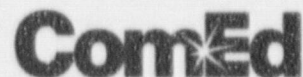


Commonwealth Edison Company  
LaSalle Generating Station  
2601 North 21st Road  
Marseilles, IL 61341-9757  
Tel 815-357-6761



February 20, 1998

**United States Nuclear Regulatory Commission**  
**Attention: Document Control Desk**  
**Washington, D.C. 20555**

Licensee Event Report #97-031-02 Docket #050-373 is being submitted to your office to update the safety analysis.

If there are any questions or comments concerning this letter, please refer them to Perry Barnes, Regulatory Assurance Manager, at (815) 357-6761, extension 2383.

Respectfully,

Fred Dacimo  
Site Vice President  
LaSalle County Station

Enclosure

cc: A. B. Beach, NRC Region III Administrator  
M. P. Huber, NRC Senior Resident Inspector - LaSalle  
C. H. Mathews, IDNS Resident Inspector - LaSalle  
F. Niziolek, IDNS Senior Reactor Analyst  
INPO - Records Center

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**LICENSEE EVENT REPORT (LER)**

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (MNBB 7714), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

**FACILITY NAME (1):** LaSalle County Station Unit One  
**DOCKET NUMBER (2):** 05000373  
**PAGE (3):** 1 of 12

**TITLE (4):** Leak Detection Area Temperature Calculation Errors Result in Plant Operation Outside Design Basis and Technical Specifications Due to Inadequate Technical Review

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
08	22	97	97	031	02	02	20	98	LaSalle County Station Unit Two	05000374
									FACILITY NAME	DOCKET NUMBER

**OPERATING MODE (9):** 4  
**POWER LEVEL (10):** 00C  
**THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §:** (Check one or more) (11)

<input type="checkbox"/>	20.2201(b)	<input type="checkbox"/>	20.2203(a)(3)(i)	<input type="checkbox"/>	50.73(a)(2)(iii)	<input type="checkbox"/>	73.71(b)
<input type="checkbox"/>	20.2203(a)(1)	<input type="checkbox"/>	20.2003(a)(3)(ii)	<input type="checkbox"/>	50.73(a)(2)(iv)	<input type="checkbox"/>	73.71(c)
<input type="checkbox"/>	20.2203(a)(2)(i)	<input type="checkbox"/>	20.2003(a)(4)	<input type="checkbox"/>	50.73(a)(2)(v)	<input type="checkbox"/>	OTHER
<input type="checkbox"/>	20.2203(a)(2)(ii)	<input type="checkbox"/>	50.36(c)(1)	<input type="checkbox"/>	50.73(a)(2)(vii)	(Specify in Abstract below and in Text. NRC Form 366A)	
<input type="checkbox"/>	20.2203(a)(2)(iii)	<input type="checkbox"/>	50.36(c)(2)	<input type="checkbox"/>	50.73(a)(2)(viii)(A)		
<input type="checkbox"/>	20.2203(a)(2)(iv)	<input checked="" type="checkbox"/>	50.73(a)(2)(i)	<input type="checkbox"/>	50.73(a)(2)(viii)(B)		
<input type="checkbox"/>	20.2003(a)(2)(iv)	<input checked="" type="checkbox"/>	50.73(a)(2)(ii)	<input type="checkbox"/>	50.73(a)(2)(x)		

**LICENSEE CONTACT FOR THIS LER (12)**  
**NAME:** Gerald Zwarich, Design Engineer  
**TELEPHONE NUMBER (Include Area Code):** (815) 357-6761 Extension 3034

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

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

**SUPPLEMENTAL REPORT EXPECTED (14)**  
 YES (If yes, complete EXPECTED SUBMISSION DATE)  
 NO  
**EXPECTED SUBMISSION DATE (15)**

MONTH	DAY	YEAR

ABSTRACT (Limit to 1400 spaces, i.e., approximately fifteen single-space typewritten lines 16)

On August 22, 1997, the LaSalle Engineering Department determined that calculations that form the analytical basis for the Technical Specifications (TS) leak detection area temperature and differential temperature isolation setpoints used a steam flash fraction that was not limiting in all cases. This event is reportable per 10 CFR 50.73(a)(2)(i)(B) as a condition prohibited by TS and 50.73(a)(2)(ii) as a condition outside the design basis. The investigation is complete. The causes were inadequate technical review of calculations, inadequate program monitoring and management deficiency following identification of inconsistency with the FSAR/UFSAR commitment, and miscommunication between the preparer of a calculation and the writer of an Operability Evaluation. Had design basis leakage occurred in the Residual Heat Removal shutdown cooling mode, Reactor Core Isolation Cooling piping, or portions of the Reactor Water Cooler Heat Exchanger Room piping, the automatic isolation as described in the TS may not have occurred. Principal corrective actions are to revise the analytical limit calculations, the Technical Specifications, and the UFSAR.



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**PLANT AND SYSTEM IDENTIFICATION**

General Electric - Boiling Water Reactor

Energy Industry Identification System (EIIS) codes are identified in the text as [XX].

**A. CONDITION PRIOR TO EVENT**

Unit(s): 1/2	Event Date: 08/22/97	Event Time: 0852 Hours
Reactor Mode(s): 4/N	Power Level(s): 0%/0%	RCS [AB] Temperature:
		Unit 1 < 200°F
		Unit 2 < 140°F
		RCS [AB] Pressure:
		Unit 1 0 psig
		Unit 2 0 psig

Mode(s) Name: Cold  
Shutdown/Defueled

**B. DESCRIPTION OF EVENT**

On August 1, 1997, during a meeting for the design change for Reactor Water Cleanup (RWCU, RT) [CE] pump and pipe replacement, preliminary data was presented for temperature and differential temperature leak detection isolation setpoints in the RWCU heat exchanger room and pump room. The data indicated that the Technical Specification (TS) setpoints may require revision. On August 22, 1997, the LaSalle Engineering Department determined that calculations that form the analytical basis for the leak detection area temperature and differential temperature isolation safety setpoints used a steam flash fraction that was not limiting in all cases.

The Leak Detection System (E31, LD) [IJ] was declared inoperable on August 22, 1997, and entered into the Station's Degraded Equipment Log. An ENS notification was made on August 22, 1997.

There were no components inoperable that contributed to this event.

The investigation is complete. Safety Analysis for pipe leaks in Residual Heat Removal Shutdown Cooling MoCs is complete. Safety analysis for leaks in the Reactor Core Isolation Cooling and the Reactor Water Cleanup Systems is provided in this supplemental LER. A summary of the previously identified issues is as follows:

During a review performed on January 26, 1996, Engineering identified that the 65 percent steam flash fraction used in the original leak detection temperature and differential temperature analytical basis calculations appeared incorrect. In March 1996, an assessment of the leak detection calculations was completed. This assessment found inconsistencies in the methodology used to predict the

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temperatures in rooms where temperature-based leak detection is present. A 65 percent flash fraction was used in some of the original calculations. The assessment concluded that a calculation error was made while converting a water leak at reactor pressure, to steam at atmospheric conditions. The correct fractions are 100 percent steam for steam lines and approximately 35 percent for water at normal reactor conditions.

On May 23, 1996, LaSalle Engineering completed an Operability Evaluation in response to the issue. This Operability Evaluation incorrectly stated that the calculation errors were corrected, and the new values were compared to the affected Technical Specification setpoints. The Operability Evaluation concluded that in all cases, the calculated values were equal to or greater than the Technical Specification setpoints; therefore, no operability concern or further action was required. The facts that the design basis calculations were not revised and still contained errors, and that the Operability Evaluation was erroneously closed was not discovered until the August 1, 1997 review of the RWCU modification.

Residual Heat Removal System (RHR, RH) [BO]

The architect engineer's original calculations for leak detection by temperature measurements have been reviewed. Following the August, 1997 review, a new RHR corrected calculation was developed for the RHR equipment areas evaluating both RHR Steam Condensing Mode and RHR Shutdown Cooling Mode. This calculation concluded that the Technical Specifications setpoints are bounded by the original results in regard to high temperature and delta temperature isolation setpoints except for RHR Shutdown Cooling Mode. The original calculations were based on the normal ambient area temperatures and did not differentiate between winter or summer conditions. It appears the original isolation setpoints for both RHR Shutdown Cooling Mode and RHR Steam Condensing Mode were based on the RHR Steam Condensing Mode of operation. This RHR mode of operation was in the original design basis for the plant but was subsequently removed in April 1993.

Original calculations indicate that a 25 gpm leak occurring while operating in the RHR Shutdown Cooling Mode would result in room temperatures of 180 degrees Fahrenheit for "A" RHR Room and 154 degrees Fahrenheit for "B" and "C" RHR Room. Because these temperatures are below the Technical Specification setpoint of 200 degrees Fahrenheit (for the RHR Shutdown Cooling Mode), an automatic isolation would not occur. The design basis calculation performed has determined that there is negligible temperature rise in the piping or equipment areas during a leak. This is due to the relatively low temperature of the RHR fluid when in the shutdown cooling mode while in hot shutdown (Operational Condition 3). A license amendment to delete the high temperature and differential temperature isolation function for the RHR equipment has been submitted for NRC approval.



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Reactor Water Cleanup (RWCU, RT) [CE] System

Under normal conditions, four different process flow temperature conditions exist in the RWCU Heat Exchanger Room (See Attachment 1). Stream (1): The process stream entering the regenerative heat exchanger is nearly reactor coolant temperature (maximum 550 degrees Fahrenheit). Stream (2): The process stream exiting the regenerative heat exchanger to the non-regenerative heat exchanger has been cooled to approximately 230 degrees Fahrenheit by the process stream returning from the demineralizers before entering the non-regenerative heat exchanger. Stream (3): The outlet of the non-regenerative heat exchanger is cooled to approximately 120 degrees Fahrenheit which then flows to the demineralizers before returning to cool the process stream (1) flow. Stream (4): The return to the feedwater system from the regenerative heat exchanger, which has been heated to approximately 435 degrees Fahrenheit.

A General Electric Design Specification states leakage from high temperature piping of the RWCU system equipment areas shall be detected by temperature sensing elements. In the design basis calculations, this was process stream (1). Since the temperature in process stream (4) is lower than stream (1), leaks less than 49 gpm (winter conditions) would not have isolated on high temperature. A leak from process streams (2) or (3) would not be detected.

The Technical Specification allowable values are 187 degrees Fahrenheit for the high temperature isolation and 91 degrees Fahrenheit for the high differential temperature isolation. The new design basis calculation establishes the setpoint for a 25 gpm leak from process stream (4) under summer conditions. The calculation indicates that the analytical basis is approximately 160 degrees Fahrenheit for the high temperature isolation and 42 degrees Fahrenheit for the differential temperature isolation. This results in a high temperature isolation for a 36.4 gpm leak or a 17 gpm leak on differential temperature during the winter. The new Technical Specification values will need to be lower than the current setpoint, which was based on process stream (1) leak conditions.

The original General Electric design specification establishes the temperature based leak detection limits on a simple heat balance of the equipment area. The GE design specification also stated that the differential temperature sensors shall be located in the inlet and outlet ventilation ducts.

The existing location of the differential temperature detectors may not have been effective in sensing the design basis 25 gpm leak. Due to the fact that both detectors are located in the same space, the differential temperature sensors may concurrently indicate rising temperature and consequently not actuate the isolation logic. A new design basis model which accounts for the inlet ventilation reduction due to steam leaks has been developed. Consequently, the inlet ventilation differential temperature detectors are being relocated to outside the room to more accurately reflect this model. The areas affected are the RWCU heat exchanger rooms and the RCIC pipe tunnel.

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Reactor Core Isolation Cooling (RCIC) [BN]

The differential temperature in the RCIC/Low Pressure Core Spray (LPCS) cubicle is sensed across the cubicle cooler system. The RCIC/LPCS cubicle cooler has a dampening effect on the differential temperature measurement and therefore for the design basis 25 gpm leak, the differential temperature isolation logic would not be effective. The differential temperature monitors are being switched to the Reactor Building ventilation system with the inlet ventilation differential temperature detectors being relocated to outside the room.

The original design basis and new design basis models conclude that effective isolation on high temperature, regardless of the ambient conditions, occurs at 25 gpm leakage.

Reportability

This condition is reportable per 10 CFR 50.73 (a)(2)(i)(B) as a condition prohibited by the plant's Technical Specifications because the high temperature and differential temperature TS allowable setpoints for the RHR Shutdown Cooling Mode, Reactor Core Isolation Cooling, and Reactor Water Cleanup System isolation would not have resulted in an automatic isolation at 25 gpm. This condition is reportable per 10 CFR 50.73(a)(2)(ii) because the plant operated outside the design basis.

C. CAUSE OF EVENT

The root cause investigation is complete. Inappropriate actions were identified during the preparation of the initial calculations performed in 1981 and 1984, the review and follow-up actions regarding General Electric Letter PRC 88-17 in 1989, and the preparation of an operability evaluation in 1996. Each of these errors is described below. The causes were inadequate technical review of calculations, inadequate program monitoring and management deficiency following identification of inconsistency with the FSAR/UFSAR commitment, and miscommunication between the preparer of a preliminary calculation and the writer of an Operability Evaluation.

Technical errors in eight calculations occurred in 1981 and 1984. These temperature setpoint calculations established the values for the leak detection for Main Steam [SB] Tunnel, Residual Heat Removal [BO] (RHR) pump rooms, Reactor Core Isolation Cooling [BN] (RCIC) pipe tunnel and equipment rooms, and RWCU [CE] (RWCU) heat exchanger cubicles which contain primary coolant outside containment. The investigation concluded that due to the age of the calculations, the true cause cannot be determined but the apparent cause is human performance errors by the preparers and reviewers. A contributing cause was that the work was performed by the architect engineering firm's Heating-Ventilation and Air Conditioning (HVAC) Group which was not as familiar with the type of calculations as the Mechanical Analytical Group. Although the preparers and reviewers had several years engineering experience, it appears that the most probable cause for



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how steam flash fraction errors occurred was that a "rule of thumb" was incorrectly used. Seven of the eight calculations utilized an assumption that 65 percent of the fluid would flash to steam. This may or may not have been determined using the steam tables, but the correct value for the liquid lines was the complement (approximately 35 percent). This error was carried forward to other calculations, even to the calculations dealing with steam lines instead of liquid. For the steam lines, the incorrect values provided conservative results in the calculations for main steam tunnel and RCIC steam tunnel. For the liquid lines, the results are nonconservative for the RWCU heat exchanger cubicles. Calculations for the RCIC equipment room and RHR equipment rooms were performed using correct flashing percentages in 1981 by the Mechanical Analytical Group since room cooler operation was involved. The architect engineering firm revised the calculation procedures: GC-3.08, Revision 6 (May 24, 1991), "Design Calculations", and GES-320.10, Revision 3 (September 16, 1997), "Preparation, Review and Approval of Design Calculation," to improve the quality of the review such as by documentation of method of review, verification of design input, and extent of verification.

High temperature and differential temperature setpoint calculations were performed for the RHR Steam Condensing Mode to verify that the temperature setpoints would fulfill General Electric Design Specification 27A2870. This specification states that isolation should occur on a detected leak of 25 gpm. The same temperature instruments are used for both the shutdown cooling and steam condensing isolation logic and the same setpoints were used in the Technical Specifications. Calculations were performed in 1981 that indicated that the high temperature and differential temperature instrumentation were not effective for the RHR Shutdown Cooling Mode. The reasons why the same values for the high temperature and differential temperature isolation setpoints were used for the shutdown cooling mode and steam condensing mode in the Technical Specifications is not known. It is postulated that since the leak detection isolation for shutdown cooling mode was a General Electric design and was part of BWR-5 Standard Technical Specifications (NUREG-0123), it was included in the LaSalle Technical Specifications although no calculation or analysis basis existed.

The root cause for the errors in the RWCU calculation is not known. The General Electric Design Specification states that high temperature process leaks shall be detected. This statement was interpreted in 1981 to mean that the highest temperature process stream [RWCU process stream (1) noted above] should be the basis for the leak detection heat balance. The justification for this interpretation is not documented. The new design basis considers the lower high temperature process stream (stream 4) as the source of the leak and establishes the setpoints accordingly.

The root cause for the errors in the RCIC calculation is not known. The effect of RCIC room cooler on differential temperature measuring does not appear to have been adequately understood at the design basis stage in 1981.

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In 1985, the effectiveness of the RCIC room cooler differential temperature was documented in a calculation. A recommendation that Reactor Building ventilation [VA] (VR) should be used for differential temperature was made but not implemented. The reason why it was not implemented is not documented or known. The new design basis will relocate the differential temperature sensors such that they will detect leaks with Reactor Building Ventilation system operating.

An opportunity to identify the inaccurate temperature setpoints occurred during the review of a May 1989 General Electric Potentially Reportable Condition (PRC) 88-17 letter, Main Steam Tunnel Temperature Instrumentation and Isolation. This letter was promptly assigned to site engineering. The scope of the recommendations was 1) to review the reactor coolant pressure boundary piping leak detection/isolation instrumentation to confirm consistent application of the Leak Detection System design intent and, 2) that the high energy line break analysis assumptions be reviewed for installed Leak Detection System capability. A review completed in May 1990 (by a consulting engineering firm) identified that several areas outside the containment through which hot reactor coolant piping is routed do not have temperature based leak detection systems. The results of this study were reviewed by the architect engineering firm for validation in January 1991 and recommended making enhancements in the leak detection instrumentation and clarification in the UFSAR. However, these recommendations were not implemented due to the low priority of the work.

In April 1995, a re-evaluation of the leak detection system and temperature monitoring system commenced. In September 1995, the station concluded that no operability issues existed and no immediate corrective actions were required. The engineering organization had concluded that clarifications to the UFSAR were warranted but these differences did not constitute operation outside the design bases. The cause of these delays was due to an inadequate program monitoring and management deficiency.

Although the station had not made use of the RHR steam condensing mode, with the review of Generic Letter 89-10 Supplement 3, "Consideration Of The Results Of NRC-Sponsored Tests Of Motor-Operated Valves," the station decided to evaluate removing that RHR steam condensing mode function from the plant design. Site Engineering and the architect engineering firm performed a safety evaluation and concluded that it was acceptable to remove the steam condensing mode. This mode of operation was removed in April 1993 with the approval of this change in Revision 9 of the UFSAR. Plant procedures were appropriately changed. Although the Technical Specifications referenced the RHR steam condensing mode isolation function and listed high temperature, delta temperature setpoints and allowable values, these changes were not pursued because it was considered to be low priority. Having the function out of service did not affect plant operation because the action statement for inoperable RHR steam condensing mode was met by isolating the function. The instrumentation could not be physically removed because these temperature instruments are also input to the RHR shutdown cooling isolation logic. Therefore, the calibration surveillance was maintained.



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The cause for inaccurately performing the Operability Evaluation that concluded the leak detection system was operable in May, 1996, was a miscommunication of information between the site engineering person who prepared the Operability Evaluation and the preparer of the calculation. The Operability Evaluation preparer had understood that the calculations, which corrected the steam fraction errors in the 1981 through 1984 leak detection calculations, had preliminary approval by the architect engineering firm and were therefore available for use as input to the Operability Evaluation. Since the Operability Evaluation stated that the corrected calculated value was equal to or greater than the Technical Specification setpoints, the required trip would occur as designed, the station's reviewers accepted this statement without verifying the referenced document had been approved for use. The architect engineering firm did not have a documented review. The preliminary calculation identified which referenced calculations should be reworked to validate or revise the current temperature setpoints. Work on the formal calculations was not started pending instructions from the site engineering organization regarding the commitments made in the Operability Evaluation because no additional corrective action was listed.

**D. SAFETY ANALYSIS**

Leak detection calculations have been revised to eliminate the identified errors and show that high temperature isolation would have occurred at design conditions for the established leakage rate of 25 gpm for all functions except for RHR shutdown cooling and some piping sections in the RWCU heat exchanger rooms.

The Leakage Detection System Technical Specification requirements for automatic isolation for RHR shutdown cooling are required during Operational Conditions 1, 2, and 3. These requirements were not met. The RHR shutdown cooling mode is treated as a moderate energy system due to the small percentage of the time that the system operates as a high energy system above 200 degrees Fahrenheit. Therefore, a break in it is not within the design basis of the plant and is not required to be analyzed. If a leak were to occur, little if any water will flash to steam and therefore leakage would not have been detected by the temperature and delta temperature instrumentation. Since there were no leaks in which the high temperature and high differential temperature leak detection was required to mitigate an accident, there are no safety consequences for this event. If there had been a leak in the RHR Rooms during normal plant operating conditions, there are alternate means of detection provided by the sump level equipment and area radiation monitors. Line breaks would be detected by RHR shutdown cooling mode reactor vessel water level low isolation or RHR pump suction high flow isolation. Offsite radiation dose for this postulated leak are bounded by the Main Steam Line Break accident scenario.

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For RWCU a leakage rate higher than 25 gpm may be necessary in some instances to actuate isolation, because the original design basis calculations established the temperature analytical limits by considering the possibility of leakage from the highest process temperature (546 degrees Fahrenheit) lines in the room. Leakage from the lower temperature process return line (437 degrees Fahrenheit) would have to be on the order of 45 gpm to actuate the ambient temperature isolation, or 48 gpm to actuate the isolation on high differential temperature. The consequences of leaks of these magnitudes are bounded by the main steamline/feedwater break discussed in the UFSAR Chapter 15 "Accident Analysis". The 45 and 48 gpm leakage rates are also below the leakage expected from a crack of critical dimensions as shown in UFSAR Figure 5.2-11. Sump alarms from floor drains in the RWCU Heat Exchanger and Valve Rooms would actuate in approximately 30-45 minutes from leaks of these magnitudes. This alarm would result in operator investigation and action to isolate the leak. Additionally, there is no safety-related equipment in this area except the leak detection temperature sensors. These sensors are qualified for a steam environment; therefore, there is minimal safety significance from the possible larger leakage rates that would have been necessary to actuate isolation as a result of only utilizing the highest process temperature when determining analytical temperature limits in this area.

For RCIC the existing design basis analysis does not indicate at what leakage rate the differential temperature detection/isolation would actuate. The room coolers operate whenever the associated Emergency Core Cooling Systems (ECCS) equipment is operating. This would typically only be during a postulated accident requiring ECCS actuation, or during planned surveillance tests. The analysis shows that the ambient temperature leak detection monitor will effectively function to detect and actuate isolation at the established leakage rate limit (25 gpm). Additionally, other leak detection methods, such as radiation monitoring and sump monitoring, will also function to detect a leak. Therefore, the ineffectiveness of the differential leak detection monitors is assessed to have minimal safety significance.

**E. CORRECTIVE ACTIONS**

1. Prior to restart of the respective Unit, analytical limit and setpoint calculations will be completed which document the areas which require temperature-based leak detection monitoring and the reason other areas do not require monitoring, or other approved leak detection methods. This action addresses the recommendations in the GE PRC 88-17 document. (NTS #373-180-97-SCAQ00031.01)
2. Prior to restart of the respective Unit, a revision to the applicable leak detection Technical Specification will be approved. (NTS #373-180-97-SCAQ00031.02)
3. Modifications to the leak detection equipment will be completed prior to restart the respective Unit. (NTS #373-180-97-SCAQ00031S101 and 02)



**LICENSEE EVENT REPORT (LER)**  
TEXT CONTINUATION

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (MNBB 7714), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			PAGE (3)
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4. The safety analysis of leaks in the RCIC and RWCU systems is provided in this supplemental LER. (NTS #373-180-97-SCAQ00031S103)
5. The applicable sections of the leak detection system described in the UFSAR will also be revised contingent upon approval of the Technical Specification change. (NTS #373-180-97-SCAQ00031S104)
6. The architect engineering (AE) firm now uses ComEd NEP-12-02, "Preparation, Review and Approval of Calculations" rather than the AE procedures when performing calculations on ComEd projects. An assessment to verify the AE's compliance with the NEP will be performed. (NTS #373-180-97-SCAQ00031S105)
7. A sample of other LaSalle County Station calculations performed by the same architect engineering KAC preparers and reviewers has been reviewed. Additionally, a sample of non-traditional calculations performed by the architect engineering firm's HVAC group was reviewed and no errors were found in either sample. (NTS #373-180-97-SCAQ00031S106)
8. The preparer of the Operability Evaluation was counseled on the inappropriate action resulting in the human performance error. (NTS #373-201-97-CAQD01651.02)
9. As a result of self-assessments performed by Corporate Engineering, revision 5 of ComEd NEP-12-02, "Preparation, Review and Approval of Calculations," was revised (June 30, 1997) to require ComEd review of calculations by outside organizations before use in quality-related documents.
10. Major changes to the management and implementation of the Corrective Action Program were proceduralized in May 1997. Assignment of personnel to the Corrective Action program to monitor and administer the Corrective Action Program has been completed. New procedures for the identification, evaluation, tracking and review of vendor (Operating Experience information) generated concerns and subsequent corrective actions are considered to be adequate to preclude recurrence of the type of delay that occurred with implementing the recommendations from General Electric letter 88-17.
11. A sample of Operability Evaluations where a calculation was referenced to determine operability will be reviewed to determine whether information prepared was in accordance with either LaSalle's or the vendor's quality assurance program. (NTS #373-180-97-SCAQ00031S107)

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**F. PREVIOUS OCCURRENCES**

LER NUMBER                      TITLE

None.

**G. COMPONENT FAILURE DATA**

Since no component failure occurred, this section is not applicable.



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ATTACHMENT 1

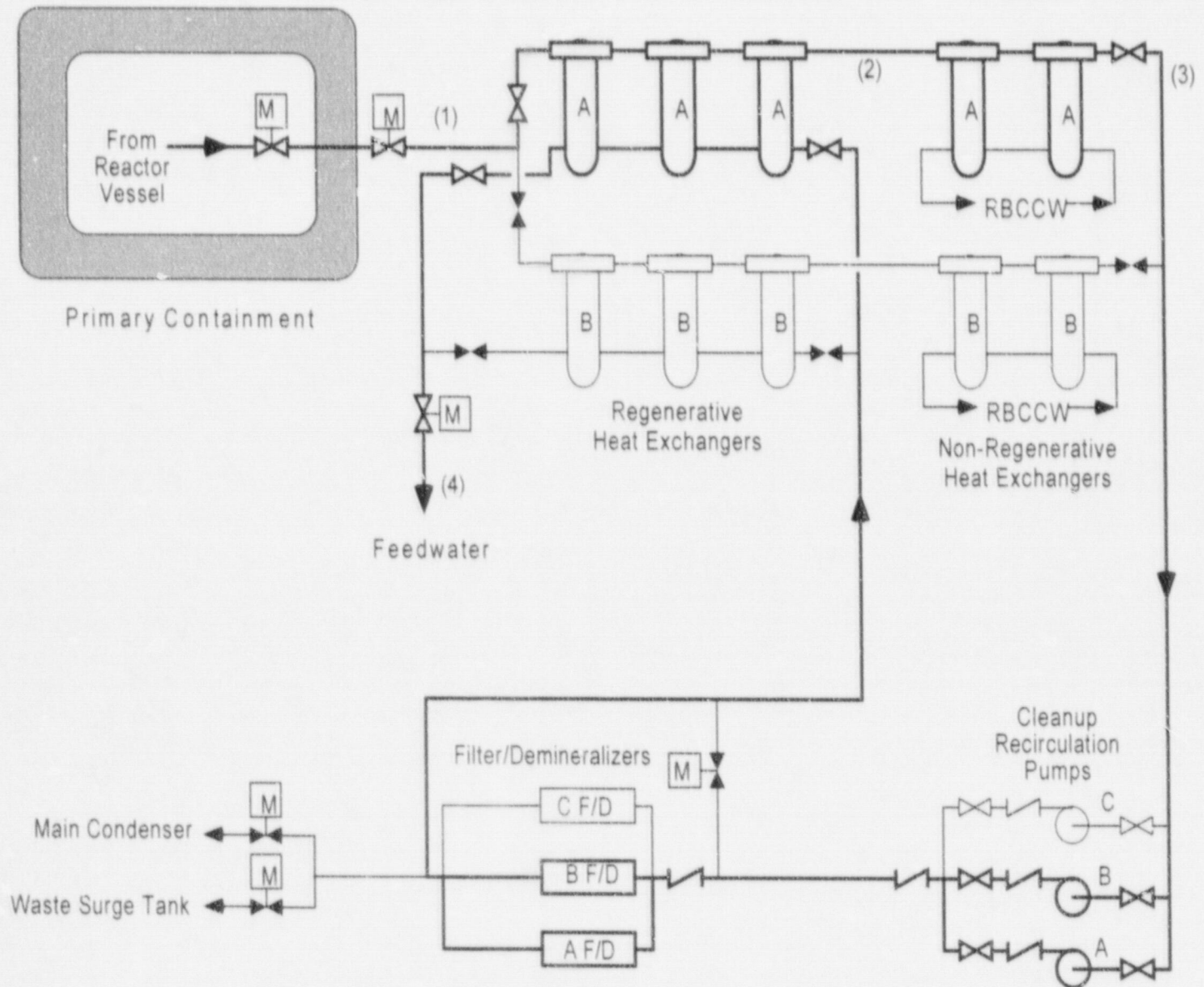


Figure 1: Reactor Water Cleanup System