Commonwealth Edison
1400 Opus Place
Downers Grove, Illinois 60515

July 30, 1993

Dr. Thomas Murley
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Attention: Document Control Desk

Subject: Commonwealth Edison Response to
NRC Bulletin 93-03, "Resolution of Issues Related to
Reactor Vessel Water Level Instrumentation in BWRs"
Dresden Units 2 and 3, Docket #50-237 and 50-249
Quad Cities Units 1 and 2, Docket #50-254 and 50-265
LaSalle Units 1 and 2, Docket #50-373 and 50-374

Reference: 1. NRC Generic Letter 92-04, "Resolution of the Issues Related to Reactor Vessel Water Level Instrumentation in BWRs Pursuant to 10CFR50.54(f)"
2. NRC Information Notice 92-54, "Level Instrumentation Inaccuracies Caused by Rapid Depressurization"

Dear Dr. Murley,

In response to NRC Bulletin 93-03, "Resolution of Issues Related to Reactor Vessel Water Level Instrumentation in BWRs", Commonwealth Edison (CECo) submits this joint response containing information particular to each of our BWR sites - Dresden, Quad Cities and LaSalle County Stations.

Previously CECo requested, through the BWR Owners Group (BWROG), a delay in the implementation of long term corrective actions until a test program could be completed by the BWROG. This request was made by the BWROG to the staff on behalf of the industry and was approved. Subsequent to the deadline being moved to July 1993, an event occurred at Washington Public Power System Unit 2 (WNP-2), where a sustained level error was observed during a normal plant cooldown. This event led to concerns that automatic isolation instrumentation may not function to isolate a leak in the Residual Heat Removal (RHR) system during shutdown cooling operation.

The NRC staff requested the BWROG to evaluate the effect of level indication errors on events having the potential to drain the vessel. The Owners Group replied with a report that stated the most limiting drain-down event is an RPV drain-down to the suppression pool through the low pressure coolant injection suction flow path. Based on this information the staff concluded that additional compensatory measures are needed for normal cooldown evolutions. The staff also concluded that for long term resolution of the problem, hardware modifications are needed.

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In Bulletin 93-03, the staff requested specifics on both short term compensatory actions and hardware modifications that the utilities intend to implement to resolve the de-gas problem. In compliance with the request outlined in Bulletin 93-03, Commonwealth Edison hereby submits the required information concerning compensatory measures and hardware modifications. Attachments A through F provide all information on short term compensatory actions and hardware modifications for each of the generating sites.

In addition, Attachment G contains a summation of the technical information provided to the staff at the July 26, 1993 meeting held at NRR, wherein CECo outlined a number of technical issues related to the backfill modification and provided the staff with a proposed implementation schedule for the modification at each CECo BWR generating station.

If you have any questions or comments, please direct them to this office.

Respectfully,

L. O. Del George

L. O. Del George
Vice President
Nuclear Operations Support

Attachments:
- A. CECo Actions to Implement Requested Actions
  - Attachment A1: Dresden
  - Attachment A2: Quad Cities
  - Attachment A3: LaSalle
  - Attachment A4: Summation of Compensatory Actions
- B. Hardware Modifications
- C. Evaluation of Compensatory Measures
- D. Design Development and Phased Implementation
- E. CECo Nuclear Generating Stations Refuel Schedule
- F. Modification Team Project Plan
- G. Summary of July 26, 1993 Meeting

cc: J. Martin, Regional Administrator-RIII
J. Stang, Project Manager, Dresden-NRR
C. Patel, Project Manager, Quad Cities-NRR
J. Kennedy, Project Manager, LaSalle-NRR
M. Leach, Senior Resident Inspector-Dresden
T. Taylor, Senior Resident Inspector-Quad Cities
D. Hills, Senior Resident Inspector-LaSalle
Office of Nuclear Safety-IDNS

Attachment A
CECo Actions to Implement Requested Actions

In Bulletin 93-03 all licensees were asked to implement short term compensatory actions to deal with the reactor vessel level degas issue until modifications could be implemented. Those requested actions fell into four broad categories:

1. Enhanced Monitoring,
2. Enhanced Procedures,
3. Immediate Operator Training, and
4. Near Term Operator Training (prior to July 30, 1993)

All three CECo BWR sites have taken actions to comply with those categories, as seen in Attachments A1 (Dresden), A2 (Quad Cities) and A3 (LaSalle County). Each station has implemented enhanced monitoring, reviewed its procedures for the opportunity to enhance them and trained its operators, both in the immediate and near term time frame.

As part of the response to this Bulletin, the Heightened Level of Awareness (HLA) Program was used at the stations to convey the seriousness of management's approach to this problem. The HLA program was developed from INPO SOER 91-01 to aid the stations in ensuring that issues and problems are given the proper amount of attention by all levels of plant personnel.

The Training Departments at all three sites played an integral role in timely and appropriate development of training modules and labs for the operators. The training provided covered all aspects of the issue - detection, identification and appropriate actions to mitigate the occurrence.

Criteria were developed by the Engineering groups at each site for identification of a notch event to ensure consistent entry into compensatory actions and possibly into Technical Specification Action statements. Each site developed their criteria based on individual plant configuration and reference leg function, which ensure that the reliability of plant safety functions are preserved.

All three stations have completed Operability Evaluations for this issue. With the implementation of compensatory actions, all stations declared their water level instrumentation systems operable.

The actions taken, that is: HLA, enhanced procedures and monitoring, and training, justify the continued safe operation of the units until appropriate modifications can be installed. This installation schedule is described in Attachment D.

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

Dresden Station has established enhanced monitoring of the RPV level instruments during reactor shutdown activities when the effects of non-condensible gasses on the reference legs have the potential to cause reactor level indication offsets. The enhanced monitoring provides the following guidance for the operators:

> Prior to reaching Hot Shutdown conditions, the Control Room computer trend recorders will be placed in service at high speed to trend the narrow and medium range level indication computer points. If a computer trend recorder is unavailable, utilize the computer scan function to monitor levels.
>
> Prior to reaching Hot Shutdown conditions, a Heightened Level of Awareness (HLA) will be conducted on the potential for RPV level indication error occurrence as a result of non-condensible gasses coming out of solution. This HLA will be maintained until the reactor is in Cold Shutdown.

This guidance, along with the information provided during routine training, enables the Operating staff to identify the phenomena of non-condensible gasses should degassing occur. This guidance compares the individual level indications against an established acceptance criteria which ensures that all protective trips will occur should degassing be encountered on a particular reference leg. Acceptance criteria have been established that define a notch as an anomalous high level indication greater than 9 inches for more than 5 minutes. The basis for this criteria is the large margin between the trip setpoint and the variable leg tap for the limiting event of concern (Shutdown Cooling System Isolation). This margin is approximately 68 inches, which is significantly larger than the worst case observed indication offsets (3" at Quad Cities, 8" at LaSalle). The other constraint on the criteria selection is that the criteria must be large enough to prevent unnecessarily declaring instruments inoperable. Such a declaration would be non-conservative during Shutdown Cooling operation because it would necessitate manually isolating a successfully operating core cooling system. Level channels that fail to meet this criteria will be declared inoperable. Should this occur, the actions outlined in Technical Specification Section 3.1, Reactor Protection System and 3.2, Protective Instrumentation, will be taken. Any reference leg that has been declared inoperable will be backfilled prior to declaring the instruments operable.

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

Dresden Station has performed a review of Station procedures that have the potential to drain the vessel. The results of the review confirmed that administrative controls have been established to ensure that valve alignments and maintenance activities with the potential to drain the RPV are controlled. Significant Operating Experience Report (SOER) 87-2, Inadvertent Draining of Reactor Vessel to Suppression Pool at BWRs, addressed concerns with plant design, procedures, and training while discussing an inadvertent draindown of the vessel. Dresden Station's review of this SOER prompted many of these administrative controls. The following procedures currently contain administrative controls to warn the operator of the potential for draining the vessel.

Procedure	Actions
DOP 1000-03, Shutdown Cooling Mode of Operation	Procedural steps advise the operator of valve line-ups that may drain the vessel.
DOP 1000-05, Reactor Water Heating Mode of Operation of Shutdown Cooling	Procedural steps advise the operator of valve line-ups that may drain the vessel.
DOP 1500-02, Torus Water Cooling Mode of Low Pressure Coolant Injection System	Procedural steps advise the operator of valve line-ups that may drain the vessel.
DOP 1500-03, Containment Spray Cooling Mode of Low Pressure Coolant Injection System	Procedural steps advise the operator of valve line-ups that may drain the vessel.
DOS 0040-07, Verification of Remote Position Indication for Valves Included in In-Service Testing (IST)	A procedural note and steps advise the operator of line-ups that will drain the vessel when valve interlocks are defeated.
DOS 1500-01, LPCI System Valve Operability Test	Procedural steps advise the operator of valve line-ups that may drain the vessel.
DOS 1500-04, LPCI System Operability Test with the Torus Unavailable	Precautions in the procedure provide the operator with valve line-ups that may inadvertently drain the vessel.
DOS 1500-05, LPCI System Quarterly Flow Rate Test	Procedural steps advise the operator of valve line-ups that may drain the vessel.
DOS 1500-06, LPCI System Pump Operability Test with Torus Available	Procedural steps advise the operator of valve line-ups that may drain the vessel.
DOS 1500-10, Quarterly LPCI System Pump Operability Test	Procedural steps advise the operator of valve line-ups that may drain the vessel.
DSSP 0100-E, Hot Shutdown Procedure, Path E	A Caution warns the operator that automatic isolations may not occur when supply breakers to the valves are open.
DSSP 0100-F, Hot Shutdown Procedure, Path F	A caution warns the operator that automatic isolations may not occur when supply breakers to the valves are open

The above list demonstrates that Dresden Station has adequate administrative controls to prevent an inadvertent draindown of the vessel.

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

The Dresden Station Training Department has conducted several training sessions on the non-condensible gas issue. Included in this training was a discussion on the degassing phenomena and the effects on level instrumentation and anticipated corrective actions. These training sessions have included classroom lectures, shift briefings, required reading packages, operator orders, and brief memorandum.

The training was originally initiated when Generic Letter 92-04 was issued by the NRC. The primary concern for the 1992 event was a rapid depressurization. The material was covered extensively in conjunction with training for a modification that is rerouting the reference leg piping in response to Generic Letter 84-23. The training included a lab exercise and a videotape review of the issue.

NRC Bulletin 93-03 deals with the effects of non-condensible gasses during a routine shutdown. The training provided covered events encountered within the industry as well as other actions required by this bulletin.

(b) By July 30, 1993, each licensees requested to complete augmented operator training on loss of RPV inventory scenarios during mode 3, including RPV draindown events and cracks or breaks in piping.

Dresden Station Training Department has developed an augmented training module including simulator training for all personnel with active RO and SRO licenses. This training provides all licensed operators with an understanding of the conditions and actions required during depressurizations of the RPV. The enhanced training module was completed prior to July 30, 1993.

All of the above short term actions will remain in effect until hardware modifications are installed at Dresden Station.

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

Quad Cities Station has established enhanced monitoring of the RPV level instruments during reactor shutdown activities when the effects of reference leg degassing have the potential to cause reactor level indication offsets. Prior to commencing planned shutdowns, a philosophy of HLA (heightened level of awareness) has been established which emphasizes the need of monitoring reactor level indications for abnormalities during reactor vessel depressurization and the significance in regard to activities with the potential to drain the reactor vessel. Procedure QCOP 201-12, Monitoring RPV Level Instrumentation During Normal Unit Depressurization, has been implemented to enable the Operating staff to determine when reference leg degassing has occurred on a level instrumentation channel. This procedure established the criteria for the identification of notching at a nine inch level increase for five minutes. The basis for this criteria is the large margin between the trip setpoint and the variable leg tap for the limiting event of concern (Shutdown Cooling System Isolation). This margin is approximately 68 inches, which is significantly larger than the worst case observed indication offsets (3" at Quad Cities, 8" at LaSalle). The other constraint on the criteria selection is that the criteria must be large enough to prevent unnecessarily declaring instruments inoperable. Such a declaration would be non-conservative during shutdown Cooling operation because it would necessitate manually isolating a successfully operating core cooling system. It directs the operators in the steps necessary to identify which level indicators are accurate, and how to determine adequate core cooling if all level instruments experience notching. It also provides steps to assure documentation, investigation, General Station Emergency Plan declaration if appropriate, and backfilling of the appropriate level instruments.

The philosophy of HLA of activities with the potential to drain the reactor vessel in conjunction with the enhanced monitoring ensures that abnormal reactor level conditions will be observed and proper corrective action taken.

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

Quad Cities Station has performed a review of Station procedures that have the potential to drain the vessel. The results of the review confirmed that administrative controls have been established to ensure that valve alignments and maintenance activities with the potential to drain the RPV are controlled. Significant Operating Experience Report (SOER) 87-2, Inadvertent Draining of Reactor Vessel to Suppression Pool at BWRs, addressed concerns with plant design, procedures, and training while discussing an inadvertent draindown of the vessel. Quad Cities Station's review of this SOER prompted many of these administrative controls. The following procedures currently contain administrative controls to warn the operator of the potential for draining the vessel.

Procedure	Actions
QCOP 1000-5, Shutdown Cooling Start-up and Operation	Limits valves to be used; valve position verifications; cautions concerning possible drain paths; actions to take if unexpected level drops are noted.
QCOP 1000-29, Shutdown Cooling Start-up From Outside the Control Room	Limits valves to be used; valve position verification, actions to take for unexpected level decrease.

QCOP 1000-6, Shutdown Cooling Shutdown	Valve position verifications; precautions concerning possible drain paths
QCOP 1000-13, Rejecting Water from the Reactor to the Condenser Using RHR Pumps when the RWCU System is Unavailable	Valve and interlock verifications, valve out-of-services; level monitoring precautions.
QCOP 201-9, Lowering Reactor Water Level Via Reactor Recirculation Loop Drains to not Less than 60" Vessel Level	Restricts valves to be use and level range of use; provides for monitoring of level.
QCOP 1000-23, Draining the Reactor Cavity, Vessel or Recirculation Loops to the Torus	Cautions concerning draining flow rates and level monitoring requirements.
QCOP 1000-24, Draining Reactor Cavity to the Torus	Pump availability requirements and interlock verifications.
QCOP 1000-9, Quarterly RHR Power Operated Valve Test	Valve interlocks; refers to previous drain down events; cautions the operator concerning off-normal valve line-ups.
QCOP 1000-11, RHR Motor Operated Valve Local Controller Test	Communications requirements; valve use limitations referring to possible drain paths.
QCOP 1000-7, Cold Shutdown RHR Valve Test	Limits valve use with cautions about possible drain paths.
QCGP 2-1, Normal Unit Shutdown	Cautions concerning availability of make-up systems prior to starting a reactor cooldown.
QCGP 2-3, Reactor Scram	Cautions concerning availability of make-up systems prior to starting a reactor cooldown.

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

Quad Cities Station provided a briefing to all operators. This briefing explained the issue, and detailed the new developments experienced since the previous training. The briefing also explained how to identify the problem, how to respond, and the possible impacts on operations, with particular emphasis on drain-down events while shutting down or shutdown.

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

Quad Cities Station Training Department has developed and presented an augmented training module for all operators with active RO and SRO licenses which discusses loss of RPV inventory during Mode 3. This training was completed by July 30, 1993.

Quad Cities Station believes that these short term compensatory actions provide for the safe operation of the plant during the interval until effective modifications can be developed and implemented to correct the degassing problem.

1. Short Term Compensatory Actions

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

LaSalle Station has established enhanced monitoring of the RPV level instruments during reactor shutdown activities when the effects of reference leg de-gassing have the potential to cause reactor level indication offsets. Prior to commencing planned shutdowns, a philosophy of Heightened Level of Awareness (HLA) has been established which emphasizes the need of monitoring reactor level indications for abnormalities during reactor vessel depressurization and the significance in regard to activities with the potential to drain the reactor vessel. Special Operating Procedure LLP-93-050, "Rx Water Level Notching Determination During Rx Vessel Depressurization", has been written and functionally verified during a recent scheduled unit shutdown, to enable the Operating staff to determine when reference leg degassing has occurred on a level instrumentation channel. This procedure monitors and compares the individual narrow range level indications against an established criteria (detected notching of greater than 6 inches for more than 5 minutes) which ensures that all design basis protective actions will be met for all associated instrumentation of the reference leg. The basis for this criteria is the large margin between the trip setpoint and the variable leg tap for the limiting event of concern (Shutdown Cooling System Isolation). This margin is approximately 22 inches, which is significantly larger than the worst case observed indication offsets (3" at Quad Cities, 8" at LaSalle). The other constraint on the criteria selection is that the criteria must be large enough to prevent unnecessarily declaring instruments inoperable. Such a declaration would be non-conservative during Shutdown Cooling operation because it would necessitate manually isolating a successfully operating core cooling system. Level channels which fail to meet the criteria are declared inoperable with the appropriate Technical Specification Action taken until operability of the channel is restored.

The philosophy of heightened awareness of activities with the potential to drain the reactor vessel in conjunction with the enhanced monitoring ensures that abnormal reactor level conditions will be observed and proper corrective action taken. The unit shutdown procedure LaSalle General Procedure LGP 2-1, has been revised to require initiation of the special monitoring and awareness actions as described above.

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

LaSalle Station has performed a review of Operating Procedures that are used prior to entering Mode 3 and while in Modes 3, 4 and 5 that have the potential to drain the RPV. The result of the review confirmed that administrative controls have been established to ensure that valve alignments and maintenance with the potential to drain the vessel are properly controlled.

Automatic interlocks are provided on the Residual Heat Removal (RHR) system to prevent the opening of the RHR Shutdown Cooling suction valve while the RHR Suppression Pool Suction Valve is open, as well as preventing the RHR Suppression Pool Suction Valve from opening when the other suction valve is open. The RHR Full Flow Test and Suppression Pool Spray Valves are administratively controlled in the closed position with the valve breakers open prior to the suction path from the reactor vessel being established.

Because the RHR system is safety related, these out-of service (OOS) administrative controls are second verified in addition to the normal OOS tags which are placed on the control room handswitch, the motor power supply breaker, and the valve handwheel. The location of this required OOS in the procedure sequence, and its resultant deenergization of the valve power supplies, ensures that the valves cannot be inadvertently operated from the control room. If the OOS on the local handwheel were to fail (i.e. the

handwheel OOS card were incorrectly hung a;nd incorrectly verified by the second operator or inappropriately removed to allow valve operation manually), adequate reactor level control will be assured by 1) Heightened Awareness of the Operators to the potential to drain the vessel, 2) PCIS and ECCS would initiate as required due to the adequate margin between the safety settings and the variable leg tap with reference to the maximum size observed sustained notch of about 8 inches, and 3) draindown due to manual operation of a valve would be slow with the initial valve opening.

Adequate administrative control was also confirmed for affected procedures related to the shutdown of RHR Shutdown Cooling as well as operation of the system from the Remote Shutdown Panel. Multiple level indications are available at the Remote Shutdown Panel due to the previous installation of analog trip units for ECCS and ATWS circuitry. These card units have narrow and wide range level meters in each division. Administrative procedures for removing and returning equipment to service was verified to adequately control the system configuration to ensure that inadvertent system draining is prevented which will then ensure that the potential to drain the vessel is also prevented. The station had previously addressed the concern of potential to drain the RPV and the actions from these previous concerns are still in effect. The review determined that no additional enhancements to the current procedures which are needed to ensure compliance.

The procedures provide the operator with sufficient guidance and direction on valve operation, interlocks, precautions, and controls to prevent inadvertent drain down of the RPV. As administrative protection from improper maintenance or work group activities (contractors), a licensed operations supervisor reviews all proposed work activities with the work crew leader. The Operations/Work Group interaction further reduces the probability of improper actions by assuring that the work group understands the location of equipment and limitations of activities to be performed.

A list of Operating and Administrative procedures reviewed includes:

LOP-RH-07,	Shutdown Cooling Start-up, Operation and Transfer
LOP-RH-08,	Shutdown Cooling System Shutdown
LOP-RX-08,	Start-up of Shutdown Cooling from the Remote Shutdown Panel
LAP-820-5,	Use of Station Mechanical and Electrical Checklists
LAP-900-4,	Equipment Out of Service Procedure
LAP-1600-2	Conduct of Operations
LEP-GM-102,	Limitorque Valve Operator Electrical Maintenance

LaSalle Station Maintenance Dept. Memo No. 23, Guidelines for Valve Stroking by Maintenance Personnel

Tech Spec Clarification 06-92, Operations with a Potential to Drain the Vessel

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

LaSalle Station Training Department has conducted classroom training on the de-gassing issue and the effects on indicated level. Included in this training module was an in-depth discussion on the principles of reference leg de-gassing and the effects on level instrumentation as well as anticipated corrective actions.

Other training included shift briefings, daily operating orders, and licensed required reading. The training was initially prompted by Generic Letter 92-04 "Resolution of the Issues Related to Reactor Vessel Water Level Instrumentation in BWR's Pursuant to 10 CFR 50.54(f)" in which the primary concern was rapid depressurization of the RPV during an accident. A lab exercise was developed on the LaSalle Simulator which stepped the operators through level indication anomalies during execution of Emergency Operating Procedures (EOP). These exercises served to validate and develop the EOP Committee's evaluation of de-gassing effects on EOP execution, as well as familiarize operators with these effects. All active licensed personnel were trained on this exercise during 1993 First Quarter Training.

The focus of this Bulletin is more concerned with the de-gassing that may occur during normal depressurization. The enhanced training that was given to the licensed operators contained training on the recognition of de-gassing during normal unit shutdown as well as the determination of level inaccuracies during transient events. As with the rapid blowdown case, a simulator lab and exercise was developed and implemented which demonstrated the conditions where loss of level instrumentation occurs during a unit shutdown with a reactor vessel draindown event in progress, resulting in failure of the required RHR Shutdown Cooling isolation. All personnel with active RO and SRO licenses have completed this training.

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

LaSalle Station Training Department has developed an augmented training module for Licensed Operators to sensitize the operators to the conditions and actions required during normal and transient depressurizations of the RPV. The enhanced training module was completed by July 30, 1993 for all Licensed Operators at LaSalle.

LaSalle Station will keep in effect all of the above Short Term Corrective Actions until the hardware modifications are completed.

Attachment A4
Summary of Compensatory Actions

The following summary of compensatory measures demonstrates Commonwealth Edison's commitment to maintaining the safety of its BWR plants until the hardware modifications are installed.

1. All short term compensatory measures which were implemented as requested by the Bulletin will remain in effect, including the detailed level instrument monitoring during plant shutdown evolutions.

2. Each site will perform enhanced level instrumentation performance monitoring, including power operation monitoring and trending, to allow early detection of instrumentation anomalies. During power operation, specific attention will be paid to upward trending channels that may indicate reference leg inventory drawdown due to piping, tubing, or valve leaks. Condensing pot make-up rate calculations have verified that the make-up capability of gas bound condensing pots is sufficiently low to allow effective detection of potential instrument rack leaks through instrument performance monitoring. The effectiveness of this monitoring is further confirmed by two specific cases where an extremely small leak was detected at LaSalle station through level instrument trending. Data from one of these events was presented in a meeting with the NRC staff on June 17, 1993.

3. For all sites, operator training will be conducted Quarterly, to re-enforce the compensatory actions and awareness / measures to prevent inadvertent RPV water inventory loss.

4. Prior to each planned unit shutdown, the Heightened Level of Awareness (HLA) program will be activated to review the required monitoring and controls.

5. Both units at LaSalle County Station will perform periodic backfill of all channels of the Post Accident Wide Range level indication system during power operation. The periodicity of this backfill is based on engineering and test data and data collection needs, and is currently perform approximately every 30 days. The intent of these backfills is to maintain the affected level instrument legs with an insignificant gas inventory, thereby minimizing the potential for de-gassing related indication anomalies. The backfill procedure methods and periodicity are reviewed and approved by the site On Site Review function. (During these backfills, system performance / interaction data to support the modification design development are also being collected. This affects the procedure methods and timing).

   For the LaSalle site only, the observed notching of the Division II reference leg ("B" channel) on both units will be precluded by manual backfill of that leg prior to reactor pressure dropping below 450 psig, during controlled shutdowns. The Operating Supervisor may waive this backfill if plant conditions warrant.

   Dresden and Quad Cities stations have evaluated the possibility of performing periodic backfills at their units. The LaSalle configuration is unique, providing an opportunity to backfill and collect data on reference legs which do not have any automatic actions associated with them. There are no similar configurations at Dresden or Quad Cities, making the backfill of a leg with automatic actions a risky evolution. It was decided by these sites (i.e., Dresden and Quad Cities) that the operations risk of performing the backfill while on line outweighed the short term benefit of performing the backfill.

I. CECo Actions Towards Design and Schedule Development

The Commonwealth Edison Company evaluation of the potential effects of Reactor Water Level (RWL) instrument leg gas inventory changes has determined that hardware modifications are desirable to ensure that RWL instrumentation system performance is maintained, with minimal reliance on manual actions or compensatory measures. These modifications are planned for each of the affected reactors.

The modification schedule in Section II of this attachment incorporates the results of CECo's modification development schedule that meets the following criteria:

1) Applicable licensing basis design standards (Site specific requirements for applicable ASME code revision, containment isolation, channel independence, etc).

2) Instrumentation system functional design specifications for reliability, system interface, surveillance capability, and channel independence.

3) Design development sequence must include component and system performance verification to minimize reliance on untested analytical assumptions that may affect compliance with the performance criteria of 1) and 2) above during postulated plant conditions. These conditions include expected and potential evolutions in interfacing systems and equipment.

4) Minimized design development and installation cycle time.

The original CECo effort in this design development was focused on a proposed "venting" design, due to its passive nature. Due to potential design complexities that arose during the design scoping efforts which partially negated the venting approach's inherent superiority, CECo determined that a continuous backfill system was necessary to meet criteria #4 above .

The backfill design approach introduces two new, fundamental effects to the existing RWL sensing leg design. These effects are 1) the conversion of the sensing legs from a passive, non-flowing system to a flowing system, and 2) the interconnection of active, moveable components inside the sensing leg boundary which can cause liquid volume displacements several times larger than the full scale displacement requirements of the existing level and pressure transducers. Because of these effects, the design development effort requires extensive use of system transient modeling and verification by plant and/or mock-up testing. A detailed depiction of the design cycle is given in Attachment D.

The design input and verification testing necessitates the following implementation schedule. This schedule utilizes a phased implementation approach to maximize the performance benefits of the modifications, while still minimizing the introduction of new risk factors associated with the fundamental changes in the system design. Justification and discussion of the schedule is given in Attachment C, as required by Reporting Requirement 2b of the Bulletin.

II. Modification Installation and Activation Schedule

The installation of a continuous backfill design will be completed during the first shutdown outage of greater than 20 days commencing after December 1, 1993, including the subsequent scheduled refueling outage for each unit, with the exception of Dresden Unit 3 and LaSalle Unit 2.

All reference legs which provide Safety System Actuation functions or are required for long term operator information under plant emergency conditions will be modified. This includes the normal level instrumentation ranges, but does not include extended range instrumentation legs such as the Refueling Zone reference leg(s). The design of the backfill systems will be developed to essentially eliminate non-condensible gas migration down the reference line from the condensing pot. For some shared leg geometries, it will not be necessary to backfill both legs because the common portion of the reference line will be kept purged, thereby making gas migration into the non-flowing legs impossible.

Although the hardware will be installed for backfill of the required cold leg reference lines, the activation of the modification(s) will be phased in as described below. In all cases, the Interim Compensatory Measures will remain in effect until the affected sensing legs have been modified as described.

Dresden 2, Quad Cities Units 1 and 2:

1. The unit(s) will start up with the reference legs for both channels of the Fuel Zone / Feedwater control system valved in. This refers to the first start up following backfill modification installation.

2. The activation of the remaining instrument legs will take place prior to or during the next unit start-up from Hot or Cold Shutdown conditions after 6 months experience has been accumulated with the new system.

Dresden 3:

The modification in response to GL 84-23 that will remove the Yarway columns from Unit 3 is scheduled for Refuel Outage # 13 (D3R13 beginning March 1994). Since the Yarway columns cannot accumulate significant non-condensible gas buildup, the installation of this mod on Unit 3 can be postponed until D3R13. Potential changes to the GL 84-23 modification are being evaluated by the station. These changes would be a result of technical issues raised by the non-condensible gas issue. Phasing of the modification activation will be consistent with Dresden 2 and Quad Cities 1 and 2.

LaSalle Units 1 and 2:

1. For LaSalle Unit 2: During the 5th refuel outage (L2RO5 beginning Sept 1993), the backfill modification will be installed. When the unit starts up, only the Post Accident Monitoring legs will be valved in. At the first shutdown after 6 months experience has been accumulated with the system, the reference leg for the Division 2 trip systems will be valved in. At the second shutdown after the same six months experience, the remaining reference legs will be valved in.

2. For LaSalle Unit 1: The backfill modification will be installed and phased in as described above. Given the expected refueling outage schedule, LaSalle Unit 1 would start up with the modification fully installed and activated (i.e., the 6 months experience from Unit 2 will be applied to Unit 1).

The phased approach will allow accumulation of experience with the reactor water level and backfill system interactions without exposing the plant systems to untested interconnection. The period of six months was chosen to envelope a maximum number of Control Rod Drive System periodic surveillances, during the performance of which it is expected that data will be collected. Most of the event responses are likely to show significant site specific behavior. Due to the schedule acceleration, and to ensure prudent contingency provisions, CECo plans to phase in the activation of the backfill modifications. The opportunity to collect event response data during the six month period following activation of the reference legs that are initially valved in ensures that site specific behavior can be analyzed and effectively used to alleviate possible problems.

Outage length of >20 days can refer to a refueling outage or a forced outage. The outage length of >20 days was chosen after each Construction group had assessed the modification and prepared a detailed installation schedule. It was determined that even with as much work as possible performed non-outage, the installation of the modification would take at least twenty days, unless the unit were in a refueling outage. With a unit defueled, there are fewer restrictions on out of services, enabling parallel path installation. With a unit in a forced outage (with fuel in the vessel), Shutdown Risk Management and Technical Specification Operability requirements necessitates that only one reference leg at a time can be taken out of service, forcing a pattern of consecutive installations, retests and operability determinations. The consecutive installations were found to average from 24 to 35 days for the three sites, thus, the additional duration beyond 20 days for any outage will be absorbed by the installations.

Attachment C
EVALUATION OF COMPENSATORY MEASURES
AND JUSTIFICATION FOR PROPOSED SCHEDULE

I. Evaluation of Compensatory Measures

Prior to the release of Bulletin 93-03, Commonwealth Edison Company was at an advanced stage in a program to evaluate the phenomenon of gas inventory effects on the CECo BWR reactor level instruments. LaSalle was being used as the lead site for data collection and evaluation. This effort was intended to:

1. Quantify the observed behavior with respect to instrument performance, plant conditions, and sensing leg geometry.

2. Provide input to the development of Compensatory Measures, Operability Evaluations, and Operator Training.

3. Complement CECo's use of the BWROG information which was securing phenomonological data for bounding case evaluations, by providing real plant performance data.

This program led to the development of Operability Evaluations and Compensatory Measures at all 3 CECo BWR sites. These Compensatory Measures were developed, validated, and implemented to achieve the maximum possible margin to plant safety and uncomplicated use of RWL indications by plant operators. The measures which have been implemented exceed the Compensatory Action requirements of the Bulletin and for the following reasons, ensure that the existing conditions do not constitute any significant change in plant safety margin from the initial plant design:

1. The completed Operability Evaluations confirm that needed automatic functions are secure and operable, thereby assuring that initial equipment actuations will take place as required by plant analyses.

2. The occurrence of de-gassing effects on the level instrumentation is strongly affected by the reference leg geometry. Carefully installed piping with continuous sloping sections has minimal capability to accumulate released gas into the large 'bubble' needed to cause significant level indication offsets during depressurization. Review of the CECo plants' instrument performance indicates that the newer piping installations achieve the observed minimal de-gassing effects largely because of the optimized pipe routing. This conclusion is supported by the data collected during shutdowns of Quad Cities Units 1 and 2, and the LaSalle Post Accident Monitoring System legs (all of which are newer installations). Similar performance is expected for Dresden Unit 2, which has newly installed cold leg piping. For Dresden Unit 3, the Safety Grade level indications are presently derived from self vented Yarway columns, which are unable to accumulate significant entrained gas inventories.

3. The LaSalle station Post Accident Wide Range level instrumentation system is kept immune from the subject degassing concern by periodic manual backfills. This ensures that the reactor operator will always have multiple, redundant indication channels which are <u>unaffected</u> by the de-gassing phenomenon and will therefore always maintain the ability to manually operate systems based on proper level indications.

4. CECo and available BWR industry data strongly indicate that significant (> 6") indication offsets occur below 100 psig, or not at all. Review of the spectrum of transient and accident event scenarios, and experience with their associated Emergency Operating Procedures (EOPs) verifies that the number of events which involve uncontrolled depressurization below 100 psig is very limited. Because of this, the reactor operator is essentially always in complete control of pressure changes during the conditions where de-gassing poses a significant potential to challenge the accuracy of the level indication systems. The operator is therefore able to closely associate manually initiated evolutions with reactor level changes, has minimum discovery time for unexpected level losses, and can quickly restore reactor level inventory and inventory controls.

   The small number of potential events is evident in the EOPs, which place a very high priority on achieving control of cooldown rates, as well as from industry experience. The dominant event which poses a potentially uncontrolled rapid cooldown (depressurization) is the stuck open SRV. CECo experience in these events has shown that reclosure is achieved before reactor pressure decreases below 350 psig.

   For de-gassing during manually controlled depressurization, the physical mechanisms of the phenomenon causes behavior which has proven repeatable, because the sensing leg geometry is the primary parameter for determination of notch size. Therefore, monitoring and comparison to "baseline" data is appropriate and effective in maintaining the operators ability to determine reactor water level.

5. Extensive CECo experience with RWL indications during controlled plant shutdowns supports the CECo observation that instrument channel deviations are minor in extent, and very unlikely to pose a significant challenge to system operation or operator information. In particular, data collected during shutdowns after extended unit operation (LaSalle 2: 193 days, LaSalle 1: 130 days, Quad Cities 1: 100 days, Dresden 3: 80 days) showed de-gassing effects that fell well within the limits for assurance of safety system operation.

6. The validated methodologies of enhanced level channel monitoring have been incorporated into plant procedures to ensure that actuation (and isolation) systems will perform within their analyzed limits.

7. System design interlocks and high integrity administrative controls on systems and components which affect the potential for RPV draindown events are in place. These controls have been reviewed and re-emphasized to plant operating personnel. Industry data indicates no known inadvertent draindown events from Mode 3. Mode 4 and 5 conditions cannot experience degassing because depressurization is complete prior to entry in either of these modes. Therefore, significant margin from challenges to the level indication systems exists while using the above mentioned compensatory measures.

8. Extensive Operator training on the degassing phenomenon has been completed, including both classroom instruction and simulator exercises. This ensures a thorough awareness by Operations personnel of the potential instrument indications and their impact on procedural actions.

For the above reasons, the risk associated with the interim use of compensatory measures is very small and is validated by the extensive industry experience with the existing reactor level instrumentation systems.

II. Justification for Proposed Schedule

Commonwealth Edison assembled a design modification team consisting of personnel from all 3 BWR sites and appropriate support personnel from the Architect Engineers (Bechtel and Sargent & Lundy) and the reactor vendor, General Electric.

This team has assembled a modification development plan and implementation sequence. The modification development plan was designed to achieve a technically adequate solution on a very aggressive schedule. The technical development of the modification is required by Commonwealth Edison procedures to ensure that the modification solves the subject problem without introducing new, unacceptable conditions or events. In particular, the procedure requires a detailed evaluation of system interactions for any modification.

For CRD/RWL system interactions, no specific interaction data or industry experience database exists which would allow completion of those interaction portions of the design checklists. In addition, specific information does exist in GE SIL-463, Rev. 1, and industry experience data related to the potential for inadvertent equipment actuations caused by hydraulic pulses in the RWL instrument piping. For these reasons, the modification plan required testing (both in plant, and with developed models), and analytical evaluations.

In a presentation to NRC on July 26, 1993, CECo presented the testing data obtained to date, which indicated that further testing and design efforts are necessary to achieve a design that does not introduce unacceptable equipment actuations or performance. This data showed the presence of significant acoustical waves in the CRD system that would be fed into the RWL instrumentation piping, and required carefully designed attenuation devices to assure acceptable response. The complete testing spectrum and design development is being pursued by Commonwealth Edison on an expedited basis.

The full design development schedule achieves all the required steps of the modification process in an efficient and expedited manner, and is therefore justified on the basis of the need for proper engineering development.

Of these milestones, the most limiting involve the hydraulic analysis of the system performance and system interactions. The system interaction determination must be performed as an iterative process in order to support the minimized design time. (Normally, a full spectrum of interaction events response data would be collected prior to initiating design activities). This testing and validation is necessary to ensure that uncertainties in the hydraulic interaction models do not allow unacceptable system performance during postulated events such as CRD pump trip/restart or full core scram. Plant test data for full core scram will not be available until the LaSalle Unit 2 shutdown for L2R05 in September 1993.

This schedule does not contain contingencies for complications which could arise in the technical analyses, material procurement, or licensing reviews, as would typically be the case.

The CECo commitment to a phased implementation schedule was generated to alleviate complications which can result from an expedited design development effort. System interaction considerations for such a significant change in level instrument sensing legs' service must include a full spectrum of events and event responses. To assure that actual operating conditions and procedures do not cause CRD system changes different than the testing program, the phased implementation program will accrue a limited amount of actual experience with the level instruments that do not cause trips, prior to activating the backfill for the trip systems.

The dates contained in that project plan are the currently scheduled best estimates, and the intermediate plan steps are not intended to be commitments but are provided to support the adequacy of the final projected modification availability of December 1, 1993. CECo's commitment to the installation schedule has been described earlier. Contained on the project plan is the current status information (i.e. completed steps are shown), which reflects CECo's efforts towards timely completion of the project, including system data collection and testing. CECo is committed to an expedited engineering effort for the realization of this modification and will make every effort to condense the schedule to facilitate an installation during LaSalle's Unit 2 fall 1993 outage. However, the staff must realize that any compression of the schedule, which has a target completion date of December 1, 1993, will require CECo to take some measure of calculated risk to accomplish an earlier completion date.

As an example of the potential system interaction effects, one preliminary interaction test of a single rod motion showed significant (20" RWL indicated) , level signal oscillations of a high frequency, short duration nature that required high speed data acquisition equipment to detect. This data showed transmitter responses that would be detected by the system trip cards, but not the installed panel meters. Further testing involving more CRD system events, alterations to the backfilling hardware and operation, and backfill injection flow rate changes for response characterization are scheduled.

II. Phased Implementation

The implementation of the backfill modifications will be completed by phasing the activation of installed backfill channels in two steps:

1)The first backfill system at each site will be activated only on the 'non-trip' legs. For Dresden and Quad Cities stations, this is the "Feedwater / Fuel Zone" channels. For LaSalle station, the non-trip legs are the Post Accident level monitors.

The purpose of this initial step is to allow experience and testing of the modification without undue risk of trips or unexpected problems that pose operability questions for the Safety Systems actuation channels.

ATTACHMENT D

Modification / Design Schedule

Milestone or Activity	Start Date	Completion Date
BWROG degas test	1Sept 92	1 April 93
CECo Station Operability Evaluations		1 June 93
Abandon (Passive) VENTING Design efforts		June 93
CRD System Interaction Testing		
Single rod notch moves	28 June 93	28 June 93
Rod SCRAM	17 July 93	21 July 93
MINIMUM BACKFLOW DETERMINATION	2 June93	26 July 93
THERMAL HYDRAULIC ANALYSIS	20 July 93	27 Aug 93
PRELIMINARY PIPE ROUTING	18 June 93	31 Aug 93
CRD SYSEM TESTS (remainder of system evolutions)	28 June 93	13 Sept 93
THERMAL STRESS ANALYSIS	24 Aug 93	17 Sept 93
MAXIMUM SYSTEM TRANSIENTS (BWROG testing to determine design limting events)	~15 Aug 93	10 Oct 93
CRD SYSTEM TRANSIENTS EVALUATION	26 July 93	10 Oct 93
50.59 / LICENSING ISSUES USQ determination, Tech Spec / LCO / OOS Times	1 Sept 92	7 Nov 93
APPROVED DESIGN ISSUED		15 Nov 93
COMPLETE INSTALLATION PKG (Package ready for Work Group installation)	15 Nov 93	1 Dec 93

This first period is a 6 month period for each site. Once the backfill system has been operating on the non-trip legs for 6 months for either unit at a site, the backfill systems for the installed 'trip' channels will be activated during the next unit shutdown of any duration. The 6 month period is intended to provided experience data during the full range of plant operations, including:

*CRD system evolutions (pump swaps, trips, filter changes, etc)
*Various plant conditions i.e., start-ups, shutdowns, surveillances, or possible plant trips with larger (scramming) rod density changes than are possible during testing, such as a scram from high power.

Six months was chosen to provide a reasonable interval during which most of the expected CRD system events, surveillances, and conditions are expected to be complete. Since the CRD system does not routinely experience a high frequency of major alignment changes, a shorter observation period would not be useful. The six month period allows ample time to collect experience data for plant surveillances such as semi-annual surveillances. Because of common unit configurations and procedures, the observation period is not necessary for both units at a given site.

2)Activation of 'Trip Channels'
Following the 6 month experience period, backfill for reference legs feeding 'trip' channels will be activated. Completion of this step marks the full implementation of the backfill modification, and use of the previously mentioned Compensatory Measures will be terminated.

For LaSalle station, this activation will be further broken up to activate the ECCS Division II ("B") leg first, with the other ECCS divisions being activated during the first outage after the Division II backfill is activated. This sequence is consistent with the established LaSalle practice of minimizing the potential for common mode or multiple channel effects with a single modification. Work at LaSalle station is normally planned under this practice, including RPS and ECCS system maintenance. The LaSalle installation of RWL analog trip systems was performed under this practice after historical experience with mechanical dP switches ("Static O Ring" RWL instruments) which demonstrated common cause setpoint drifting.

The phased implementation is justified on the basis that the channels of most value to the operators (wide range RWL) is protected from the subject concern as a first priority, while avoiding the potential to introduce common problems across redundant Safety System divisions and operator indications. Once one of the two units at a CECo BWR site have completed the 6 month experience interval, the other unit will not require that interval, and will be able to proceed to full implementation. Because the Compensatory Actions have been verified through plant experience to be highly effective, and will be kept in force until the necessary hardware changes are fully implemented, safe plant operation is assured in the interim period.

ATTACHMENT E

PROJECT OUTAGES
PLOT A
PAGE 1/1 SHEET 1/1
START 4/20/70
FINISH 11/16/98
DATA DATE 11/23/92

OUTAGE SCHEDULE TO SUPPORT GOALS
NUCLEAR STATIONS DIVISION G.O.
COMMONWEALTH EDISON

WORKING SCHEDULE

LINE BREAK ON

MODE C/FE
INTERVAL: 1 MONTH(S)

RUN 1/15/93 12.37
PROJECT/2 92A2
SCHEDULE BAR CHART

1993 1994 1995 1996 1997 1998

J F M A M J J A S O N D J F M A M J J A S O N D J F M A M J J A S O N D J F M A M J J A S O N D J F M A M J J A S O N D J F M A M J J A S O N D

D2R13 - 1/16 - 4/18 (13, 13, 0) LFPC 10/11
D2R14 9/10 -11/20 (10, 10, 0)
D2R15 2/17 -4/28 (10, 10, 0)
D2R16 - 9/6 -11/17 (10, 10, 0)

D3R13 - 1/15 - 4/8 (13, 12, 0) LFPC 6/23
D3R14 - 9/9 - 11/18 (10, 10, 0)
D3R15 - 1/11 - 3/22 (10, 10, 0)
D3R16 - 3/7 -5/18 (10, 10, 0)

D1R13 - 3/12 - 5/21 (10, 10, 0)
D1R14 10/14-12/24 (10, 10, 0)
D1R15 - 3/8 -5/15 (10, 10, 0)
D1R16 - 9/5 -11/16 (10, 10, 0)

D2R12 - 3/6 - 5/29 (12, 12, 0) LFPC 4/3
D2R13 9/24-12/4 (10, 10, 0)
D2R14 - 3/23 - 6/3 (10, 10, 0)
D2R15 - 9/6 -11/17 (10, 10, 0)

L1R05 - 10/3 - 12/30/93 (12, 12, 0) LFPC 7/2
L1R06 3/5 - 5/15 (10, 10, 0) 6/5 (13)
L1R07 9/30- 12/10 (10, 10, 0)
L1R08 - 3/8 -5/19 (10, 10, 0)
L1R09 - 9/5 -11/16 (10, 10, 0)

L2R05 - 09/4 - 11/21 (11, 11, 0) LFPC 7/2
L2R06 3/11-5/20 (10, 10, 0) (13)
L2R07 5/07-11/17 (10, 10, 0)
L2R08 - 3/7 -5/18 (10, 10, 0)

A1R13 - 10/25 - 4/04 (23, 10, 13) LFPC 12/19
A1R14 9/2 - 11/12/95 (10, 10, 0) LFPC 6/1
A1R15 - 1/11 - 3/22 (10, 10, 0)
A1R16 - 3/7 -5/11 (9, 9, 0)

A2R12 - 11/14 - 02/28/93 (14, 7, 8) LFPC 08/10
A2R13 - 10/11 - 03/14 (22, 12, 10)
A2R14 12/31 - 3/12/95 (10, 6, 10) LFPC 12/1
A2R15 - 9/1 - 11/10/96 (10, 10, 0)
A2R16 - 3/23 - 6/03 (12, 12, 0) LFPC 4/3
A2R17 - 9/5 -11/9 (9, 9, 0)

B1R05 - 02/09 - 04/11 LFPC 1/27
B1R06 3/18 - 11/13 (9, 9, 0)
B1R07 3/02-5/5 (9, 9, 0)
B1R08 - 9/6 -11/10 (9, 9, 0)

B2R03 - 09/04 - 11/07 LFPC 8/17
B2R05 2/11-4/16 (9, 9, 0)
B2R06 5/07-11/10 (9, 9, 0)
B2R07 - 3/7 -5/11 (9, 9, 0)

A1R04 2/26 - 5/1/94 (9, 9, 0)
A1R05 - 9/30 - 12/3/95 (9, 9, 0)
A1R06 - 3/8 -5/12 (9, 9, 0)
A1R07 9/5 -11/9 (9, 9, 0)

A2R03 3/6-5/9 (9, 9, 0)
A2R04 9/10 - 11/13 (9, 9, 0)
A2R05 3/02-5/05 (9, 9, 0)
A2R06 - 9/6 -11/10 (9, 9, 0)

NOTE: FOR INFORMATION OR COMMENTS
CONTACT - MR R. SPEAR X-6587
EXPLANATION OF ACTIVITY DESCRIPTIONS
OUTAGE I.D / START DATE - COMPLETE DATE
(DUR IN WEEKS TOTAL, 1ST YEAR, 2ND YEAR
LFPC - LOSS OF FULL POWER CAPACITY

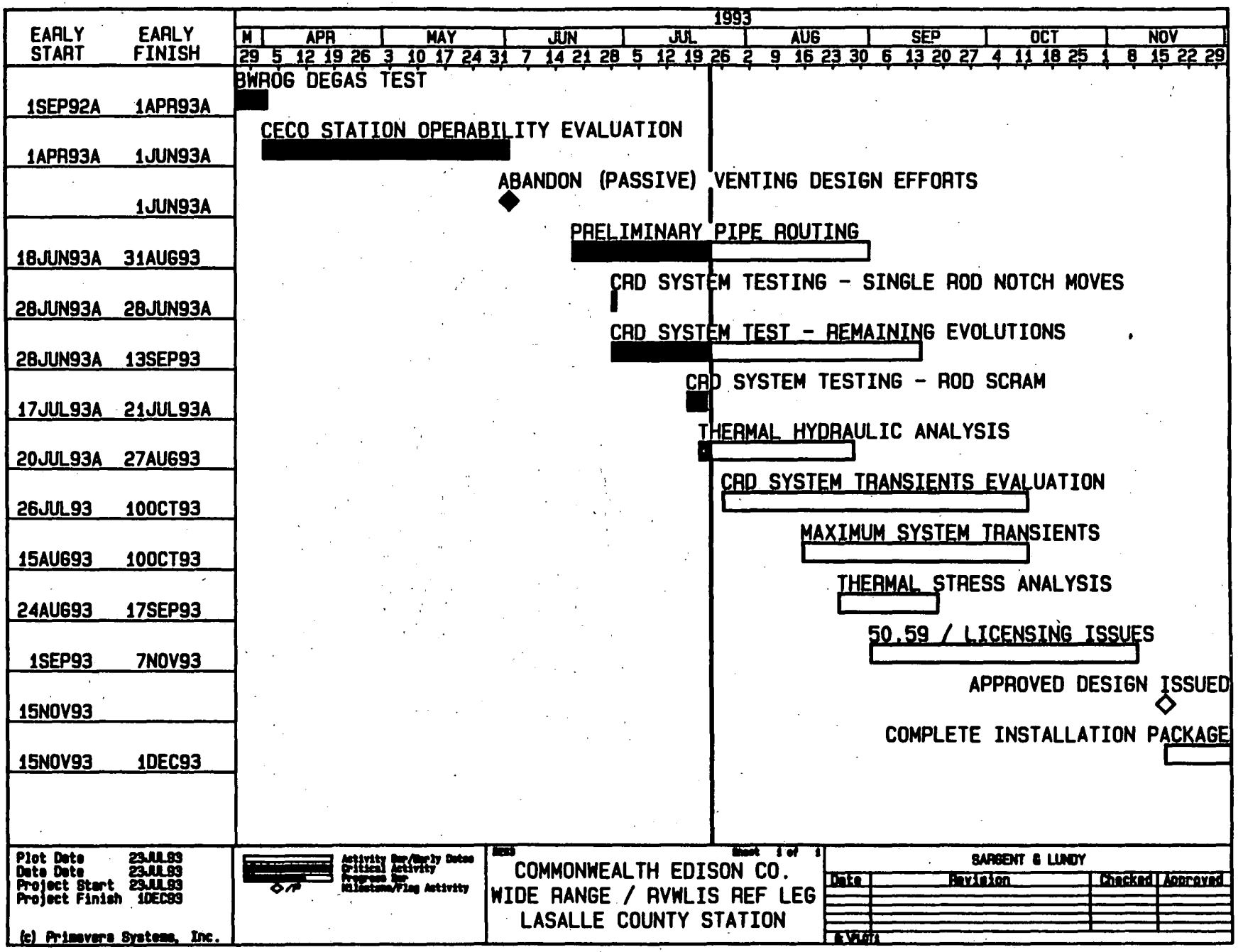
NOV 11, 1992 REV. #23

LEGEND
COASTDOWN
OUTAGE

J F M A M J J A S O N D J F M A M J J A S O N D J F M A M J J A S O N D J F M A M J J A S O N D J F M A M J J A S O N D J F M A M J J A S O N D

1993 1994 1995 1996 1997 1998

ATTACHMENT F
MODIFICATION TEAM PROJECT PLAN



Purpose:

Commonwealth Edison met with NRC personnel on July 26, 1993 to present information related to the development of the reference leg backfill modifications for the six CECo BWR units. At this meeting, information was presented to NRC which showed data collected during tests performed at LaSalle station during routine backfills of the LaSalle Post-Accident Level Monitoring instrumentation.

Information Presented:

The LaSalle Post Accident Monitoring indication system has no trips or safety actuations derived from the level instrument piping. The LaSalle station operability evaluation performed for the reference leg degassing issue prescribed that routine backfills of these reference legs be done to maintain the legs with a minimal probability of containing sufficient quantities of non-condensible gas, so that significant level instrument indication errors are not possible during depressurization.

The basis for the backfills is that it is possible to establish an interval that ensures minimal gas entrainment because the primary mechanism for gas migration down the reference leg (leakage at the instrument rack) is detectable, and of a low probability. The subject legs each have only 2 instruments connected to them. The condensing pots are gas bound and capable of approximately .02 #/hr makeup capability(most accurate, diffusion limited calculation) , and have a volume of approximately 8 gallons (64 lbm). For the worst case (undetectable) leak, this gives a complete leg inventory turnover time of 130 days. The detectability of rack leakage is supported by recorded experience at LaSalle (data presented in June 17, 1993 meeting with NRC).

During the backfills, CRD system evolution data is collected according to a schedule assembled by the plant system engineer.

CRD Data Results:

Backfill testing results presented were shown to involve very short duration (200 msec pulse-width at half maximum) spikes, caused by acoustic waves entering the backfill hardware from the CRD system. These data are summarized below:

Flow Rate	**Flow Source**	**CRD System Evolution**	**Spike (p-p)**
0.2 gpm	Drive Water	Single Rod Notch	30"
.01 gpm	Drive Water	Single Rod Notch	18"
.01 gpm	Drive Water	Single Rod SCRAM	14"
.01 gpm	Charging Water	Single Rod SCRAM	15"
.01 gpm	Drive Water	Continuous Rod W/D	23"

Commonwealth Edison will utilize the observed response data to complete a preliminary design and performance test for an acoustic dampener. This dampener will be tested prior to or during the shutdown of LaSalle Unit 2 for Refuel Outage #5.

Discussions of the test methodology, results, and implications were shared. The CECo conclusions from the testing to date were that the spectrum of CRD system events were needed to assure that adequate design safeguards were built in.

Also presented was the design development schedule. This schedule and it's components were discussed.

NRC suggested that bounding assumptions could be made to allow the design development and minimize delay in the sequence. This suggestion is being pursued by CECo, along with other efforts to minimize the testing / development / installation cycle.

An installation and phased implementation schedule similar to the commitment in Attachment B of this Bulletin response was discussed, including the impact of installation during a forced (non-refuel) outage where the reactor is not defueled. Because installation during this condition requires 25 to 35 days to complete, the CECo commitment to perform the modification during any Cold Shutdown of greater than 20 days represents a commitment to good faith effort.

Commonwealth Edison agreed to meet with NRC within approximately 1 month to update on the status of testing and design development efforts.