI submitted an additional preliminary draft report that provided additional analysis and proposed modifications. The general conclusions in that report supported SLI's original analysis. In addition, Pilgrim has provided their plant data, including temperature profiles of the instrumented condensing pots, to the BWROG. Further actions by the licensee also include another possible modification to improve the sloping of the reference leg at the instrument racks. This negative slope down to the measuring instrument is a recommended action by General Electric presented in Service Information Letter (SIL) 470, dated September 16, 1988, to minimize or eliminate level mismatch occurrences. The licensee expects to complete this modification during the next refueling outage scheduled for April 1993.

On July 29, 1992, the BWROG Regulatory Response Group presented the NRC with information related to the postulated effects of non-condensable gases on reactor vessel level indication. Pilgrim actively participated in this meeting and presented their plant data addressing level spiking under depressurized conditions. Conclusions reached from this meeting involved the need to determine this phenomenon's effect on each plant based on specific configuration and application. NRC asked the BWROG to gather information from each BWR and develop a generic analysis to envelope all BWRs.

3.5 **Current Conditions**

At the end of the inspection, Pilgrim was operating at 100% power, continuing to perform routine channel checks of the water level instruments. These checks continue to show good correlation among level instruments. BECo was monitoring the temperature of the condensing pots and the nozzle leading back to the reactor vessel through temporarily installed surface-mounted resistance temperature detectors (RTDs). The "B" ECCS condensing chamber surface temperature has continued to trend downward. They had established a project plan to pursue preliminary review of possible plant modifications, if warranted. As part of the BWROG, they will be participating in the generic resolution of the issue, which includes responding to the Generic Letter as required. BECo also plans to monitor closely any shutdowns and cooldowns performed to characterize any change in condensing pot performance.

Finally, BECo maintains that the reactor vessel level instruments remain operable at Pilgrim. They base this conclusion, as contained in their documented Safety Assessment, on several factors: (1) the level fluctuations above 600 psig would be minor and would not affect FSAR transient and accident analysis, i.e., automatic initiations of safety systems would occur as designed; (2) core uncovery is not expected below 600 psig if the low vessel level containment isolations have not occurred and operators could take actions to close isolation valves as necessary; (3) they expect all level actuations to occur since, on a rapid depressurization, the gases coming out of solution would also exert a pressure on the reference leg side of the differential pressure instruments, tending to reduce the magnitude of the level error; and (4) if operators are unable to determine reactor vessel level, the EOPs provide clear direction to flood the reactor vessel.

4.0 **OPERATOR PERFORMANCE, TRAINING, AND PROCEDURE REVIEW**

The inspectors reviewed the performance of the operators with their current procedures and training. This review attempted to judge the appropriateness of BECo's assertion that the operators would decide correctly when reactor vessel level could not be determined.

4.1 Operator Training, Performance and Knowledge

The inspector evaluated two operating crews' response to reactor water level instrument failures in the simulator. The simulator scenarios were unannounced and varied from rapid vessel depressurization events with reactor level instruments failing upscale to a loss of offsite power event with a gradual reactor level instrument line failure, one channel at a time. In addition, six licensed operators were interviewed to determine their knowledge of the reactor water level instrument concern and related reactor water level instrumentation design considerations.

The operators quickly diagnosed the different reactor level instrument line failures and used the appropriate plant emergency operating procedures (EOPs) to maintain adequate core cooling at all times. Both crews displayed a safe and conservative approach to combat the simulated plant transients. The crew communications, command and control were excellent. The operators understood how a rapid depressurization could affect the response and accuracy of the reactor water level instrumentation. The operators explained the phenomena of non-condensable gases coming out of solution and its effect on indicated reactor water level. Correct responses were provided by the operators in response to questions about postulated instrument malfunctions, including reference leg flashing to steam. The operators provided positive comments about the training departments instruction regarding this issue.

The operators were familiar with recent reactor water level modifications that were implemented at Pilgrim. A detailed knowledge of the EOPs that applied to reactor water level instrument failures was demonstrated. The definition of the EOP term "If RPV water level CANNOT be determined," was consistent with the BWROG definition. All operators observed in the simulator or interviewed knew how to ascertain if RPV water level could NOT be determined.

One area identified for improvement related to the operators capabilities for using the Safety Parameter Display System (SPDS) computer. Some operators were not familiar with the computer systems response to reactor water level instrument failures. The inspector recognized that part of the problem could be that the SPDS computer is not functional in the plant specific simulator. The SPDS computer is scheduled to be operational in the simulator by January 1993. The operators current knowledge was gained from using the SPDS computer in the control room.

4.2 Training Material Adequacy

The inspector reviewed the training administered to the licensed operators regarding reactor water level instrument response to a rapid RPV depressurization. The plant Training department instructors conducted the training using a structured lesson plan. The lesson plan was comprehensive and provided a detailed description of non-condensable gas formation, migration of the gases to the reactor water level instrument reference leg, and how the gases could come out of solution and affect level indication. The licensed operators were provided a copy of the information from the BWROG presentation to the NRC discussed above in section 4.1.

The inspector also reviewed the training administered during the licensed operator requalification program. The training is covered in classroom presentations and dynamic simulator scenarios which are both part of the annual licensed operator requalification program.

The training instructor's reactor water level instrument lesson plans were comprehensive and included recent plant modifications, licensee event reports, and plant operating experience. Also, the reactor water level instrument failure scenarios' covered a broad spectrum of plant conditions, including rapid depressurization, failure to scram, and full and low reactor pressure events. The bank included an appropriate number of reactor level instrument failure scenarios to test all legs of the applicable EOPs.

4.3 Procedure Content and Implementation

The inspector reviewed the plant operating and alarm response procedures (ARPs) related to reactor water level instrumentation malfunctions.

The "Rx water level HI/LO," ARP provides adequate procedure guidance for reactor water level instrumentation failures during normal operation. The ARPs contain a list of the specific reactor water level instruments, by name and panel location, for the operators to check in the event of a failure. The ARPs direct the operators to verify that all appropriate automatic actions, i.e., reactor scram, have occurred.

The EOPs provide detailed procedure guidance for instrument failures during degraded plant conditions. If "reactor water level CANNOT be determined," the EOPs provide clear direction to depressurize the reactor vessel and manually start emergency core cooling system (ECCS) pumps, to flood the vessel above the reactor core. The RPV flooding EOP relies on the ECCS pump discharge pressure and open safety relief valves (SRVs) to ensure adequate core cooling. The core flooding EOP can be performed without the use of any reactor water level indications.

The EOPs also contain a specific caution step, caution one (1), that directs the operators not to use certain reactor water level instruments, based on plant conditions. The caution contains a graph to alert operators when the use of reactor water level instruments are not safe.

The inspector reviewed the comparison of the plant's implementation of the EOPs compared to the BWROGs guidance. The plant EOPs do not deviate from the intent of the BWROG for the RPV flooding EOP or caution one. The plant has changed individual words to make their EOPs easier to use. Also, caution one was located in the required EOPs at the appropriate location of the flow charts.

5.0 CONCLUSIONS

Based on the inspectors' review of the licensee's actions, independent review of the contractor's report, review of past behavior of reactor vessel level indications, and observations and interviews with operators, the inspectors reached the following conclusions, based on the NRC's current understanding.

5.1 Operator Training and Performance

Operators were sufficiently aware of possible reactor vessel level perturbations such that there is a high confidence that they would take the proper actions in the event of a vessel level anomaly occurring during a rapid depressurization.

5.2 Adequacy of BECo's Corrective Actions

Based on the inspectors' review of actions to date, BECo continues to apply appropriate resources in an attempt to resolve the issue.

BECo declared the level instruments operable after the post-modification testing following the removal of insulation from the drain lines. That conclusion was reached before the SLI report described the postulated mechanism for transporting gas down the reference leg. While the inspectors determined that the adequacy of the post-modification testing may have been appropriate with the information then in hand, the test did not show that gases could not be a potential problem following further operating time. Further, continued decreasing surface temperature of the "B" ECCS condensing chamber strongly suggests that non-condensable gases continued to buildup in the chamber. Thus, the inspectors could not conclude whether previous modifications had eliminated the problem. The inspectors did, however, determine that BECo's current course of action was reasonable. This assessment included the operator training completed.

5.3 BECO's Plans

The inspectors reviewed BECo's contingency plans and current schedules to resolve the water level anomalies. The contingency plan was revised on July 20, 1992, to account for the new information supplied by SLI. That plan included additional work by SLI, development of possible design changes, and assigning a response team for reactor water level spike data evaluations.

Based on BECo's actions to date and current plans, the inspectors concluded that their plans were reasonable.

5.4 BWROG Conclusions

The BWROG stated that safety system initiations occur before reactor pressure vessel depressurizations could induce significant water level errors (effect is negligible above ~450 psig), safety system initiation on high drywell pressure is unaffected, and that Emergency Procedure Guidelines (EPGs) specify post-LOCA actions. Further, the BWROG stated that actions are planned to assure appropriate operator response.

The inspectors concluded that: 1) based on licensee information and independent calculations, the anticipated level error will be small enough not to interfere with automatic ECCS initiation; 2) operators are now sufficiently knowledgeable to recognize and react to the postulated vessel level indication problems; 3) the EOPs continue to provide sufficient guidance, coupled with operator training, on operator actions when vessel level cannot be determined. The inspectors found that the key point was the operator decision that "vessel level cannot be determined" and that operators reached that conclusion during the simulator scenarios using past experience and judgement. The EOPs did not provide detailed guidance on when to conclude that vessel level was indeterminate. The inspectors found that the operators displayed sufficient knowledge of the normal behavior of level during a depressurization to enable them to recognize water level anomalies.

5.5 Magnitude of Level Error at Pilgrim

Based on review of the licensee's data, the SLI report, the BWROG presentation, and other industry information, the inspectors concluded that the non-condensable gas problem has existed at Pilgrim for at least several years, most notably in the "B" ECCS reference leg. The continued downward trend of temperature observed on the "B" ECCS condensing chamber suggests to the inspector that non-condensable gases are continuing to build up in the "B" ECCS pot. However, since the reference leg continues to appear full as shown by the instruments continuing to pass routine channel checks, the pot continues to keep the reference leg full by condensing steam. Further demonstration of steam condensing within the pot is the continued makeup of the current leakage (measured by BECo as 100 grams/24 hours).

The inspector could not determine how large the level error would be during a rapid depressurization. Unknowns included actual gas concentrations, uncertainties in the condensing pot temperature measurements, and the effect of flow orifices on the depressurization behavior. Some uncertainty in the temperature measurement was demonstrated by RTD temperature indication changes due to a change in drywell ventilation. However, rough calculations performed by the NRC showed that significant level errors should only occur after ECCS equipment initiates on low vessel level signals.

5.6 BECO's Operability Assessment

The inspector did not have any significant concerns with the licensee's conclusion that the associated instruments were operable. The level channels have passed daily channel checks as required by the licensee's technical specifications. ECCS equipment should initiate before significant errors appear. Operators have been appropriately trained to deal with level errors that occur. However, the instruments used for the post-accident monitoring function (Regulatory Guide 1.97) clearly do not fully support the operators' actions post-accident if the problem magnitude is ultimately confirmed. The inspector noted that the instruments that provide the wide range post-accident vessel level monitoring function were not included in the technical specifications.

The licensee's safety assessment alluded to taking some credit for the positive level effects on the level instruments due to a postulated depressurization although there was no data to support this claim. BECo stated that they did not take credit for this effect, but included it in the safety assessment as another factor to be considered.

6.0 EXIT MEETING

The inspectors reported their preliminary findings to the licensee on August 14, 1992, with one inspector at the site and the team leader participating by telephone from the regional office. No draft documents were supplied to the licensee.

Attachments:
1. Persons Contacted
2. Excerpt from Report 50-293/92-04
3. Licensee Documents Reviewed
4. References

ATTACHMENT 1

PERSONS CONTACTED

Bostor Edison Company

J. Alexander, Training Manager
E. Almeida, Systems Engineer
R. Anderson, Senior Vice President, Nuclear
J. Bellefeuille, Technical Section Manager
R. Bolduc, Instructor
N. Desmond, Compliance Division Manager
R. Fairbank, Nuclear Engineering Dept. Manager
P. Hamilton, Licensing Division Manager
K. Kampschneider, Lead System Engineer
E. Kraft, Plant Manager
T. McElhinney, Sr. Compliance Engineer
H. Oheim, Regulatory Affairs Manager
W. Rothent, Nuclear Engineering Director
M. Santiago, Simulator Instructor
P. Smith, Acting Level Issue Manager
P. Smith, Operations Support Manager
T. Swan, Training Supervisor
T. Trepanier, Chief, Operating Engineer/Acting Operations Section Manager
T. White, Systems and Safety Assessment Manager

Commonwealth of Massachusetts

J. Muckerheide, State Nuclear Engineer

U.S. Nuclear Regulatory Commission

D. Kern, Resident Inspector
J. MacDonald, Senior Resident Inspector

Visual inspection of th~ MO-1301-53 actuator during the testing evolution indicated a degradation of the grease lubricant and a twisted declutch shaft. The valve was locked in the fully closed position and the actuator was rebuilt. Maintenance personnel utilized this opportunity to install an improved type of grease lubricant and to upgrade the torque switch. Post work VOTES testing was performed as required by procedure 8.Q.3-8 following replacement of the torque switch. A loud ratcheting noise was heard upon cycling of the valve. The test coordinator immediately directed a prepositioned operator to open the power supply breaker to MO-1301-53. Prudent prepositioning of an operator and quick action by the test coordinator prevented damage to the actuator. Subsequent inspection determined that a : allic spacer designed to provide alignment between the worm shaft and the worm shaft clutch gear had not been reinstalled during actuator reassembly. The missing spacer resulted from miscommunication between technicians during reassembly. The maintenance supervisor properly addressed communications practices with maintenance technicians to preclude recurrence. Maintenance personnel inspected the actuator, replaced the missing spacer, and reassembled the actuator. The torque switch was adjusted during post work VOTES testing to account for such variables as equipment error, torq''e switch repeatability, and rate of loading effects consistent with NRC Generic Letter 89-10. The resultant "as left" thrust value was measured to be 17,587 pounds. The inspector determined the adjustments to the torque switch and resultant thrust to be appropriate to assure valve operability.

8.2 Reactor Vessel Water Level Instrumentation Spiking

In recent years, the licensee has experienced reactor vessel water level instrumentation spiking during reactor depressurization following plant shutdowns. Typically, the level instrumentation spiking has been observed to begin at less than 600 psig reactor pressure and has been much more prominent on the reactor vessel "B" reference leg instrumentation. The spiking has been of sufficient magnitude to cause several automatic primary containment isolation system (PCIS) Group I isolations. Corrective actions to previous level instrument spiking included improved sensing line backfill procedures and implementation of a modification which increased the reactor vessel to condensing chamber equalizing line diameter from one inch to two inches.

Notwithstanding these corrective actions, level instrument spiking and three automatic Group I PCIS actuations occurred during the March 26 and 27 post shutdown reactor depressurization. The second of the three isolations was the result of an actual high vessel water level and is documented in Section 2.3.

As during previous events, the March 26 spiking was initially observed at approximately 450 psig reactor pressure and was initially limited to "B" reference leg instrumentation which did not result in PCIS actuation. The spikes were typically of 30 seconds duration, were similar to a square wave recorder trace, and were of approximately positive four inches in amplitude. Level spiking on "A" reference leg instrumentation was initially observed at 65 psig reactor pressure, but the spikes were typically of positive one to two inches in amplitude.

Both the first and third automatic Group I isolations were initiated from instrument spiking on the "B" reference leg. The first isolation occurred on March 26 at 8:59 pm, with the reactor shutdown and at approximately 55 psig pressure when the "B" reference leg instrumentation experienced a spike of positive nineteen inches from +29 to +48 inches. Instrumentation on the "A" reference leg remained unchanged through this occurrence. The third isolation occurred on March 27 at 5:45 am, with the reactor at 10 psig pressure and with the shutdown cooling system being placed in service. Instrumentation on the "B" reference leg spiked positive eighteen inches from +29 inches to +47 inches causing the PCIS actuation. Instrumentation on the "A" reference leg spiked positive fourteen inches from +29 inches to +43 inches but remained below the PCIS actuation setpoint. All components responded to each Group I isolation signal as designed.

The licensee formed a root cause analysis team (RCAT) to further investigate the continuing level instrumentation spiking. In addition to licensee personnel from operations, system engineering, and nuclear engineering, the team received technical expertise from General Electric Company and Yankee Atomic Electric Company instrumentation specialists. The team determined the root cause of the level instrumentation spiking to be unsatisfactory thermodynamic performance of the "B" reference leg condensing chamber and associated steam drain line. The team concluded that, during reactor depressurization the temperature of the condensing chamber and drain line metal surfaces exceeds the saturation temperature of the reactor coolant, causing condensate in the drain line to vaporize and flow rapidly into the reactor vessel. This action would cause more vapor to vacate the condensing chamber, creating a reduced pressure condition within the chamber. The reduced chamber pressure would be sensed by the level instrumentation as a high vessel level spike.

The team also identified the buildup of non-condensible gasses in the condensing chamber as a potential contributor to these events. It was believed that, as non-condensible gasses accumulate in the chamber during plant operations, the condensation rate is decreased which in turn reduces the chamber and drain line heat transfer capability.

In order to improve heat transfer from the condensing chamber and drain line to the drywell atmosphere which would serve to reduce surface metal temperatures, the licensee removed the "B" reference leg condensing chamber and drain line insulation via a temporary modification (TM 92-13). Additionally, more temporary temperature instrumentation was installed (via TM 91-44) on both the "A" and "B" reference leg condensing chambers and associated drain lines in order to monitor component thermodynamic performance during power operations and during reactor shutdowns. This instrumentation should also provide capability to trend potential buildup of non-condensible gasses in the condensing chambers.

8.2.1 Engineered Safety Features Actuation During Troubleshooting

The RCAT also identified trapped air in the sensing lines as a potential contributing factor to the instrument spiking. Although the licensee previously implemented procedures to improve sensing line backfill, the lines had never been verified to be free of trapped air. Therefore, in order to evaluate this potential, the licensee developed Temporary Procedure, TP 92-20, "Reactor Level Instrument Line Test and Investigation on Rack 2206 (Constant Level, Decreasing Pressure)." The procedure was intended to be a test which simulated a reactor shutdown by decreasing a static pressure applied to the level instrumentation.

On April 3, TP 92-20 was initiated. Procedure step 10.2.2 directed the removal of analog trip system (ATS) master trip units for reactor level and pressure which input into the emergency core cooling system (ECCS) logic. However, the sequence established by the procedure created a configuration in which "B" logic circuitry slave trip units LS-263-72D-1 and LS-263-72B-1 concurrently received low-low reactor vessel water level trip signals when their respective master trip units were removed. The concurrent signals satisfied ECCS initiation logic which resulted in the automatic start of both emergency diesel generators, automatic start of the "A" train of RHR with associated valve repositioning, repositioning of associated valves in the "B" train of RHR (which was in the shutdown cooling mode of operation), and the automatic opening of HPCI steam supply and injection valves. Because the reactor was shutdown, the HPCI system did not initiate. Additionally, because the LPCI cross-tie line was isolated during the outage and because the LPCI loop select logic was selected to the "B" loop, the "A" train RHR pumps operated in minimum flow recirculation. All affected equipment responded to the ECCS initiation signal as designed.

The test was immediately terminated, the master trip units were reinstalled and the ATS logic was reset, and normal safety system status was restored. Technically, the event had minimal safety significance. Decay heat removal was maintained throughout the event and all systems and components performed as designed.

The licensee promptly conducted a thorough review of this event. Proper system responses were verified and a causal analysis was initiated. The cause of this event was determined to be inadequate procedure development and review. As a result, Temporary Procedure 92-20 did not establish appropriate actions to preclude the actuation. Additionally, subsequent reviews of the procedure by a procedure validator and the Onsite Review Committee failed to identify the procedure deficiency. Specifically, the reviewers did not identify that removal of master trip units with the respective slave trip units that have low reactor water level or low reactor pressure functions in service, would cause the associated trip relays to be energized.

The individuals involved in the development and issuance of TP 92-20 were counselled to ensure their responsibilities were understood. Subsequently, the licensee retired the faulted procedure and generated a new procedure (TP 92-22) to test for the presence of air in the sensing lines. This procedure required the individual slave trip unit to be removed before its associated master trip unit. Procedure TP 92-22 was performed without further incident on April 5-7.

Technical Specification (TS) 6.8.A requires that written procedures shall be established and implemented that meet or exceed the requirements of Sections 5.1 and 5.3 of ANSI N18.7-1972 and Appendix "A" of USNRC Regulatory Guide 1.33. Additionally, TS 6.8.B requires that such procedures be reviewed by the ORC and approved by the responsible department manager.

Contrary to these requirements, TP 92-20 was established and implemented with deficiencies that caused a partial ECCS actuation. Further, the ORC review failed to question the deficiencies. Notwithstanding, the licensee effectively identified the cause of this event, initiated prompt corrective actions, and properly reported the event in Licensee Event Report 92-05 (Section 7.2.5). Therefore, the licensee-identified violation is not being cited because the criteria specified in Section VII.B of the NRC's Enforcement Policy were satisfied.

8.2.2 Temporary Waiver of Compliance and Post Modification Testing

The RCAT concluded the most probable cause for the level instrumentation spiking was unsatisfactory "B" reference leg condensing chamber thermodynamic performance. Additionally, the team concluded with reasonable assurance that removal of the condensing chamber and drain line insulation would be the appropriate corrective action to the spiking. By letter dated April 7, the licensee requested a Temporary Waiver of Compliance (TWOC) from TS limiting conditions for operations requirements associated with the affected reactor vessel water level instrumentation. The purpose of the waiver request was to permit reactor startup to not more than 15% of rated power to conduct post modification (i.e. insulation removal) testing of the "B" reference leg instrumentation. The licensee concluded the waiver request was necessary to comply with the post modification testing requirements as stated in the Boston Edison Quality Assurance Manual. The waiver request included a detailed bounding safety evaluation. After comprehensive staff review, a Regional Waiver of Compliance was granted consistent with NRC letter dated April 8.

On April 9 at 8:59 am, the licensee commenced reactor restart. Reactor power was increased to approximately 12% where it was maintained for approximately ten hours to allow equilibrium temperatures to be achieved. On April 10 at 6:18 pm, reactor shutdown was initiated and the reactor entered cold shutdown on April 11 at 3:35 am. Reactor vessel level and condensing chamber performance data was recorded throughout the reactor power evolution in accordance with post modification test TP 92-21. The results of the test evolution identified all instrumentation level spikes to be one inch or less in magnitude. Based on these results, the licensee declared the affected instrumentation operable and prepared the station for reactor restart to full power operations.

The post modification testing was extremely well controlled. Control room operators maintained excellent awareness of all test related activities, including outstanding control of all reactivity manipulations. Test coordinators ensured complete data acquisition and analysis. Although the test was effective in evaluating most aspects of condensing chamber thermodynamic performance, a test limitation was the inability to establish potential effects of the buildup of non-condensible gasses. Extended level instrument performance with respect to potential non-condensible gas buildup effects is identified as an unresolved item (UNR 50-293/92-04-02).

The licensee investigation of the continuing level instrumentation spiking was very well controlled with the noted exception of the unplanned partial ECCS actuation. Licensee management provided the RCAT with necessary support and technical expertise. The RCAT evaluations were comprehensive and reflected safety conscious perspectives. The TWOC and associated safety evaluation were effectively supported by design and licensing bases documentation. Post modification testing was performed in a deliberate and conservative manner.

9.0 NRC MANAGEMENT MEETINGS AND OTHER ACTIVITIES (30702)

9.1 Routine Meetings

At periodic intervals during this inspection, meetings were held with senior plant management to discuss licensee activities and areas of concern to the inspectors. At the conclusion of the reporting period, the resident inspector staff conducted an exit meeting with licensee management summarizing inspection activity and findings for this report period. No proprietary information was identified as being included in the report.

9.2 Management Meetings

On April 7, a conference call was conducted between representatives of NRC: Region I, NRR, and the licensee to discuss operability of certain reactor vessel water level instrumentation and a related Temporary Waiver of Compliance request. This subject is discussed further in Section 8.2.2 of this report.

9.3 Other NRC Activities

On March 18, Mr. Thomas Martin, the Regional Administrator, NRC:Region I toured PNPS and met with licensee management to discuss current licensee performance.

On May 1, NRC Chairman Ivan Selin, Executive Director for Operations James Taylor, and Region I Administrator Thomas Martin toured PNPS and met with Boston Edison Company corporate officers to discuss current performance and future licensee initiatives. A press conference was conducted at the conclusion of the site visit.

On May 3, Mr. John Rogge, NRC Region I Chief, Reactor Projects Section 3A responsible for inspection program management at PNPS was reassigned to become Chief, Reactor Projects Section 4B. Mr. Eugene Kelly, previously Region I Chief, Reactor Projects Section 4A, has succeeded Mr. J. Rogge.

ATTACHMENT 3

LICENSEE DOCUMENTS REVIEWED

BECo Safety Assessment of Potential Effects of Non-Condensables in Reference Legs during Rapid Depressurization Events, Revision 0, August 11, 1992.

Most Probable Cause of Water Level Indicators Anomalies at Pilgrim Station, SLI Report BEC-001-R-01(Q), Revision 0, July 15, 1992.

Piping and Instrumentation Drawing (P&ID) Nuclear Boiler Vessel Instrumentation, M253, Revision E22, November 4, 1991.

Nuclear Boiler System Reference Leg Isometric Piping Drawings.

Pilgrim Updated FSAR Section 7.8.5.2

Pilgrim LERs 90-003, 91-08-01, 92-004

Multi-Discipline Analysis Team Report for 10/30/91 Event, Revision 1, November 20, 1991.

Root Cause Analysis Report for Reactor Water Level Spiking, April 16, 1992.

Reactor Water Level Spiking Contingency Plan, July 20, 1992 and May 15, 1992.

Condensing Chamber Temperature vs. Time, Various Graphs.

EOP-1, "RPV Control," and Caution One, Revision 1

EOP-16, "RPV Flooding," Revision 0

EOP-17, "RPV Depressurization," Revision 0

Pilgrim PSTG for Implementation of the BWROG EOPs, Revision 4

LOR Training Instruction Module, "Reference Leg Perturbations due to Non-Condensable Gases," Revision 0

LOR Training Instruction Module, "EOP-16 and EOP-26 RPV Flooding"

ATTACHMENT 4

REFERENCES

1. Carslaw and Jaeger, "Conduction of Heat in Solids," 2nd Edition. Oxford at the Carendon Press, pp. 92-99.

2. Govier and Aziz, "The Flour of Complex Mixture in Pipes," Van Nostrand Reinhold Co., pp. 395-396.