NRC Form 366 (9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES 8/31/85

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Haddam Neck	0 5 0 0 0 2 1 3	1 OF 1 0

TITLE (4): Post-LOCA Release Paths Outside Containment

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

OPERATING MODE (9): 1

POWER LEVEL (10): 1 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)	[X] 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)	[ ] 50.73(a)(2)(i)	[ ] 50.73(a)(2)(viii)(A)	
[ ] 20.405(a)(1)(iv)	[ ] 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)
Pierre L'Heureux, Assistant Engineering Supervisor	2 0 3 2 6 7 - 2 5 5 6

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) [X] NO

EXPECTED SUBMISSION DATE (15): MONTH | DAY | YEAR

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

It has been discovered during a review of past plant modifications that a design change implemented in 1980, which removed several containment isolation valves from the automatic actuation logic, could, under certain circumstances, provide a release pathway to the environment which is not considered in the design basis accident analysis. This has been determined to be reportable pursuant to 10CFR50.73(a)(2)(v), since the removal of these valves from containment isolation could prevent the containment from adequately performing its safety function of controlling the release of radioactive material.

During this review, it was also identified that administrative controls, including technical specifications limiting primary coolant leakage outside containment, did not specifically include charging system outleakage. Such leakage, if excessive, could cause the 10CFR100 limits to be exceeded.

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

1. Background

Following the issuance of the TMI lessons learned requirements in 1979, Connecticut Yankee Atomic Power Company (CYAPCO) evaluated the desirability of retaining the capability to operate the reactor coolant pumps during or following an accident. At that time, the plant design was such that isolation valves associated with the reactor coolant pump auxiliaries (i.e., seal water return, thermal barrier and oil cooler returns) would automatically shut upon receipt of a safety injection or containment isolation signal. These auxiliaries are essential for sustained reactor coolant pump operation. In 1980, CYAPCO determined that operation of the reactor coolant pumps following an accident should be an option available to the operator. A plant design change was initiated to remove seven (7) trip valves from the automatic actuation logic. This design change (Plant Design Change Request - 380, "RCP Component Cooling Water and Seal Water Return Isolation Modification") was implemented during the spring 1980 refueling outage.

CYAPCO described these modifications in a letter to the NRC dated May 23, 1980. These modifications were proposed to address a concern expressed by the Staff in a May 7, 1980 letter to CYAPCO documenting the results of the Staff's evaluation of the implementation of the NUREG-0578 "Category A" requirements at the Haddam Neck plant.

As a result of the failure of the refueling cavity seal in August 1984, the NRC on December 13, 1984, ordered CYAPCO to conduct an independent review of all plant design changes implemented at the Haddam Neck plant since January 1, 1979. CYAPCO's January 28, 1985, response to the Order Modifying License described the scope and conduct of the independent review that would be performed by the Connecticut Yankee Plant Design Change Task Group (CYPDCTG). CYAPCO's proposal was accepted by the NRC in a letter dated April 23, 1985.

One of the design changes selected by the CYPDCTG for detailed review was Plant Design Change Request (PDCR) #380. The detailed review of this design change identified two significant concerns which were determined to be reportable pursuant to 10CFR50.73(a)(2)(v). The CYPDCTG notified Haddam Neck plant management of these concerns at approximately 0930 hours on August 2, 1985. These concerns are as follows:

- The removal of four of the seven containment isolation valves from automatic actuation resulted in a potential release pathway which, under certain conditions, may bypass the containment.

- The current technical specifications restrict the leakage from the residual heat removal (RHR) system to three (3) liters/hr., in accordance with the assumptions of the design basis accident analysis. However, leakage from the charging system is not limited by technical specifications or any other administrative controls. Such charging system outleakage is not considered in the design basis accident analysis.

Each of these concerns is described in detail below.

II. Identification of New Release Pathway Resulting From PDCR 380

When the charging system is running and aligned to provide seal injection flow to the Reactor Coolant Pump (RCP) seals, borated water is injected into the seal area at a pressure higher than Reactor Coolant System (RCS) pressure and this water flows through the thermal barrier into the primary system (Figure A). Some of the seal injection water flows through the #1 seal and returns through a filter to the charging pump suction via the seal water return header (Figure B). In the post-LOCA situation, the system operates essentially the same with the exception that the injection water is drawn from the Refueling Water Storage Tank (RWST) rather than the Volume Control Tank (VCT) and the charging pumps will automatically start only if normal power is available. As long as seal injection continues, the flow of fluid through the thermal barrier is into the RCS, thus no release path of potentially highly contaminated water would exist. If seal injection were to cease however, then RCS fluid would leak cut through the RCP #1 seal and enter the seal water return header.

Prior to the 1980 refueling outage, the isolation valves on the seal water return lines (CH-MOV-311, 312, 313, 314) would automatically close on a safety injection actuation signal and prevent seal return flow from leaving the containment. Since the closure of these valves would prevent sustained operation of the RCP, these valves were removed from the automatic actuation scheme via the implementation of PDCR 380. As a result, a pathway for reactor coolant to bypass containment isolation was created. This potential source of radioactivity was not considered in the accident analysis.

There are two scenarios to consider.

The first scenario would be a loss of coolant accident coincident with a loss of normal power. Since the charging pumps would not automatically start on emergency power, primary coolant would initially leak through the #1 RCP seal and enter the seal water return header. This leak flow would begin to pressurize the seal water return header and charging system piping at a rate dependent upon RCS pressure and the condition of the RCP seals. At greater than 140 psig, the seal water return relief valve (CH-RV-322) will lift, discharging to the VCT, which would have been isolated from the charging pump suction header by the closure of CH-MOV-257. At this point, the control room operator would have received several indications and alarms identifying an RCP seal failure. Specifically, the following alarms would actuate:

low thermal barrier differential pressure
high RCP upper bearing water temperature
high seal water return temperature
high seal water return flow
high VCT temperature
high VCT level
high VCT pressure

The manual closure of the seal water return MOVs (CH-MOV-311, 312, 313, and 314) would terminate the scenario with little or no impact on off-site doses. If the above indications were to be ignored, however, then the VCT would pressurize, lift its 75 psig relief valve, and discharge to the radioactive waste processing system. At this point, there is a potential for a direct gaseous release to the environment, via a 9.5 psig relief valve on the waste gas surge tank.

The second scenario involves the containment sump recirculation phase. At some time after a LOCA when 100,000 gallons have been pumped from the RWST, the RCS will be placed on long-term recirculation via the containment sump. If the RCS pressure is higher than the shutoff head of the RHR pumps, the charging system will be manually aligned to take a suction on the RHR system and discharge to the RCS at a higher pressure. If RCP seal injection were to be supplied in this configuration and if the seal return header isolation valve CH-TV-334 were to be manually closed or fail closed (upon loss of control air), the seal water return header would again pressurize to the point of safety relief at 140 psig.

Evaluation

Regarding the first scenario, the current LOCA emergency operating procedure (EOP 3.1-4, Loss of Coolant) requires that seal injection be maintained, if possible subsequent to a safety injection signal. As long as a centrifugal charging pump is in operation, the seal water return relief valve is not expected to lift. The current RCP seal failure abnormal operating procedure (AOP 3.2-17, Failure of an RCP Seal) requires that the seal water return MOVs be shut within five minutes of indications of seal failure. In order for a release pathway to occur, all of the following must occur:

1. A LOCA occurs coincident with a loss of normal power, in which reactor coolant system pressure is sustained above 140 psig.

2. Neither charging pump is placed in operation manually.

3. The charging system is not providing seal water injection.

4. The operator does not manually isolate the seal water return lines upon determining that the charging system is not providing seal injection.

5. Return flow from the RCP seals is sustained long enough to pressurize not only the seal water system piping and return header, but also the charging pump suction and discharge headers to 140 psig.

6. Upon receiving VCT high level, pressure, and/or temperature alarms, the operator does not manually isolate CH-MOV-311, 312, 313, and 314.

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

7. Upon receiving indications and alarms of failure of an RCP seal, the operator does not manually isolate the seal water return lines.

Regarding the second scenario, the operator would receive the high VCT pressure and level alarms identifying a diversion of seal water return flow. In addition, the trip valve position indicating lights would alert the operator to the source of the problem (CH-TV-334 shut). Shutting the seal water return MOVs would still effectively stop the diversion.

The root cause of this situation was the failure to identify the creation of this release pathway during the technical and safety reviews of PDCR 380.

Corrective Action

Immediately following identification of the problem, the following corrective action was taken:

- The operating personnel were alerted of the possibility of an inadvertent radioactive release under certain conditions, post-LOCA.

- The emergency operating procedures were reviewed to ensure that the seal water return isolation MOV's are required to be closed during a RCP seal failure and that seal injection is required to be maintained post-LOCA.

- An evaluation was performed to determine the probability of a LOCA scenario coupled with significant fuel failure, no seal injection and no manual isolation. The results of this evaluation showed this scenario to be a low probability event.

While the existing configuration is clearly not desirable for the long term, we believe that based upon the low probability of the above scenario, continued operation is justified.

Subsequent to the submittal of LER 85-017-00, the following longer-term corrective action has been or will be taken:

- A conceptual design has been prepared to provide seal water return header relief protection within the containment. As described in the reference, a schedule for implementation will be determined based on further evaluation and analysis.

- Emergency operating procedures have been revised to require isolation of the seal water return header by closing the isolation motor-operated valves (CH-MOV- 311, 312, 313, 314) whenever the Reactor Coolant Pumps are secured subsequent to a safety injection signal.

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As far as addressing the failure mechanism (inadequate review), the implementation of new Nuclear Engineering and Operations procedures in late 1984 upgrades the level and quality of the design review process associated with plant modifications when compared to that in effect in 1980.

III. Additional Pathway

While investigating other potential post-LOCA release paths, it was discovered that the RCS drain header relief valve (CH-RV-1847) could, under certain conditions, also provide a release path outside containment. The relief valve (lift setpoint: 150psig) is normally isolated from the RCS by DH-MOV-310, however if the normal CVCS letdown and charging pathways were to be unavailable, an alternate letdown path via DH-MOV-310 would be established per Normal Operating Procedure 2.6-2. If, in this lineup, a safety injection signal were to be received, the containment isolation valves downstream of the relief valve would close and the relief valve will lift, discharging primary coolant into the Volume Control Tank. This configuration is the original plant design.

Evaluation and Corrective Action

Since the alternate letdown lineup is an extremely unusual event, the probability of this scenario is even less likely than that associated with the seal water return relief valve. Nonetheless the following actions have been or will be taken:

- A conceptual design has been prepared to provide drain header relief protection within the containment. As described in the reference, the schedule for implementation will be determined based on further evaluation and analysis.

- Emergency operating procedures have been revised to require isolation of the RCS drain header by closing DH-MOV-310 upon receipt of a safety injection signal.

Note that during normal operation DH-MOV-310 is shut and the power supply circuit breaker to the motor operator is locked open.

IV. Consideration of Charging System Outleakage and System Integrity Requirements

During the review of PDCR 380 as described above, a related though different concern was identified. This concern relates to the fact that the radiological evaluation of a LOCA for Haddam Neck considers only the Residual Heat Removal (RHR) system as a source of radioactive release outside containment during sump recirculation. Since the charging system may be used in a high head or two-path recirculation mode, it could be circulating highly radioactive water. Thus, such potential outleakage from the system could result in a measurable off-site dose. Leakage from various charging system components (e.g., valves, flanges, pumps, etc.) should be addressed in a manner similar to the RHR system. Such an approach is consistent with the provisions of Standard Review Plan, Section 15.

On March 21, 1978, CYAPCO proposed several changes to the Haddam Neck Plant Technical Specifications. These changes were the result of a reanalysis of the radiological consequences of a loss of coolant accident. One of the proposed changes (Specification 3.14) would reduce the allowable leakage rate from the RHR system from six liters per hour to three liters per hour. Neither this analysis nor the revised analysis performed by CYAPCO for the Systematic Evaluation Program considered leakage outside containment other than the three liters/hour assigned to the RHR system.

The three liter per hour limit on RHR leakage is currently controlled by an Administrative Technical Specification. However, all plant systems which contain radioactive liquids are routinely inspected for leakage. Since January 1984, piping systems carrying radioactive fluid inside the Radiological Control Area have been inspected on a once-per-month basis for the purpose of detecting system leakage. Each year for the past five years, inservice inspection, as required by Technical Specification 4.10, (Inservice Inspection and Reactor Vessel Surveillance) has been performed consisting of hand-over-hand walkdowns of entire safety-related systems. Any signs of leakage are noted and corrective action initiated. Additionally, operator rounds are performed in most plant areas once per shift. Continuous radioactive air monitoring equipment would also detect a sizeable leak. Any significant amounts of leakage would likely be noticed and corrected.

The integrity of fluid systems outside containment was also reviewed in response to NUREG-0578, Item 2.1.6.a. This review included an assessment of leakage from the charging system. The inspections performed for Item 2.1.6.a resulted in an estimated leakage of less than 1000 milliliters per day, as described in a letter from W. G. Counsil to H. R. Denton dated January 31, 1980.

To assess the significance of leakage from the charging system, CYAPCO reviewed the results of the inservice inspection walkdowns performed on the charging system for the past five years. Analysis of these results indicates that charging system leakage has been maintained well below one liter per hour. A recent (August 11, 1985) inspection has verified that the current charging system leakage is less than 0.1 liter/hour. CYAPCO also reviewed surveillance test results over the last five years for the RHR system. For most monthly surveillances, the RHR leakage rate was zero. Measurable leakage was found on only eight occasions and this leakage averaged approximately 0.14 liters per hour.

Evaluation

Based on the results of past system leakage inspections, CYAPCO is reasonably confident that the combined RHR and charging system leakage outside containment has been historically within the three liters per hour assumed in the accident analysis. Therefore, the dose consequence assessment remains valid and continued operation is justified.

The root cause of this situation was the omission of this radioactive source in the original (and subsequent) post-accident dose calculations.

Corrective Action

Subsequent to the submittal of LER 85-017-00, the following longer-term corrective action has been or will be taken:

- o Administrative controls have been implemented to monitor and limit the combined charging and RHR system leakage to 3 liters/hour.

- o A license amendment request will be submitted to the NRC under separate cover to limit the combined charging and RHR system leakage to 3 liters/hour in accordance with the reference.

As far as addressing the root cause (inappropriate assumptions in the design basis calculations), implementation of the new Nuclear Engineering and Operations Procedures since the calculation was performed in 1978 have improved the level of detail and extent of controls and reviews required for design basis calculations and safety analysis. Additionally, the post-cavity seal failure philosophy of stressing the "what if" questions have increased the scrutiny of assumptions made in safety-related calculations and documents.

Reference: J. F. Opeka letter to T. E. Murley dated November 6, 1985, "Response to December 13, 1984 order modifying license".

NRC Form 366A
(9-83)

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

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FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Haddam Neck	0 5 0 0 0 2 1 3	8 5	— 0 1 7 —	0 1	0 9 OF 1 0

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

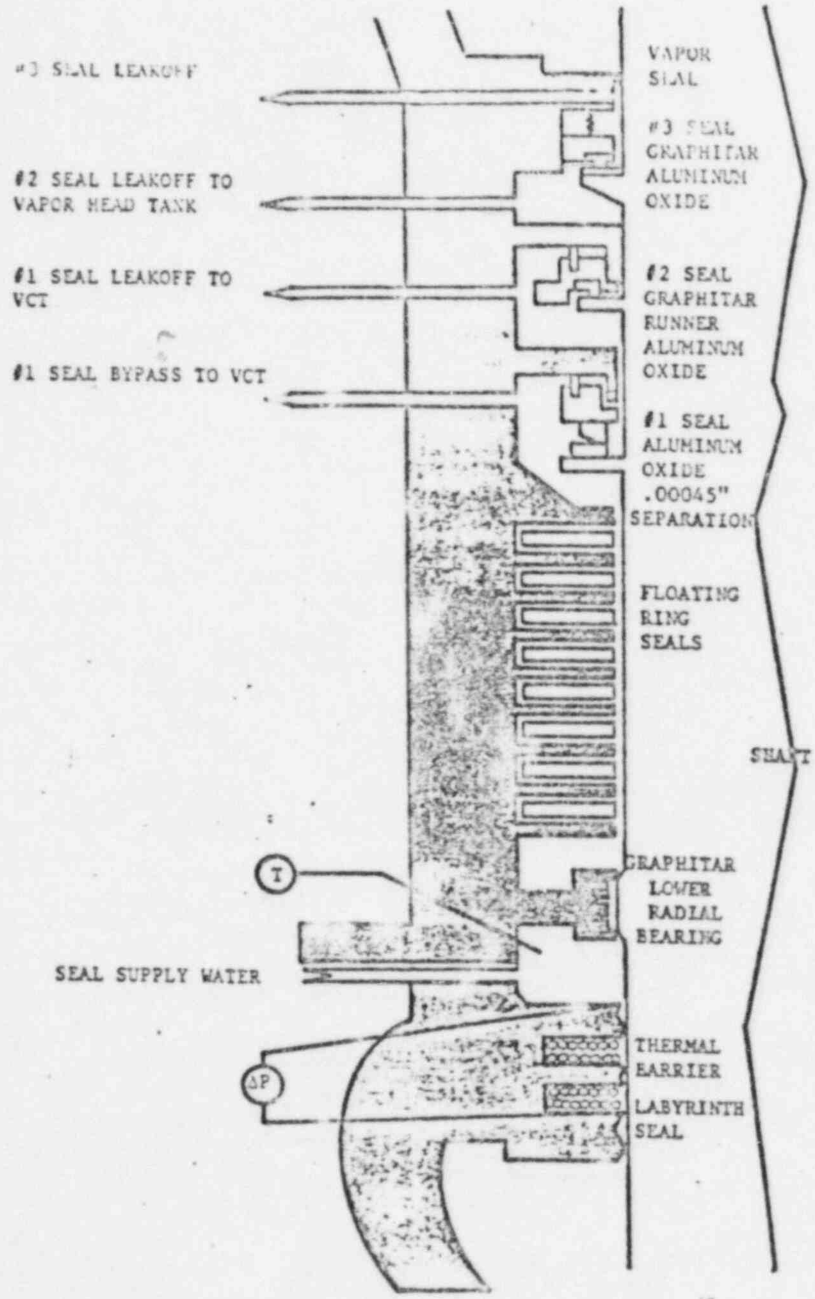

FIGURE A
RCP SEAL PACKAGE

NRC Form 366A (9-83)

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

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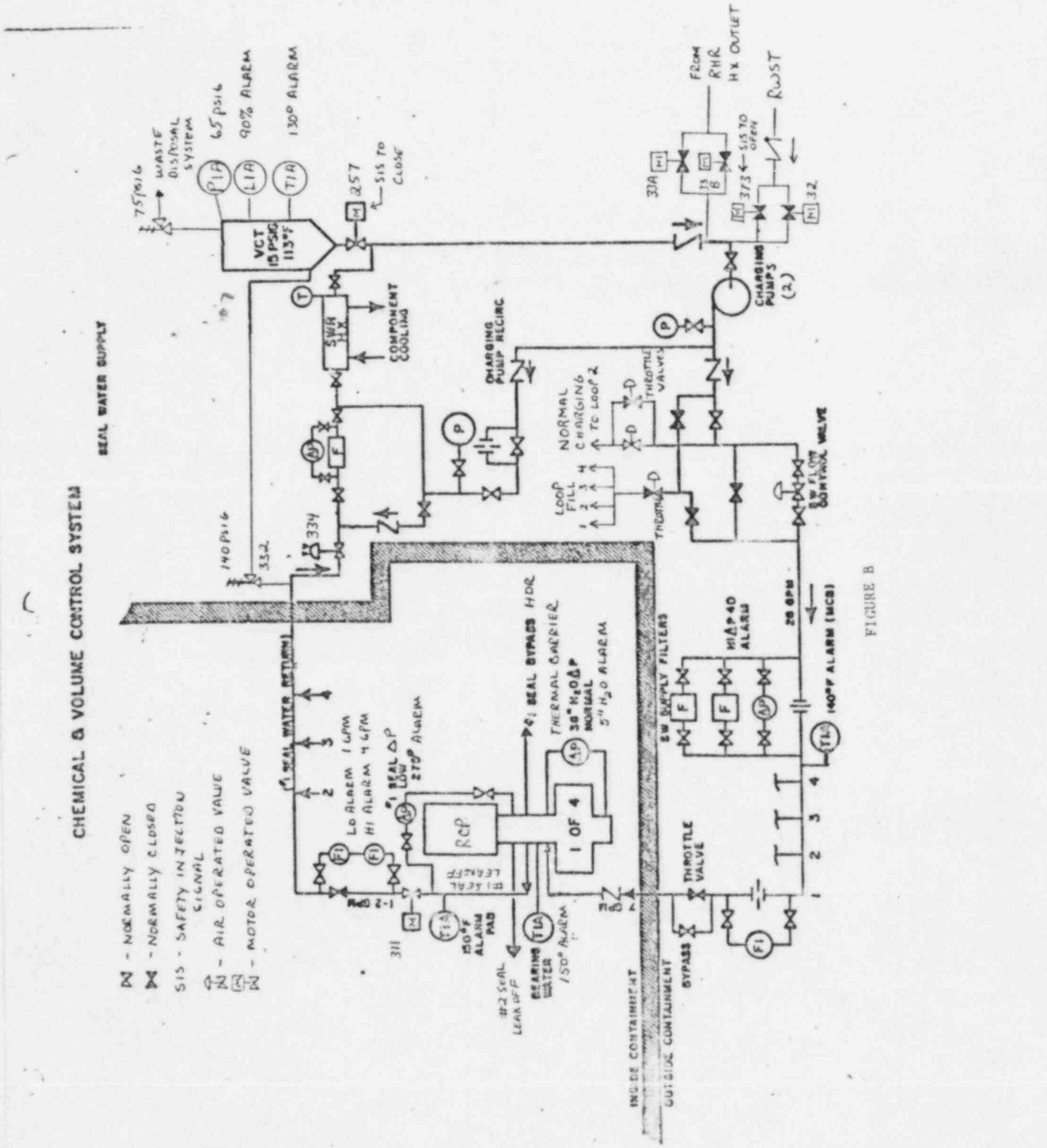

FIGURE B

CONNECTICUT YANKEE ATOMIC POWER COMPANY
HADDAM NECK PLANT
RR#1 • BOX 127E • EAST HAMPTON, CONN. 06424

November 8, 1985

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

Reference: Facility Operating License No. DPR-61
Docket No. 50-213
Reportable Occurrence LER 50-213/85-017-01

Gentlemen:

This letter forwards supplemental Licensee Event Report 85-017-01, which provides additional information concerning corrective action.

Very truly yours,

John H Ferguson
for Richard H. Graves
Station Superintendent

RHG:PFL/kak
Attachment: LER 85-017-01

cc: Dr. T. E. Murley, Region I

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