# LICENSEE EVENT REPORT (LER)

| FACILITY NAME (1) | DOCKET NUMBER (2) | PAGE (3) |
|---|---|---|
| Washington Nuclear Plant - Unit 2 | 0 5 0 0 0 3 9 7 | 1 OF 13 |

TITLE (4)
LOSS OF CONTAINMENT INTEGRITY DUE TO ORIGINAL DESIGN ERRORS AND PROCEDURE DEFICIENCIES

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

OPERATING MODE (9): 1

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)

POWER LEVEL (10): 1 0 0

| | | | |
|---|---|---|---|
| 20.402(b) | 20.405(C) | 50.73(a)(2)(iv) | 77.71(b) |
| 20.405(a)(1)(i) | 50.36(c)(1) | 50.73(a)(2)(v) | 73.73(c) |
| 20.405(a)(1)(ii) | 50.36(c)(2) | 50.73(a)(2)(vii) | OTHER (Specify in Abstract below and in Text, NRC Form 366A) |
| 20.405(a)(1)(iii) | X 50.73(a)(2)(i) | 50.73(a)(2)(viii)(A) | |
| 20.405(a)(1)(iv) | X 50.73(a)(2)(ii) | 50.73(a)(2)(viii)(B) | |
| 20.405(a)(1)(v) | 50.73(a)(2)(iii) | 50.73(a)(2)(x) | |

LICENSEE CONTACT FOR THIS LER (12)

| NAME | TELEPHONE NUMBER (AREA CODE) |
|---|---|
| C. D. Mackaman, Licensing Engineer | 5 0 9 3 7 7 - 4 4 5 1 |

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 |
|---|---|---|---|---|---|---|---|---|---|
| | | | | | | | | | |
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SUPPLEMENTAL REPORT EXPECTED (14)

YES (If yes, complete EXPECTED SUBMISSION DATE) X NO

EXPECTED SUBMISSION DATE (15): MONTH DAY YEAR

ABSTRACT (16)


On March 18, 1993, and April 12, 1993, design errors were identified by Supply System engineers working in the Component Safety Classification (CSC) and Design Requirement Document (DRD) Programs. These are ongoing design basis reconstitution programs that provide a detailed review of WNP-2 design. The CSC Program uses an Electric Power Research Institute (EPRI) Guideline based methodology to establish and document the technical basis for the safety classification of each safety-related component. The DRD Program provides a documented reference to system level design requirements and functions.

The root cause for these events was deficiencies in original plant design.

Immediate and further corrective actions include, but are not limited to, entering Technical Specification Action Statements and taking compensatory action, additional testing and maintenance, Plant Procedure changes, Technical Specification changes, design reviews, and design changes.

Plant Conditions

Power Level - 100%
Plant Mode - 1 (Power Operation)

Event Description

On March 18, 1993, and April 12, 1993, design errors were identified by Supply System engineers working in the Component Safety Classification (CSC) and Design Requirement Document (DRD) Programs. These are ongoing design basis reconstitution programs that provide a detailed review of WNP-2 design. The CSC Program uses an Electric Power Research Institute (EPRI) Guideline based methodology to establish and document the technical basis for the safety classification of each safety-related component. The DRD Program provides a documented reference to system level design requirements and functions.

This LER is written with each item discussed as a separately numbered paragraph under the major headings of Specific Event Description, Immediate Corrective Action, Further Evaluation, Specific Further Corrective Action, and Specific Safety Significance. A general discussion of all items is found under General Event Description, above, and General Further Corrective Actions, General Safety Significance and Similar Events, below.

Specific Event Description

1. Reactor Core Isolation Cooling (RCIC) Primary Containment Release Path

   On March 18, 1993, a routine review of the safety-related containment portions of the RCIC system by CSC engineers found that the containment isolation design for the RCIC pump (RCIC-P-1) suction line from the suppression pool did not meet single failure design criteria. A failure of Division 1 DC power to the RCIC system would cause the motor operators for the Pump Suction Isolation Valve (RCIC-V-31) and the Auxiliary Cooling Water Supply Valve (RCIC-V-46) to fail in an as-is open position. This single failure, combined with a Loss Of Coolant Accident (LOCA), results in an unisolable suppression pool water flow path from Primary Containment to the Reactor Building (see Figure 1). This potential release path violates commitments made to General Design Criterion (GDC) 56 for isolation of the RCIC-P-1 suction line containment penetration. The failure to comply with GDC 56 resulted in Primary Containment being inoperable.

2. <u>Miscellaneous Waste Radioactive (MWR) Drain Secondary Containment Release Path</u>

On April 12, 1993, during the preparation of the Equipment and Floor Drain DRD, a potential degradation of the Secondary Containment (Reactor Building) boundary was identified. There were two MWR drains that penetrate Secondary Containment without adequate isolation controls established. This condition could cause a release of radioactive material to the Radwaste Building or a degradation of Secondary Containment integrity. Each MWR drain line is provided with an air operated solenoid isolation valve located inside Secondary Containment. However, the valves are not safety-related and they do not automatically close during accident conditions. Furthermore, they are not administratively controlled to secure them closed in plant operating modes 1,2,3 and during core alterations. This condition is contrary to the Secondary Containment integrity requirements defined for Technical Specification 3.6.5.1.

<u>Immediate Corrective Action</u>

Immediate corrective actions were initiated for each item discovered during the CSC and DRD Program reviews. They are enumerated below in paragraphs corresponding to the event description above:

1. <u>RCIC Primary Containment Release Path</u>

a. On March 19, 1993, RCIC-V-31 was closed, and its motor operator was de-energized by tagging open the circuit breaker, to restore Primary Containment operability. This action removed the capability of RCIC to transfer RCIC-P-1 suction automatically to the suppression pool, although pump suction was still available from the Condensate Storage Tanks (CSTs). Since automatic transfer capability is a condition of operability as defined in Technical Specification 3.7.3, the RCIC system was declared inoperable at 1530 hours on March 19, 1993, and the associated Limiting Condition for Operation (LCO) was entered.

b. On March 26, 1993, the Supply System formally requested emergency relief from Technical Specification 3.7.3 to allow continued power operation until the 1993 Refueling Outage (R-8).

c. On April 1, 1993, the Supply System formally requested an emergency amendment to Technical Specification 3.7.3 to temporarily restore RCIC operability until the beginning of R-8.

d. At 1500 hours on April 2, 1993, WNP-2 received a Notice of Enforcement Discretion allowing continued power operation beyond the 14 day LCO of Technical Specification 3.7.3.

e. The RCIC system was declared operable at 2232 hours on April 9, 1993, based upon approval of the Technical Specification amendment request.

2. MWR Drain Secondary Containment Release Path

a. At 1827 hours on April 12, 1993, the MWR drain line air operated isolation valves (MWR-V-120 and MWR-V-121) were disabled and closed by isolating supplied air, then tagged closed to assure Secondary Containment integrity.

b. At 0908 hours on April 13, 1993, the loop seals upstream of MWR-V-120 and MWR-V-121 were filled with water to provide additional assurance of Secondary Containment integrity.

Further Evaluation and Corrective Action

A. Further Evaluation

These events are reportable under 10CFR50.73(a)(2)(i)(B) as "Any operation or condition prohibited by Technical Specifications," and 10CFR50.73(a)(2)(ii)(B) as "Any event or condition . . . that resulted in the nuclear power plant being . . . In a condition that was outside the design basis of the plant . . .."

There were no structures, components or systems that were inoperable prior to the start of these events that contributed to the events.

Further evaluations were performed on each item discovered during the CSC and DRD Program reviews. They are enumerated below in paragraphs corresponding to the event description above:

1. RCIC Primary Containment Release Path

a. In accordance with 10CFR50.72(b)(1)(ii)(B), this event was reported to the NRC Operations Center via the Emergency Notification System (ENS) at 1413 hours on March 19, 1993, as "Any event or condition during operation that . . . results in the nuclear power plant being . . . In a condition that is outside the design basis of the plant . . .."

b. A Loss Of Coolant Accident (LOCA) event scenario that could lead to the release of suppression pool water to the Reactor Building is as follows (see Figure 1):

(1) The plant is at power with RCIC-P-1 suction in its normal alignment to the Condensate Storage Tank (CST).

(2) A Small Break Loss Of Coolant Accident (SBLOCA) event occurs causing RCIC to initiate, or a Large Break Loss Of Coolant Accident (LBLOCA) event occurs causing RCIC-V-46 to open on low Reactor Pressure Vessel (RPV) water level.

(3) During the event, RCIC-P-1 suction transfers from the CST to the suppression pool on low CST water level. This automatic suction transfer opens RCIC-V-31. For the LBLOCA event, the transfer could occur as a result of a seismically induced failure of the nonsafety-related CST.

(4) After the pump suction transfer, a loss of Division 1 DC power to the RCIC system occurs.

(5) The DC power failure causes a loss of RCIC control and power to the RCIC-V-31 and RCIC-V-46 motor operators resulting in both valves remaining open.

(6) The Lube Oil Cooler Pressure Control Valve (RCIC-PCV-15) will go full open attempting to maintain the downstream cooler pressure at approximately 50 psig.

(7) Suppression pool water will then drain through RCIC-V-31, RCIC-P-1, RCIC-V-46, RCIC-PCV-15, the lube oil cooler and the barometric condenser to the vacuum tank (see Figure 1). The elevation difference between the suppression pool water level and the vacuum tank will produce a static head sufficient to lift the vacuum tank relief valve at its setpoint of 6 psig.

(8) The suppression pool water will flow through the vacuum tank relief valve at approximately 6.5 gpm to the Equipment Drain (Radioactive) (EDR) system. The EDR system will then transfer the water to the Reactor Building equipment drain sump, where it will be collected. Since the sump is automatically isolated during a LOCA, it will fill and eventually overflow, flooding connecting rooms.

c. GDC 56 (10CFR50, Appendix A) requires each line that connects directly to the containment atmosphere and penetrates primary reactor containment to be provided with containment isolation valves. There must be one locked closed or automatic isolation valve inside containment and one locked closed or automatic isolation valve outside containment, or be demonstrated acceptable on some other defined basis. WNP-2 met containment isolation provisions for the RCIC-P-1 suction line

containment penetration in accordance with prerequisites defined in Standard Review Plan (SRP) 6.2.4, Section II, Paragraph 6.e. Specifically, the provisions are: (1) System reliability must be greater with one isolation valve (instead of two) in the line; (2) The system must be closed outside containment and a single active failure can be accommodated with only one isolation valve; (3) This closed system must be protected from missiles; (4) The closed system must be designed to Seismic Category I, Safety Class 2 requirements, and a minimum temperature and pressure rating at least equal to that for the containment; and (5) The piping between the isolation valve and containment must be enclosed in the leak-tight housing or conservative design of the piping and valve, conforming to SRP 3.6.2, to preclude a breach of piping integrity.

The WNP-2 Final Safety Analysis Report (FSAR), Section 6.2.4.3.2.2.2.1, "HPCS, LPCS, RCIC, and RHR Suction Lines," provides justification for this exception to GDC 56 on the basis that RCIC system reliability is greater with only one isolation valve. Furthermore, the RCIC-P-1 suction piping is considered an extension of containment, and is designed to the same quality standards as containment. Although, SRP 6.2.4 and the FSAR provide acceptable basis for having only one isolation valve, a postulated single failure event causing RCIC-V-31 and RCIC-V-46 to simultaneously fail in an as-is open position was not addressed in the WNP-2 FSAR. Consequently, Primary Containment was outside the plant design basis, and technically inoperable during Operational Conditions 1, 2 and 3. Technical Specification 3.6.1.1 was violated when Primary Containment integrity was not maintained and became inoperable.

2. <u>MWR Drain Secondary Containment Release Path</u>

a. In accordance with 10CFR50.72(b)(1)(ii)(B), this event was reported to the NRC Operations Center via the Emergency Notification System (ENS) at 1750 hours on April 12, 1993, as "Any event or condition during operation that . . . results in the nuclear power plant being . . . In a condition that is outside the design basis of the plant . . ." This notification was based upon a concern that the two MWR drain line penetrations identified were outside the Secondary Containment design basis as defined in ANSI/ANS-52.1-1983, Section 4.10.5.2, "System Design Criteria." However, further investigation found that WNP-2 is not committed to this criteria. The MWR drain line penetrations were found to meet WNP-2 FSAR, NUREG-0737, Section III.D.1.1, "Primary Coolant Sources Outside Containment," and post TMI-2 commitments. Consequently, this event did not cause WNP-2 to be outside the design basis of the plant, and the NRC notification pursuant to 10CFR50.72(b)(1)(ii)(B) was unnecessary.

b. The safety function of Secondary Containment prevents and controls the release of radiation to the public. Any penetration through Secondary Containment should automatically isolate during accident conditions, or be secured (deactivated) in the closed position as defined in Technical Specification Definition 1.3.9.a, "Secondary Containment Integrity."

The MWR drain line penetrations identified are provided with nonsafety-related air operated solenoid isolation valves inside Secondary Containment (MWR-V-120 and MWR-V-121). These valves fail closed on loss of supplied air or electrical control power, but have no automatic design safety or control functions. Each valve is manually controlled by a panel control switch located in the Radwaste Control Room. Operating Procedure PPM 2.11.5, "Floor Drain System," identifies MWR-V-120 as normally open and MWR-V-121 as normally closed. The procedure does not specify when MWR-V-120 should be closed. Surveillance Procedure PPM 7.4.6.5.1.1, "Secondary Containment Integrity Verification," does not include MWR-V-120 or MWR-V-121. The MWR drain lines serve Reactor Building drains for an emergency shower located on the 548' elevation; the Control Rod Drive (CRD) repair room, a decontamination station and one miscellaneous drain on the 501' elevation, and the fuel cask wash down area and service box on the 606' elevation. Each drain line is provided with a loop seal upstream of the isolation valves; however, these drains were not frequently used and the loop seals may have become dry at various times.

The Standby Gas Treatment System (SGTS) is periodically tested to meet the Secondary Containment (Reactor Building) drawdown time and leakage limit requirements of Justification for Continued Operation (JCO) 288-357, Revision 2. As of the last test, the configuration of MWR-V-120 and MWR-V-121, and the associated drains, did not impact the ability to meet the JCO requirements.

B. General Root Cause

The general root cause for these events was deficiencies in original plant design.

Specific Root Cause

Root causes were determined for each item discovered during the CSC and DRD Program reviews. They are enumerated below in paragraphs corresponding to the event description above:

1. <u>RCIC Primary Containment Release Path</u>

   The root cause for this event was a deficiency in the original design of the Containment Isolation system. The single failure criterion for the postulated event scenario described above was not considered in the safety analysis. A contributing cause was the failure to establish a corrective action for a similar event in February 1988, to look generically for similar concerns.

2. <u>MWR Drain Secondary Containment Release Path</u>

   The root cause for this event was a deficiency in the original component safety classification of the MWR system. MWR-V-120 and MWR-V-121 were not identified as having a passive safety-related function for Secondary Containment integrity and classified as nonsafety-related. A contributing cause was the failure to identify this deficiency during a similar event in December 1990.

C. <u>General Further Corrective Action</u>

No further corrective action is being considered for the original design errors and deficiencies discovered. Due to the detailed nature of the CSC Program and DRD Program engineering reviews, design errors and omissions of the type reported in this LER could be reasonably expected to be found. Because these programs are ongoing, there is the likelihood that additional reportable events will result. These events may be reported as revisions to this LER.

<u>Specific Further Corrective Action</u>

1. <u>RCIC Primary Containment Release Path</u>

   a. Perform an Engineering evaluation for a permanent resolution to ensure the RCIC-P-1 suction line meets single failure design criteria. This action to be completed by May 24, 1993.

   b. Implement the RCIC-P-1 suction line single failure Engineering evaluation recommendation(s) before startup from the R-8 refueling outage, or obtain Technical Specification 3.7.3 relief to allow startup.

   c. Perform an Engineering review of containment penetrations that rely on single valve isolation to determine if single failure related unisolable flow paths from containment exist. This action will be completed by September 30, 1993.

2. MWR Drain Secondary Containment Release Path

MWR-V-120 and MWR-V-121 will remain under administrative control in accordance with Administrative Procedure PPM 1.3.8, "Danger Tag Clearance Order," until the following corrective actions are completed:

a. Revise PPM 2.11.5 by June 15, 1993, to:

(1) Identify MWR-V-120 and MWR-V-121 as normally closed.

(2) Add direction and administrative controls as to when MWR-V-120 and MWR-V-121 may be opened.

b. Revise PPM 7.4.6.5.1.1 by June 15, 1993, to include MWR-V-120 and MWR-V-121.

c. Revise the Master Equipment List (MEL) for MWR-V-120 and MWR-V-121 with a Component Classification Evaluation Record (CCER) to reflect the passive safety-related function, and initiate a MEL/Work In Progress (WIP) Error Documentation Report (MEDR). These actions to be completed by June 15, 1993.

d. Disposition the MEDR by June 25, 1993.

## General Safety Significance

The Supply System regards the programmatic aspects of these items as an important issue that had potential safety significance. The CSC and DRD Program reviews are chartered to address these concerns, and in so doing, will improve plant design and minimize design related Technical Specifications noncompliance problems in the future.

## Specific Safety Significance

The Safety Significance was determined for each item discovered during the CSC and DRD Program reviews. They are enumerated below in paragraphs corresponding to the event description above:

1. RCIC Primary Containment Release Path

The FSAR, Section 6.3.3, "Emergency Core Cooling System (ECCS) Performance Evaluation," shows that a SBLOCA event does not lead to fuel damage. Consequently, the postulated DC power failure event scenario described above would only release reactor coolant activation and corrosion

products to the suppression pool water and Reactor Building equipment drain sump. This limited release of radionuclides would be treated by the SGTS before release to the environment.

According to FSAR, Section 15.6.5.5.1.1, "Fission Product Release from Fuel," a postulated LBLOCA and Division 1 DC power failure event scenario would result in significant fission product release from the fuel. According to FSAR, Section 15.6.5.5.1.2, "Fission Product Transport to the Environment," some of the release would be to the suppression pool. While the potential release to Secondary Containment would be treated by the SGTS, the fission product transport to the environment could have safety significance.

A Division 1 DC power failure event scenario involving fuel damage would result in a release of radionuclides to Secondary Containment. The postulated flow path to the Reactor Building equipment drain sump would be addressed by Annunciator Response Procedure PPM 4.601.A4, "601.A4 Annunciator Panel Alarms," and Emergency Operating Procedure (EOP) PPM 5.3.1, "Secondary Containment Control." This would result in action to close RCIC-V-31 when the RCIC Pump Room water level reached six inches (the maximum safe operating value). However, during this event scenario, the valve could not be closed due to the loss of DC power and local area high radiation levels. The airborne release would be minimal and would be treated by the SGTS before release to the environment.

2. <u>MWR Drain Secondary Containment Release Path</u>

The only significant source of radioactive material in the MWR drain lines is from the CRD repair room flush tank. This material is crud removed during the rebuilding of CRD assemblies. A limited quantity of material could possibly escape Secondary Containment through the drain system.

Surveillance procedure PPM 7.4.6.5.1.2, "Standby Gas Treatment Functional Test," verifies that the SGTS can drawdown the Secondary Containment to greater than or equal to -0.25 inch of water gauge in less than or equal to 120 seconds. This test is performed at least every 18 months, and there has never been a failure due to excessive Secondary Containment leakage. In accordance with Technical Specification 3/4.6.5 Bases, "Secondary Containment," this ensures there were no violations of Secondary Containment integrity. Consequently, this event posed no threat to the health and safety of either the public or plant personnel.

| LICENSEE EVENT REPORT (LER)<br>TEXT CONTINUATION | | | |
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## Similar Events

On February 19, 1988, LER 88-002 reported an original plant design error that caused the RCIC-P-1 suction line not to meet containment isolation and single failure criteria. In February 1988, the Supply System had not implemented a formal Root Cause Analysis Program, as it was still in the formative stages. Consequently, no corrective action was identified in the LER or the associated Nonconformance Report (288-0022) that would have prevented a similar occurrence. The Supply System implemented a Root Cause Analysis Program in December 1988, in accordance with Plant Administrative Procedure PPM 1.3.48, "Root Cause Analysis," that addresses similar generic concerns to prevent recurrence.

On December 11, 1990, LER 90-032 and associated NCR 290-0972 reported an original design error that cross-connected the Reactor Building Exhaust Air (REA) system and the Plant Sanitary Drain (PSD) system resulting in a Secondary Containment leakage path. System walkdowns conducted for similar conditions under Problem Evaluation Request (PER) 290-972 were not scoped to identify the MWR drain deficiency. The ongoing DRD and CSC Program reviews are expected to find any further design errors of this nature.

EIIS Information

| Text Reference | EIIS Reference System | EIIS Reference Component |
|---|---|---|
| Reactor Core Isolation System (RCIC) | BN | --- |
| RCIC System Suction Line | BN | PSP |
| Condensate Storage Tank (CST) | KA | TK |
| RCIC Pump 1 (RCIC-P-1) | BN | P |
| Suppression Pool Suction Valve (RCIC-V-31) | BN | ISV |
| Auxiliary Cooling Water Supply Valve (RCIC-V-46) | BN | ISV |
| RCIC Motor Operated Valves | BN | 20 |
| Primary Containment | NH | --- |
| Secondary Containment | NH | --- |
| Reactor Building | NG | --- |
| Pressure Control Valve (RCIC-PCV-15) | BN | PCV |
| RCIC Lube Oil Cooler | BN | CLR |
| RCIC Barometric Condenser | BN | CDU |
| Equipment Drain (Radioactive) (EDR) System | WK | --- |
| Equipment Drain Sump | WK | TK |
| DC Power | EJ | JX |
| Standby Gas Treatment System (SGTS) | BH | --- |
| Miscellaneous Waste Radioactive (MWR) System | WK | --- |
| MWR Isolation Valves (MWR-V-120 and MWR-V-121) | WK | ISV |

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TITLE (4)
LOSS OF CONTAINMENT INTEGRITY DUE TO ORIGINAL DESIGN ERRORS AND PROCEDURE DEFICIENCIES

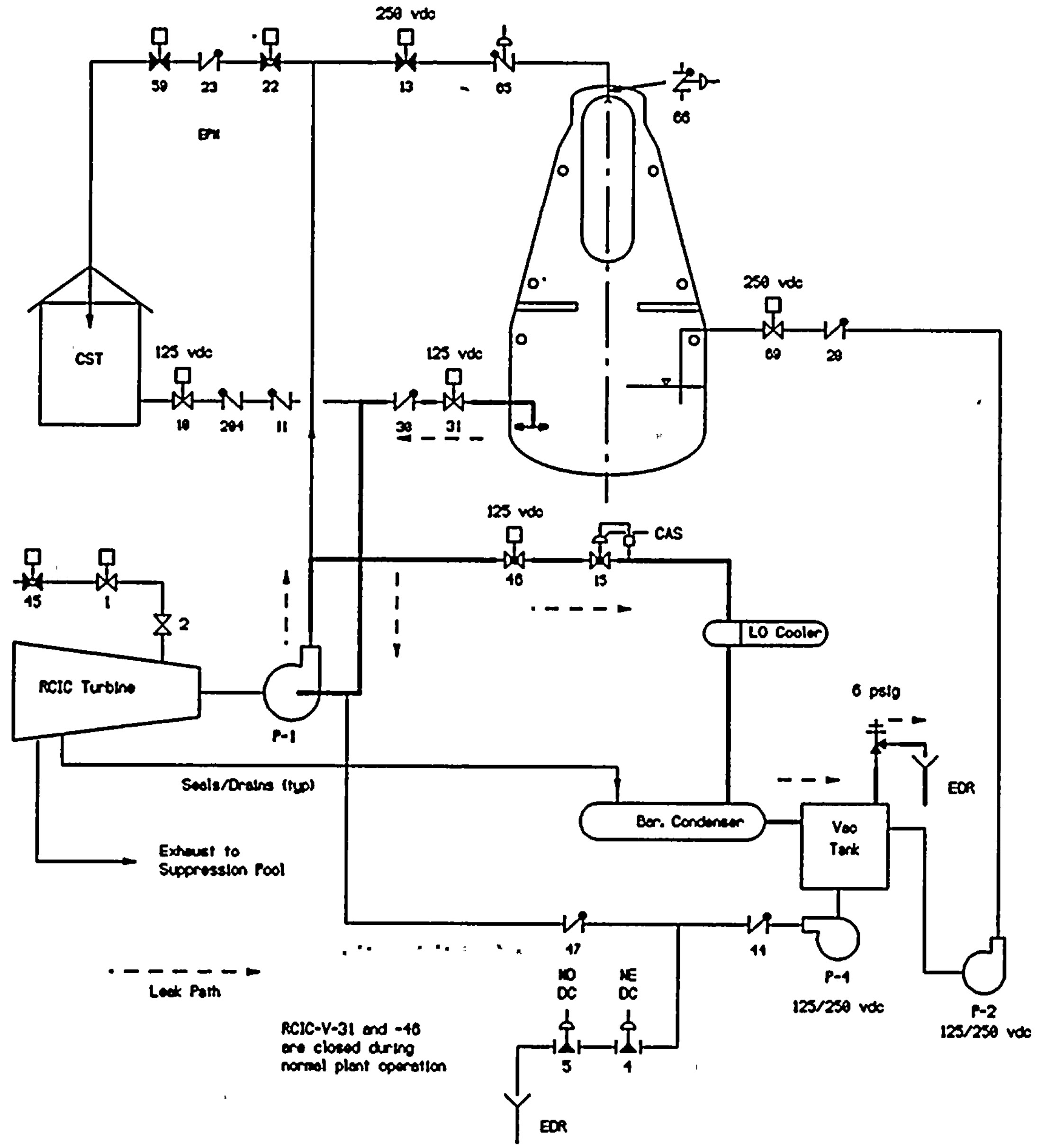


Figure 1