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ILLINOIS POWER

CLINTON POWER STATION, P.O. BOX 678, CLINTON, ILLINOIS 61727-0678, TELEPHONE (217) 935-8881

January 17, 1991

10CFR50.73

Docket No. 50-461

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, D.C. 20555

Subject: Clinton Power Station - Unit 1
Licensee Event Report No. 90-018-00

Dear Sir:

Please find enclosed Licensee Event Report No. 90-018-00: Failure to Consider All Potential Containment Atmosphere Leakage Pathways Requiring Testing In Accordance With 10CFR50, Appendix J. This report is being submitted in accordance with the requirements of 10CFR50.73.

Sincerely yours,

D L Holtzscher for

F. A. Spangenberg, III
Manager - Licensing and Safety

STH/alh

Enclosure

cc: NRC Resident Office
NRC Region III, Regional Administrator
INPO Records Center
Illinois Department of Nuclear Safety
NRC Clinton Licensing Project Manager

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NRC FORM 366 (6-89)

U.S. NUCLEAR REGULATORY COMMISSION

APPROVED OMB NO. 3150-0104
EXPIRES: 4/30/92

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE RECORDS AND REPORTS MANAGEMENT BRANCH (P-530), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Clinton Power Station	0 5 0 0 0 4 6 1	1 OF 1 4

TITLE (4) Failure to Consider All Potential Containment Atmosphere Leakage Pathways Requiring Testing In Accordance with 10CFR50, Appendix J

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 NUMBER(S)
1 2	1 8	9 0	9 0	0 1 8	0 0	0 1	1 7	9 1	None	0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 5

POWER LEVEL (10): 0 0 0

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

20.402(b)	20.405(c)	50.73(a)(2)(iv)	73.71(b)
20.405(a)(1)(i)	50.36(c)(1)	50.73(a)(2)(v)	73.71(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
J. A. Puzauskas, Assistant Director - Design and Analysis Engineering, Extension 3094	AREA CODE 2 1 7 9 3 5 - 8 8 8 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

SUPPLEMENTAL REPORT EXPECTED (14)		EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
X YES (If yes, complete EXPECTED SUBMISSION DATE)	NO		0 4	0 1	9 1

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

On December 18 1990, during the plant's second refueling outage, with the plant in Mode 5 (REFUELING), Illinois Power Company determined that the mechanical joints on the discharge side of 1E12-F055A and 1E12-F055B, Residual Heat Removal (RHR) 'A' and 'B' Heat Exchanger (HX) Relief Line Valves, had not been local leak rate tested as required by 10CFR50, Appendix J. Investigation into this event identified other penetrations for which not all potential containment atmosphere leak paths in accordance with the requirements of 10CFR50, Appendix J, were considered; and identified three penetrations which do not terminate in the suppression pool at or below the design level required to maintain the penetration lines water sealed following a Loss of Coolant Accident (LOCA). The cause of this event is still under investigation. Corrective actions include: a review of all containment penetrations to ensure that proper test criteria are being implemented; identification of mechanical joints that could be potential containment atmosphere leakage paths; and modification of the identified penetrations to facilitate conformance with the requirements of 10CFR50, Appendix A, General Design Criteria 55 and 56, and Appendix J.

DESCRIPTION OF EVENT

On December 18, 1990, at 1030 hours, during the plant's second refueling outage (RF-2), the plant was in Mode 5 (REFUELING), with reactor coolant temperature at approximately seventy-four degrees Fahrenheit and at atmospheric pressure. At that time, Illinois Power Company (IP) determined that the mechanical joints on the discharge side of 1E12-F055A and 1E12-F055B, Residual Heat Removal (RHR) [BO] 'A' and 'B' Heat Exchanger (HX) [HX] Relief Line valves [RV], had not been local leak rate tested as required by 10CFR50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," since initial plant operation. Specifically, a Local Leak Rate Test (LLRT) had not been performed on these mechanical joints.

On December 11, 1990, the Nuclear Regulatory Commission (NRC) site resident inspector questioned a Plant Technical test engineer on the plant's requirements for testing the mechanical joints on the discharge side of relief valves 1E12-F055A and 1E12-F055B. A review of the Clinton Power Station (CPS) startup Local Leak Rate Test (XTP-00-07), Technical Specifications, and the Updated Safety Analysis Report (USAR) did not identify any testing requirements for these mechanical joints. A review of CPS pre-operational Integrated Leak Rate Test (PTP-ILDW-01) and CPS surveillance procedure 9861.01, "Integrated Leak Rate Test," revealed that these mechanical joints had not been exposed to containment atmosphere during the ILRT.

The CPS USAR references the associated containment penetrations (1MC-24 and 1MC-26) [PEN] as typical containment penetrations isolable by one valve (1E12-F055A and 1E12-F055B, respectively) [V] and a Closed Loop Outside Containment (CLOC). According to Technical Specification Table 3.6.4-1, "Containment Isolation Valves," and USAR Table 6.2-47, 1E12-F055A and 1E12-F055B require a Type C water leak test, since the inboard side of 1E12-F055A and 1E12-F055B is assumed to be water sealed from containment atmosphere by the suppression pool.

However, the design of the RHR HX Relief valve discharge lines also includes vacuum breakers [VACB], which will permit containment atmosphere to enter the discharge lines, in order to prevent steam condensation in the lines from drawing water from the suppression pool following operation of the relief valves. Water in these lines could jeopardize their integrity in the event of subsequent relief valve operation. These vacuum breaker lines are connected to the Reactor Core Isolation Cooling (RCIC) [BN] Turbine [TRB] Exhaust Vacuum Relief line on the outboard side of 1E51-F078, RCIC Turbine Exhaust Vacuum Breaker Inboard Isolation Valve [ISV], for containment penetration 1MC-44. (See Figure 1) Following a postulated Loss of Coolant Accident (LOCA) event, the mechanical joints on the discharge side of 1E12-F055A and 1E12-F055B would be exposed to containment atmosphere if 1E51-F078 failed to isolate in the closed position. IP has determined that the failure of 1E51-F078 is a credible single active failure. Therefore, a Type C leak test of the penetration (including valves 1E12-F055A and 1E12-F055B and the associated mechanical joints) would be required to be performed using a gas test rather than a water test as specified by Technical Specification Table 3.6.4-1 and USAR Table 6.2-47.

RHR HX vent line containment penetrations, 1MC-89 and 1MC-172, were also identified to have similar testing deficiencies. The USAR identifies that penetrations 1MC-89 and 1MC-172 are isolable by 1E12-F074A and 1E12-F074B (RHR HXs 'A' and 'B' First Vent to Suppression Pool Containment Isolation Valves) and a CLOC. Technical Specification Table 3.6.4-1 and USAR Table 6.2-47 require a Type C water leak test of these valves. The design of the RHR HX vent lines includes vacuum breakers located inside the containment. (See Figure 2) These vacuum breakers permit containment atmosphere to enter the vent lines in order to prevent steam condensation in the lines from drawing water from the suppression pool following venting of non-condensibles and steam from the RHR HXs into the suppression pool. Water in these lines could jeopardize their integrity in the event of subsequent HX venting. Per this design, 1E12-F074A and 1E12-F074B are exposed to containment atmosphere and a Type C gas leak test should have been specified in the Technical Specifications and the USAR. The USAR did identify these valves require a Type A containment atmosphere test which had been performed during previous ILRTs. Additionally, investigation into this issue determined that mechanical joints exist on the RHR HX vent lines which are above the post-LOCA drawdown level (727-foot 1-inch elevation, which is three-feet ten-inches below the minimum suppression pool level specified in the Technical Specification). Therefore, during a post-LOCA drawdown, these mechanical joints could be potential containment atmosphere leakage paths. This is not in accordance with the USAR, which assumes the line is completely submerged.

On December 18, 1990, after review of these issues, Condition Report (CR) 1-90-12-062 was initiated to document these deficiencies. IP initiated a review of all containment penetrations' testing criteria identified in the Technical Specifications and USAR, to identify if similar potential containment atmosphere leakage paths existed. This review identified the following conditions.

1. Penetration 1MC-41

   IP has determined that the mechanical joint associated with penetration 1MC-41 on the RCIC turbine exhaust sparger line above the suppression pool is a potential containment atmosphere leakage path. (See Figure 1.) The USAR states that:

RCIC Turbine Exhaust will be sealed by the suppression pool. No air from inside containment can leak past this penetration. There may be an initial charge of clean air in this line prior to a LOCA. During a LOCA this clean air may leak out. If this air is exhausted, suppression pool water will reach these valves. Any further leakage will be of suppression pool water. Contaminated containment atmosphere can not reach these valves through this penetration.

In order to make this assumption, the mechanical joint associated with penetration 1MC-41 would also have to be sealed by the suppression pool. This mechanical joint has never been included in the Type B leak test program or included within the boundary of a Type C gas leak test of penetration 1MC-41.

2. Penetration 1MC-42

IP has determined that the mechanical joint inboard of drywell penetration 1MD-42 on the RCIC head spray line (see Figure 3) is a potential containment atmosphere leakage path that has never been Type B gas leak tested or included within the boundary of the Type C gas leak test of penetration 1MC-42. This mechanical joint connects the RCIC head spray line to the refueling bulkhead water seal penetration. The mechanical joint is hydrostatically leak tested at a pressure of 1000 pounds per square inch gauge (psig) each refueling outage and visually inspected with a zero leakage acceptance criteria.

3. Penetrations 1MC-18 and 1MC-20

RHR 'A' and 'B' suppression pool return lines penetrate containment through 1MC-18 and 1MC-20, respectively (see Figures 4 and 5). By design these return lines should terminate at or below the post-LOCA drawdown level. IP has determined that a mechanical joint exists on these lines which is only nine inches below the minimum suppression pool level. Therefore, during a post-LOCA drawdown, these mechanical joints could be potential containment atmosphere leakage paths. The USAR assumes 1MC-18 penetration containment isolation valves 1E12-F024A, 1E21-F012, 1E12-F011A, 1E21-F011, 1E21-F102, and 1E12-F064A are exposed to the hydraulic pressure of the suppression pool which provides a water seal; likewise, for 1MC-20 penetration containment isolation valves 1E12-F024B, 1E12-F011B, and 1E12-F064B. Based on this assumption, these containment isolation valves are Type C water leak tested. Due to this design configuration, these valves should be Type C gas leak tested because of the potential post-LOCA containment atmosphere leakage pathway, but have not been since initial plant operation. IP has also determined that the line terminates horizontally in the

*U.S. GPO: 1985-520-589 00070

suppression pool at a level which would be above post-LOCA drawdown level. This is not in accordance with t USAR, which assumes the line is completely submerged.

4. Penetration 1MC-87

1E12-F025A, RHR 'A' Header Relief Valve, penetrates containment through 1MC-87 and terminates vertically in the suppression pool two and one-half inches above the post-LOCA drawdown level. (See Figure 6.) During a postulated post-LOCA event, the penetration would provide a containment atmosphere leakage path. This is not in accordance with the USAR, which indicates that the line terminates at four feet below minimum suppression pool level (below post-LOCA drawdown level). Additionally, investigation into this matter determined that during construction a tolerance of plus or minus three inches was allowable for location of piping. If the extreme tolerance would have been used, it would have resulted in the penetration not meeting the minimum water coverage requirement. Notwithstanding, two Nonconformance Reports (NCRs), 61083 and 61098, were written during plant construction documenting that the flanged termination of the 1MC-87 penetration line was installed four and one-half inches above design elevation. The disposition of these NCRs was to "use as is".

The discrepancies of penetrations 1MC-18, 1MC-20, and 1MC-87 are documented in CR 1-91-01-007.

The resolution of the discrepancies noted above is described in the corrective action portion of this Licensee Event Report.

No automatic or manually initiated safety system responses were necessary to place the plant in a safe and stable condition. No other equipment or components were inoperable at the start of this event such that their inoperable condition contributed to this event.

CAUSE OF EVENT

The cause of this event is still under investigation, and will be included in a supplement to this report which is expected to be issued by April 1, 1991. IP is taking steps to determine the cause(s) for design deficiencies of penetrations which terminate in the suppression pool at a level above the post-LOCA drawdown level, and for leak rate testing deficiencies of the Safety Analysis Report in analyzing potential containment atmosphere leak paths to determine whether hydrostatic testing of containment isolation valves was permissible. The design and leak testing criteria were originally determined by Sargent and Lundy Engineers (S&L), the plant's architect engineer.

CORRECTIVE ACTION

IP requested S&L to perform a review of all containment penetrations which terminate in the suppression pool. This review was completed on January 2, 1991. Their review did not identify any additional discrepancies other than those already identified in this report.

Additionally, Nuclear Station Engineering Department (NSED) personnel performed a field walkdown to ensure all containment penetrations with lines which terminate in the suppression pool are at the depth specified by the "as-built" design drawings. This review did not discover any discrepancies in the "as-built" design drawings.

In order to prevent exposing valves 1E12-F055A and 1E12-F055B to containment atmosphere if 1E51-F078 failed to isolate in the closed position, IP is implementing a design modification in accordance with Field Engineering Change Notice (FECN) Number 25485, to replace 1E12-F102 (RHR/RCIC Vacuum Breaker Isolation Valve) with a welded blind coupling. This modification will eliminate this potential containment atmosphere leakage pathway and is possible because the vacuum breaker capability is not required for CPS operation. This modification will be worked under Maintenance Work Request (MWR) D15221 which will be completed prior to startup from RF-2.

Penetrations 1MC-89 and 1MC-172 are being modified in accordance with FECN number 25483 to eliminate the containment atmosphere leakage paths. These modifications will be implemented by installing a welded blind coupling on the penetration line inside containment, as close as possible to the containment building wall and upstream of the vacuum breakers. These modifications can be made because the RHR HX vent lines are not required for CPS operations. These modifications will be worked under MWRs D15219 and D15220, which will be completed prior to startup from RF-2.

IP will submit a letter to the NRC to provide interim justification for performing Type B gas leak testing on the mechanical joint above the suppression pool in the penetration 1MC-41 line. The Type B gas leak rate test of the mechanical joint with air at a pressure of 9.0 psig will provide adequate assurance, on an interim basis, that containment atmosphere leakage into the RCIC Turbine Exhaust line will not occur. Therefore, leak rate testing of penetration 1MC-41 containment isolation valves with water would be justified for one operating cycle. This testing and submission of the letter will be completed prior to startup from RF-2. During the plant's third refueling outage, IP will implement a plant modification to replace this mechanical joint with a welded joint.

IP will incorporate the mechanical joint inboard of drywell penetration 1MD-42 on the RCIC head spray line into the Type B leak test program. (See Figure 3.) This mechanical joint will be added to the USAR section of components requiring Type B testing and implemented in the Type B leak test program. This USAR change will be completed by September 29, 1991.

IP will implement a design modification in accordance with Field Alteration RHF015 to lines associated with penetrations 1MC-18 and 1MC-20 to remove the mechanical joints above the post-LOCA drawdown suppression pool level and lower the horizontal line terminations to an elevation below the post-LOCA drawdown level. These modifications will be worked under MWRs D15230 (1MC-18) and D15231 (1MC-20) and will be completed prior to startup from RF-2. Radiography will be used to determine the leak tight integrity of the welds performed in accordance with the design modification.

IP will implement a design modification in accordance with FECN number 25486, to the penetration 1MC-87 line to extend the line termination to a depth below the post-LOCA suppression pool drawdown level. This modification will be worked under MWR 15229 and will be completed prior to startup from RF-2.

IP will determine any corrective actions to prevent recurrence after the root cause has been determined.

ANALYSIS OF EVENT

This event is reportable under the provisions of 10CFR 50.73(a)(2)(i)(B) due to a condition prohibited by Technical Specification 3/4.6.1.2. Technical Specification 3/4.6.1.2 requires primary containment leakage rates be determined in conformance with the criteria specified in 10CFR50, Appendix J. Additionally, this event is reportable under the provisions of 10CFR50.73(a)(2)(ii)(B) because penetrations 1MC-18, 1MC-20 and 1MC-87 did not terminate below the suppression pool design depth required to prevent a containment atmosphere leakage path and were therefore outside the design basis of the plant.

IP is still evaluating the safety consequences and implications of this event. This evaluation will include as-found leak rates, where possible, when evaluating the safety consequences of not properly testing the identified mechanical joints and containment isolation valves. The results of the as-found leak rate test are not yet available, but will be determined during the plant's present refueling outage. The results of this evaluation will be included in a supplement to this report.

Additional Information

No equipment failed during this event.

No reportable events of a similar nature have occurred at Clinton Power Station.

For further information regarding this event, contact J. A. Puzauskas, Assistant Director-Design and Analysis Engineering at (217) 935-8881, extension 3094.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Clinton Powe station	0 5 0 0 0 4 6 1	9 0	— 0 1 8 —	0 0	0 9 OF 1 4

TEXT (If more space is required, use additional NRC Form 366A's) (17)

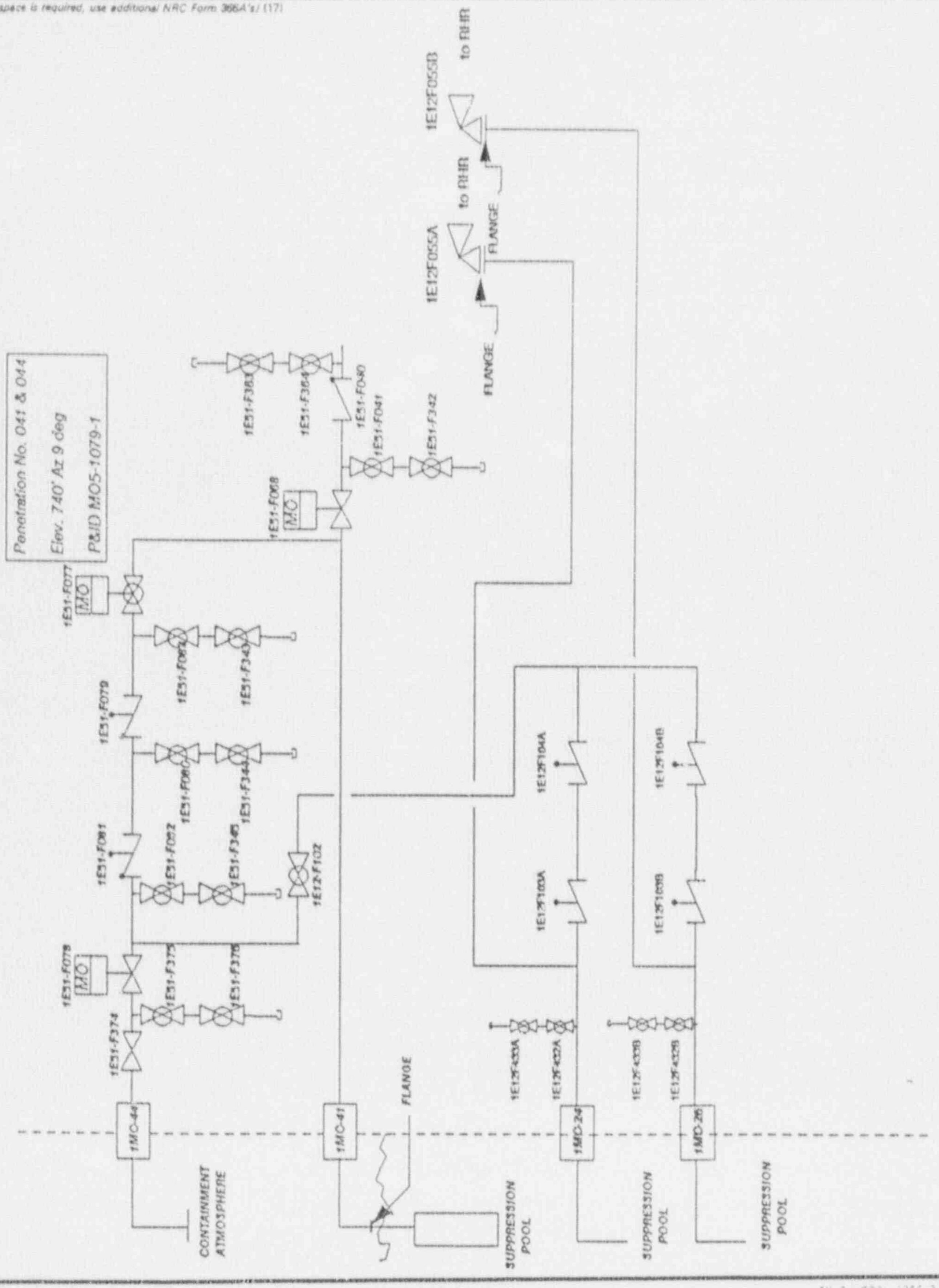
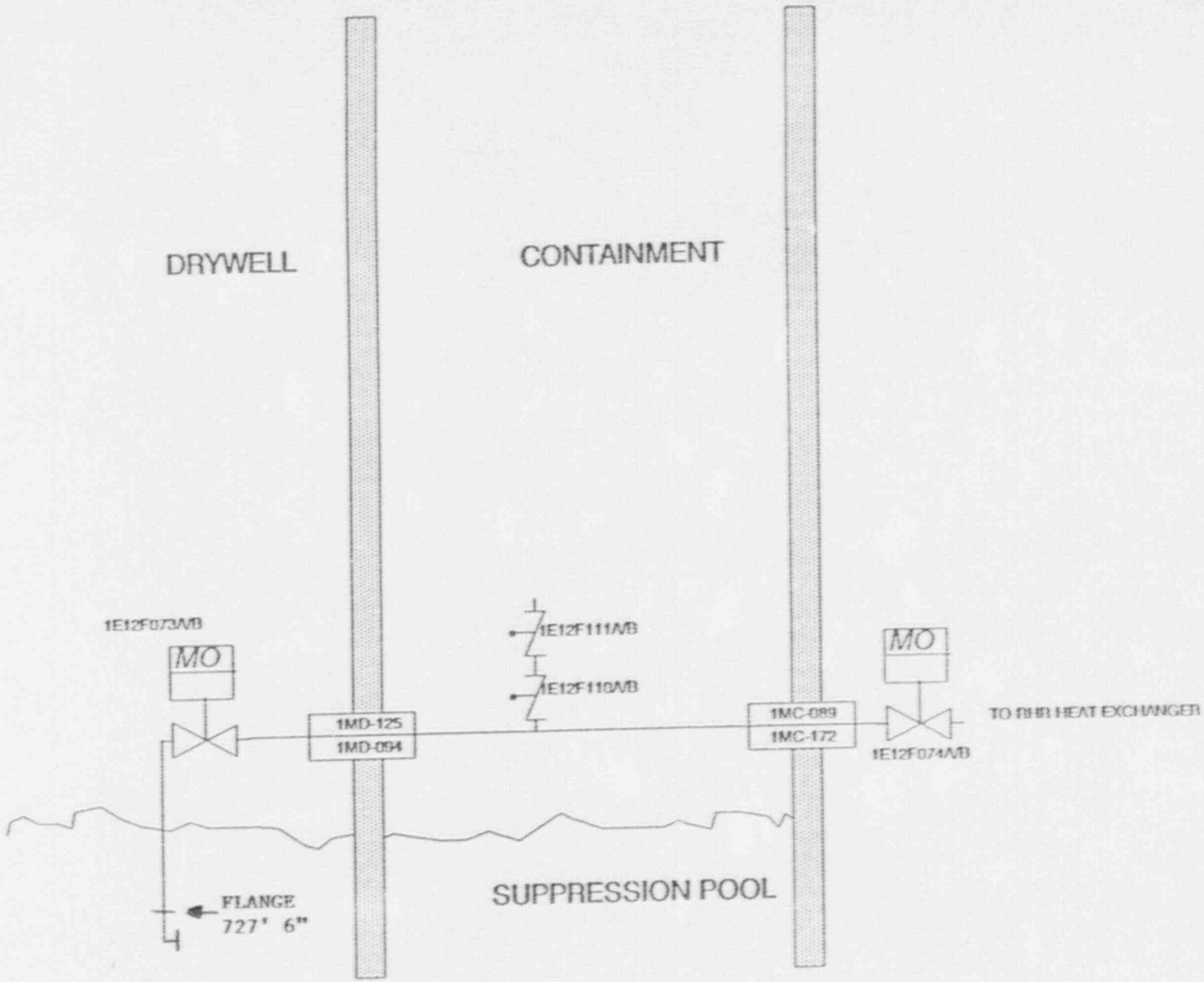


FIGURE 1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1E12F111A/B
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MO
1MD-125
1MD-094
1MC-089
1MC-172
TO RHR HEAT EXCHANGER
1E12F074A/B
FLANGE
727' 6"
SUPPRESSION POOL

FIGURE 2

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Clinton Power Station	0 5 0 0 0 4 6 1	9 0	— 0 1 8 —	0 0	1 1 OF 1 4

TEXT (If more space is required, use additional NRC Form 366A's) (17)

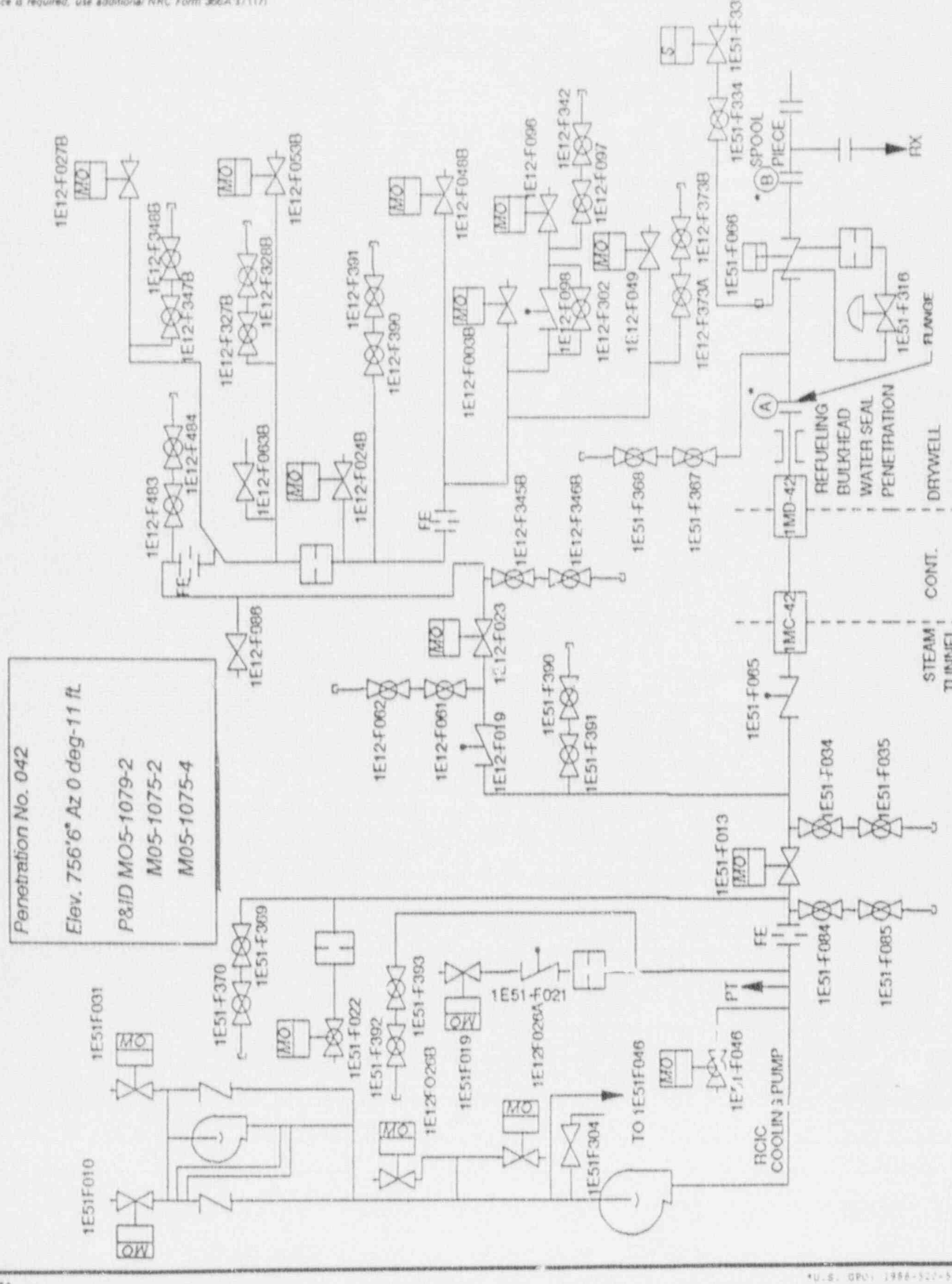

FIGURE 3

NRC Form 366A
(9-83)

U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

APPROVED OMB NO. 3150-0104
EXPIRES: 8/31/88

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
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TEXT (If more space is required, use additional NRC Form 366A's) (17)

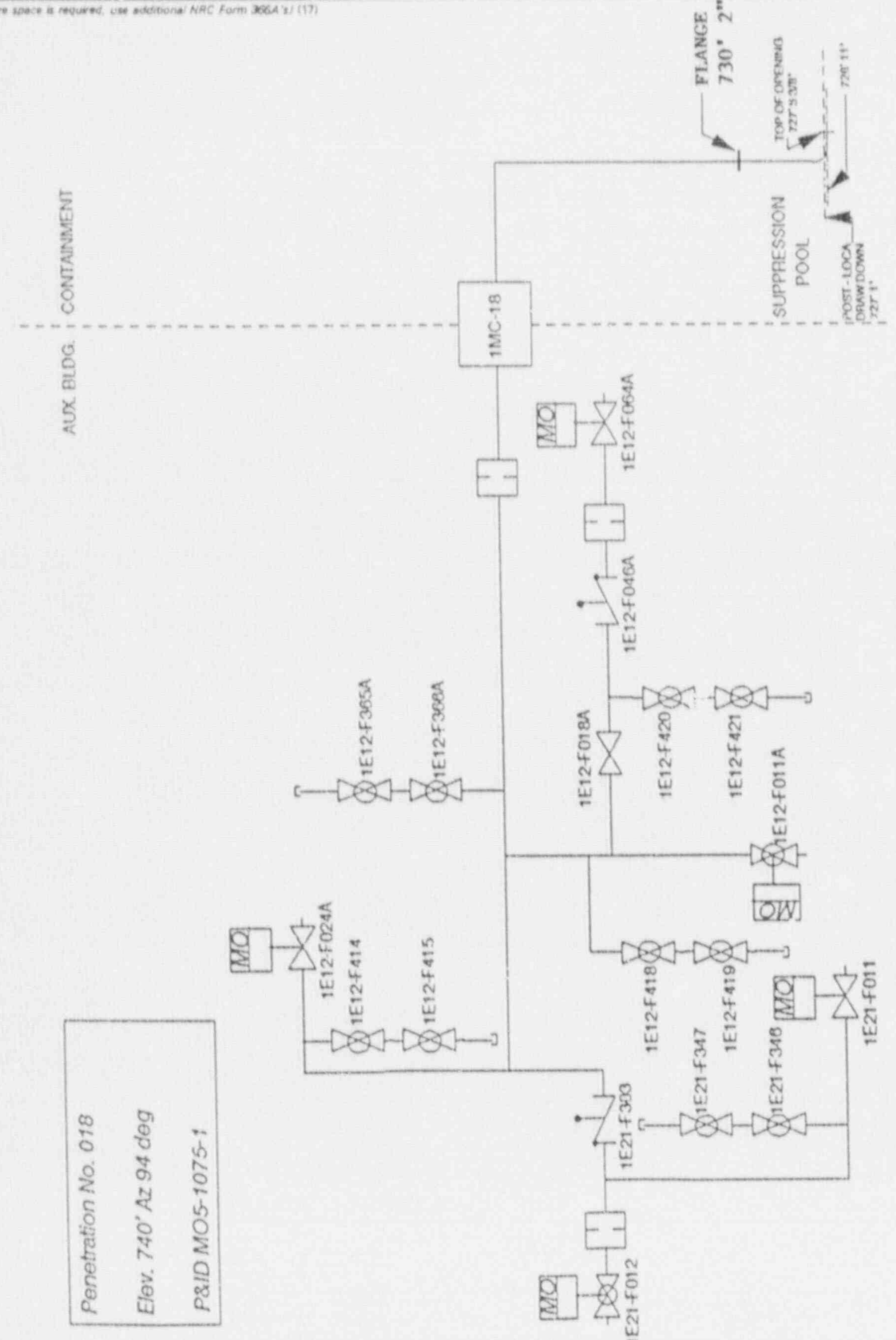

FIGURE 4

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
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TEXT (If more space is required, use additional NRC Form 366A's) (17)

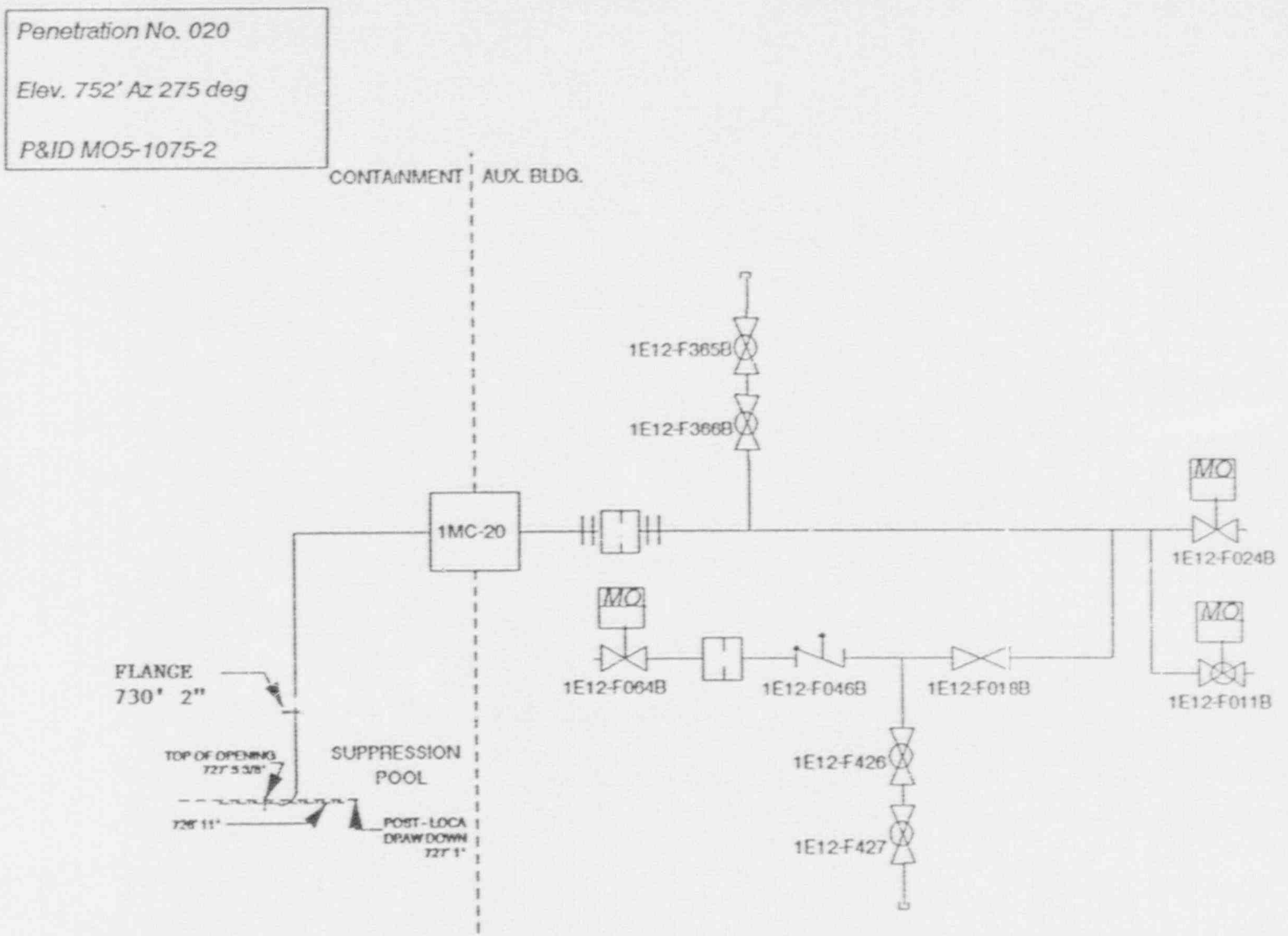

FIGURE 5

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES: 8/31/88

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Clinton Power Station	0 5 0 0 0 4 6 1	9 0 —	0 1 8 —	0 0	1 4 OF 1 4

TEXT (If more space is required, use additional NRC Form 366A's) (17)

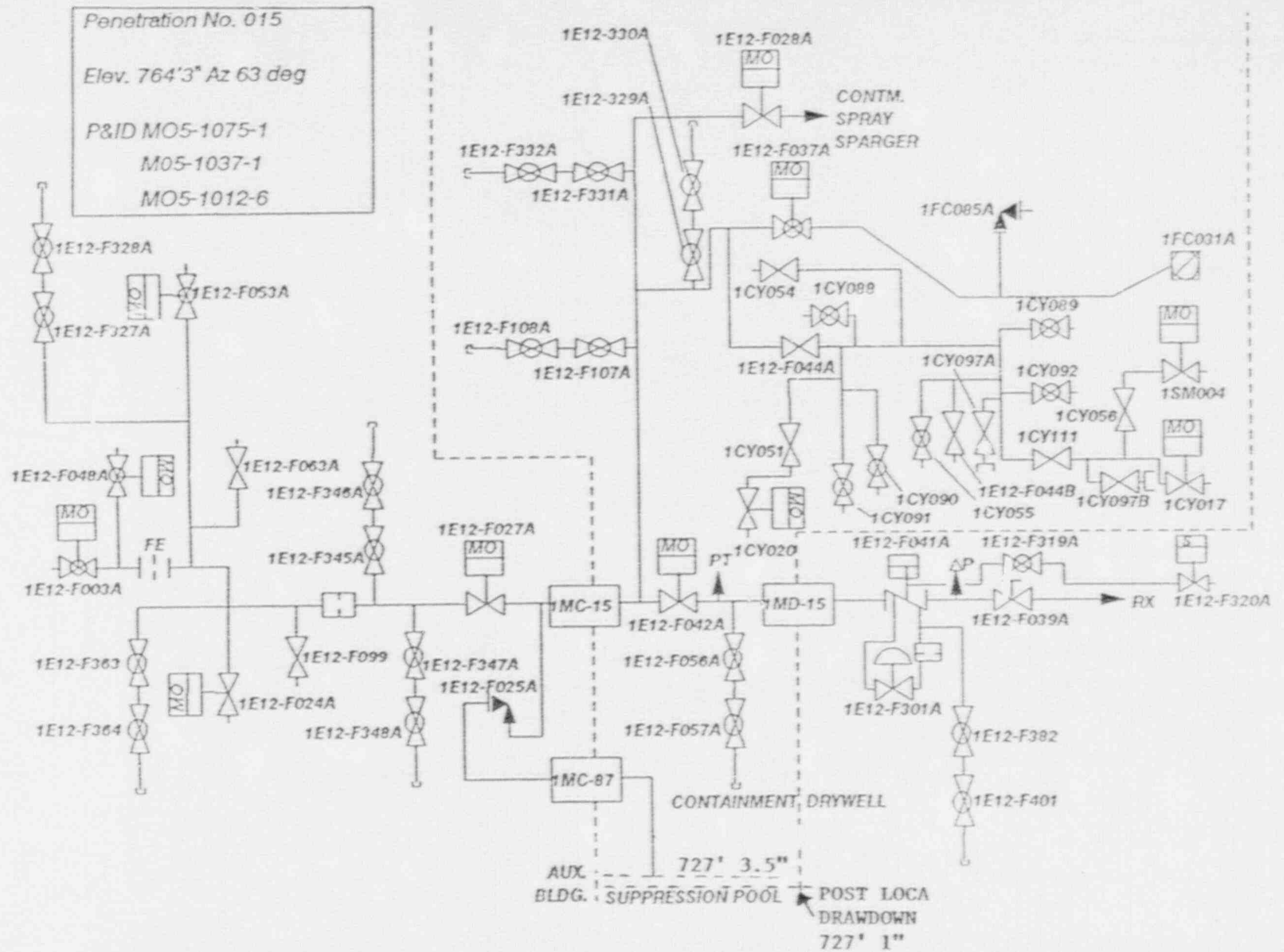

FIGURE 6