ENCLOSURE 3

PORV

FAILURE REDUCTION METHODS

FINAL REPORT

PREPARED FOR THE C-E OWNERS GROUP

NUCLEAR POWER SYSTEMS DIVISION

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1. BACKGROUND

The failure of a power-operated relief valve (PORV) to close subsequent to its actuation during an overpressure condition was a key factor in the Three Mile Island-2 (TMI-2) accident. As a result, the operating history of PORVs on all operating light water reactors (LWRs) was investigated by the Nuclear Regulatory Commission (NRC). On an overall basis, the results of the investigation indicated that the probability of a small break loss of coolant accident (LOCA) due to the failure of a PORV to close appeared to be a major contributor to the total probability of a small break LOCA from all causes.[1] Consequently, the NRC has requested[2] that methods for PORV failure reduction be evaluated by C-E for possible implementation to increase plant safety.

2. PURPOSE

The purpose of this study is to review PORV failures, to evaluate methods for failure reduction, to describe the plant changes made or recommended to reduce PORV failures, and to evaluate the effectiveness of these changes for C-E operating plants.

3. DESCRIPTION OF PORV SYSTEM

3.1 Introduction

A brief description of the provisions for overpressure protection of the typical C-E Nuclear Steam Supply System (NSSS) primary coolant system and clarification of the supporting role of the PORVs is provided below.

Overpressure protection for the primary coolant system is based on the combined action of the primary safety valves, secondary safety valves, and the reactor protection system. At operating conditions the PORVs are not formally part of the overpressure protection system; although the presence of PORVs increases the primary coolant system relieving capacity.

3.2 Function of the PORV

To reduce the number of challenges to the primary safety valves, and thus reduce the probability of gross safety valve leakage or weeping, pressurizers on all C-E operating plants (except for ANO-2) are provided with two PORVs having actuation set points below that of the primary safety valves.

Figure 1 shows a typical installation arrangement for primary system over-pressure protection. Isolation valves are provided upstream of each PORV. Throughout this report, the term "PORV System" is used whenever the PORV and its isolation valve is being considered in combination. Design and operating parameters for the primary safety valves and PORVs at C-E operating plants are given in Table 1.(1)

Additional functions, not considered in the initial NSSS design, have sin[ce] been assigned to the PORVs. These functions include low temperature over-pressure protection, venting, and long term cooling subsequent to a LOCA. These auxiliary PORV functions have been documented elsewhere and are not included in the scope of this report.

3.3 PORV Design Basis

The PORVs are designed to have an opening setpoint pressure below that of the primary safety valves and to provide sufficient relieving capacity to ensure that the primary safety valves do not lift or weep during over-pressurization transient conditions such as uncontrolled rod withdrawal, loss of load, or loss of all non-emergency AC power. The PORV opening setpoint pressure is sufficiently high to ensure that the PORVs do not open in response to normal maneuvering transients.

3.4 PORV Description

All PORVs in operating C-E NSSSs are Dresser electromatic relief valves which are pilot actuated, reverse-seated, and which use pressurizer pressure to operate the valve (Figure 2). When pressurizer pressure exceeds the valve setpoint pressure, the solenoid on the pilot valve is energized; this causes its plunger to actuate a lever to open the pilot valve. The main valve's pressure chamber above the valve disc is vented through the open pilot valve and the resulting pressure difference across the main valve disc causes the main valve to open and discharge pressurizer fluid. When pressurizer pressure decreases below the setpoint value, the solenoid is deenergized, the pilot valve closes, and steam pressure builds up in main valve pressure chamber and forces the valve disc closed.

3.5 PORV Operation

The PORVs are designed for automatic or manual operation. In automatic operation, the PORVs are opened by the high pressurizer pressure trip signal in the reactor protective system, which is actuated by a two out of four channel logic system. The PORVs, which are actuated by the same bistable trip units which actuate the reactor trip, open whenever the pressurizer pressure exceeds the high pressure reactor trip setpoint and they remain open until pressurizer pressure falls below the valve reset pressure. In the manual mode the PORVs can be operated independent of system temperature and pressurizer pressure.

The PORV actuation setpoints vary somewhat from plant to plant, at a nominal value of approximately 2400 psia, about 100 psi below the primary safety valves setpoint and 150 psi above normal operating pressure (Table 1).

3.6 PORV Isolation Valves

To permit isolation of a PORV in case of excessive seat leakage or failure to close, motor-operated block valves are provided upstream of each PORV. During power operation the block valves are normally open. However, one or both PORVs may be isolated (block valves closed) because of excessive leakage. Also, operation with one PORV isolated may be considered to avoid excessive reactor coolant discharge due to both PORVs lifting.

3.7 PORV Leakage Detection

Several methods were used prior to the TMI accident for the detection of excessive PORV leakage or failure to close. These methods include monitoring PORV discharge piping temperature, PORV pilot valve position indication, and quench tank pressure, temperature, and level. Readouts from each of these measurements are generally available in the plant main control room. Subsequent to the TMI-2 accident, the NRC required a reliable, direct means for PORV position indication. Action to respond to this requirement is described in Sections 6 and 7.

3.8 Electric Power Supplies

In performing their function to reduce the frequency of primary safety valve challenges, the PORVs provide equipment protection and as a consequence, are not considered as part of the plant safety system. Therefore, the valves as installed in the field were not provided with safety grade power sources and no credit was taken for their operation in safety analyses. Subsequent to the TMI-2 accident, consideration was given to providing the PORVs and their isolation valves with emergency power sourc[es]. Further actions on PORV system power supplies are discussed in Sections 6 and 7.

3.9 Comparison with Other PWRs

The PORV systems provided in pressurized water reactors (PWRs) supplied by Babcock and Wilcox (B&W)(3), Westinghouse (W)(4) and C-E differ in details such as the type, number, capacity, setpoint, valve vendors and control circuitry. Certain important differences among the PWR vendors' systems are described in the following sections.

On C-E plants, the initial design function of the PORVs was solely to reduce the challenges to the primary safety valves during power operation. The PORVs on B&W and W plants had an additional function, namely, to reduce the frequency of reactor trips due to high pressure. The PORV actuation set point on C-E plants coincides with the high pressure reactor trip setpoint, whereas, the other PWR vendors required that the PORV actuation pressure be below the high pressure reactor trip setpoint in order to reduce the number of high pressure trips. The C-E design allows the specification of a higher PORV actuation pressure, and therefore a greater margin above the normal plant operating pressure than do the othe[r] PWR designs. Typically, the margin between normal operating pressure and the PORV actuation setpoint was about 150 psi for C-E plants, 100 psi for W plants, and 70 psi for B&W plants. This difference provided an incremental margin to PORV challenges in C-E plants compared with those of the other PWR vendors.

The B&W plants are equipped with the same type of PORVs as those of C-E, namely, the Dresser electromatic solenoid pilot-operated valve described in Section 3.4. The majority of W plants use Copes-Vulcan spring-loaded, air-operated valves. Air pressure on the control diaphragm overcomes the spring force to open the valve. Venting the air pressure from the control diaphragm allows spring force to close the valve. A few W plants use PORVs manufactured by Masoneilan (3 plants), Dresser (1 plant), ACF Industries (1 plant), and Control Components (1 plant).

4. PORV OPERATING EXPERIENCE

4.1 Combustion Engineering Plants

The operating experience of PORVs in C-E plants has been compiled in Table 2 based on information supplied by the various plant operators during a survey conducted in early 1980. The PORV actuations noted in Table 2 do not necessarily represent the total number which have occurred, since PORV actuations were not reportable events and were not routinely recorded. Therefore, some actuations may have been overlooked. Also, since the available means for the detection of PORV actuation was not direct, but generally dependent upon an integrating effect, such as increasing quench tank level, for example, some actuations may have gone undetected.

Table 3 is a tabulation of high pressurizer pressure reactor trips occurring in C-E operating plants for which PORV actuations were not reported. The data was obtained from a review of published data, mainly from the NRC. Since, by design, a high pressurizer pressure reactor trip should be accompanied by PORV actuation, it is inferred that the actuation did occur, though it was not reported.

Table 2 indicates a total of seven confirmed PORV actuation events. Four events occurred during PORV testing or system maintenance. In two of these events the PORVs failed to close satisfactorily. The remaining three actuation events occurred during power operation, with the PORVs operating satisfactorily in each case. Table 3 indicates a total of sixteen high pressurizer pressure reactor trips, eleven of which resulted from turbine runbacks. Tables 2 and 3 extend the PORV actuation data presented in NUREG 0635[1].

It was inferred that the high pressurizer pressure trips listed in Table 3 were accompanied by PORV actuations. Combining the confirmed PORV actuation events during power operation listed in Table 2 with the inferred actuation events from Table 3, a total of nineteen events or thirty-eight PORV challenges is obtained, with no failures being reported. A total of about 29 reactor-years of operation is covered by this data.

The two PORV failures-to-close on C-E plants listed in Table 2 occurred during maintenance or testing.

The Palisades incident occurred when the Reactor Protection System (RPS) was deenergized for maintenance, which caused the PORVs to open. Due to an ambiguity in the pertinent wiring diagrams the technician failed to perceive that his action would cause PORV actuation. The spring-return-to-Auto feature of the PORV selector switch contributed to the incident since the selector switch could not be retained in the "Manual" mode and "Shut" position unless held there by the operator. Corrective action was taken to clarify the pertinent wiring drawings and eliminate the spring-return-to-Auto feature of the PORV selector switch. The PORV failure-to-close in this instance was not due to the failure of the valve.

The second PORV failure-to-close occurred in Calvert Cliffs #1 during valve operational testing following valve maintenance. The valve failed to shut completely. Modified replacement parts had been installed in the valve because original replacement parts were unavailable due to vendor upgrading of the valve design. Following adjustment of the pilot valve stroke, satsifactory valve closure was obtained.

4.2 Experience at Other PWRs

Westinghouse PWRs in the U.S. have not reported any PORV failures[4], but since they are equipped with a different type of PORV their reliability experience is not relevant to C-E PORVs.

It has been estimated that in B&W plants there have been approximately 150 actuations of PORVs[3] with six cases of failure-to-close properly. One failure occurred during low power testing upon loss of a vital bus, another during startup testing due to improper venting, and a third was a leaky valve. Three failures occurred during power operation, giving approximately 3/150 = .02 failures per demand.

5. PRIMARY SAFETY VALVES

5.1 Operating Experience

No primary safety valve lifts have been reported for C-E operating plants during approximately 30 reactor-years of operation. Westinghouse plants also have not reported any primary safety valve lifts. One primary safety valve lift has been noted[4] in a B&W plant, but no details were given. In view of the lack of challenges to the primary safety valves, a direct quantitative estimate of their reliability based on experience cannot be made.

5.2 Probabilistic Analysis

The main steam safety valves (MSSV) are much more subject to challenges than are the primary safety valves, so that data regarding their reliability has been developed. This data does not have direct applicability to the primary safety valves since, even though the MSSV bears some similarity to the primary safeties, there are distinct differences with respect to service conditions, materials, and other design features. Lacking data on the primary safety valves, the MSSV data may provide some indication of primary safety valve reliability.

A study of PWR MSSV operating experience up to May, 1978 was performed by C-E. The data sources used were NPRDS Failure Report Summaries, License Event Report Summaries, and Operating Units Status Reports.

The period reviewed included 137 reactor-years of operation at 38 PWR plants with an estimated population of 570 MSSVs. During this period there were an estimated 2070 MSSV test demands (pre-operational and annual). Assuming one demand on MSSVs for every ten scrams or turbine trips, about 2580 operational MSSV demands were estimated. The total number of MSSV demands in the study period were estimated to be 5650.

During this period two events were reported (none from C-E operating plants) in which MSSVs failed to close following a demand. The first event occurred at Turkey Point Unit 4 in 1974 when a missing cotter pin caused one MSSV to fail open. The second event occurred at Three Mile Island Unit 2 in April, 1978. A common mode failure of six MSSVs to close occurred due to cocked sleeves in the bellows assembly. Thus, the total number of MSSV failures to reseat reported during the study period was seven.

Based on the seven reported MSSV failures and the 5650 estimated MSSV demands, a failure rate of 1.24×10^{-3} per demand is estimated. This failure rate is lower than the value of 2×10^{-2} estimated for power operated relief valves in NUREG 0560.[3] Assuming that the MSSV reliability data are to some degree applicable to the primary safety valves, the data suggests that the primary safety valves may be more reliable than the PORVs. More definite conclusions must await development of operational and/or test data on primary safety valves.

6. METHODS FOR REDUCING PORV SYSTEM FAILURE

6.1 Reduction of PORV Challenges

The frequency of PORV system failures can be reduced by decreasing the fre-quency of challenges to the PORVs. These reductions must be made without adversely impacting safety or incurring unacceptable economic or performance penalties. Methods for potentially decreasing the frequency of PORV challenges on C-E plants and a brief summary of their impacts on the plant are provided below.

6.1.1 Raise PORV Setpoint

High pressurizer pressure trips the reactor when the pressure exceeds the trip setpoint pressure and the output from the same bistable comparator also actuates the PORV. Therefore, only one setpoint is available. Raising this Reactor Protection System (RPS) high pressurizer pressure reactor trip setpoint would invalidate the safety analysis and increase the challenges to the primary safety valves.

6.1.2 Lower High Pressurizer Pressure Trip Setpoint

This requires the concomitant lowering of the PORV actuation setpoint as described above. Doing so would increase the number of challenges to the PORVs.

6.1.3 Raise the setpoint for the existing PORV Opening/High Pressurizer Pressure Trip and Add Another High Pressurizer Pressure Reactor Trip at 2400 psi

The setpoint for the existing PORV Opening/High Pressurizer Pressure Reactor Trip would need to be raised approximately no higher than 20-40 psi to prevent primary safety valve challenges during a full loss of turbine load without a simultaneous reactor trip while simultaneously precluding PORV openings during milder pressure increases. The benefits of this alternative would be very small since only a very small fraction of the PORV openings would have been avoided by this modification (i.e., full load rejection where PORV opening was desired to preclude primary safety valve opening and the inadvertent initiations would not have been affected).

Further, there is no more room in the protective system cabinetry in some of the operating plants to accommodate additional bistable trip units and other circuitry that would be required. Adding additional trips would be expensive and would take a considerable amount of time to incorporate.

6.1.4 Block Out and/or Deactivate PORV During Power Operation

In the event of a full power incident which causes the turbine admission valves to close rapidly (e.g. full load rejection, electrical system over-frequency, turbine control failure), the reactor would trip on high pressurizer pressure in the absence of a turbine trip signal. The pressurizer pressure would continue rising above the 2400 psi setpoint until the reactor trip quenched the power output of the core and caused the pressurizer pressure to decrease. It is prudent to use the power operated relief valves to preclude challenging the primary safety valves during this transient. There are PORV block valves which can be closed in the unlikely event of a PORV failing to close. Such block valves are unavailable to mitigate the consequences in the unlikely event that a safety valve fails to reclose.

6.1.5 Reduce Operating Pressure

A reduction in operating pressure would tend to reduce the number of PORV openings, but by only a small proportion. Also, the lower the operating pressure, the higher the overshoot in pressure after a load rejection is terminated by the high pressurizer pressure trip. The higher overshoot in pressure results from the delay in the reactor trip. This increases the potential for challenging the primary safety valves. More importantly, decreasing the primary operating pressure would decrease the operating DNB ratio thus causing the core to be operated closer to one of the safety limits.

6.1.6 Elimination of Turbine Runback

Table 3 indicates that a relatively large number (11) of high pressure trips (and presumably 22 PORV actuations) occurred during turbine runback events. A review of this plant feature indicated that its elimination would not adversely affect plant operation, while at the same time reducing PORV challenges to a significant degree.

6.2 Improved Capability for Countermeasures

The frequency of PORV system failures can also be reduced by improving the capability for appropriate countermeasures (PORV isolation) subsequent to a PORV failure to close. Methods for potentially improving the capability to take appropriate action and a brief summary of their impacts on the plant are discussed.

6.2.1 Automatically Close Block Valve Whenever PORV Fails to Close on Command

There are several ways this could be implemented. The block valve closing signal could be armed by an initial PORV opening signal so that the block valve would remain open in normal operation but would be automatically closed if the PORV failed to close on command. Another approach would use the concurrence of an open PORV valve and and PORV valve closure command to automatically close the block valve. Although automatic valve closure would remove the requirements for operator action upon PORV failure, the additional control circuitry would introduce additional complexity to the system and would itself be subject to its own failure modes. These schemes require further detailed evaluation to determine their positive and negative impacts on overall plant safety. A simpler approach is to assure that the operator is able to utilize existing inplant instrumentation to identify a stuck-open PORV and to close the block valve..

6.2.2 PORV Position Indication

Reliable and positive control room indication of PORV position would provide vital information to the operator in a clear and timely manner to permit him to take the appropriate action necessary to prevent escalation of a minor incident into a LOCA. An ultrasonic flowmeter, located at the discharge piping of the PORV, with flow indication and alarm in the control room, would provide direct, positive, rapid-response, and reliable indication of PORV position. An advantage of this instrument is that it does not require any penetration of the piping. Alternatively, the PORV could be provided with a position indicator for the main valve disc position.

6.2.3 Electric Power Supplies

The PORVs and their associated block valves, which were designed for an equipment protective function rather than a safety function, were not initially provided with emergency power supplies. The provision of emergency power to these valves would maintain the availability of the relief system and also permit its isolation, if necessary, upon loss of all non- emergency power sources.

6.2.4 Improvement of Operator Capability

The evaluation of the TMI-2 incident indicated that a program to improve operator performance, particularly during emergency conditions, would significantly reduce the potential for serious nuclear incidents. Upgrading operator capability to recognize and to respond appropriately to a PORV failure-to-close should significantly reduce the possibility of the subsequent occurrence of a small break LOCA.

7. IMPLEMENTATION OF PORV SYSTEM FAILURE REDUCTION PROGRAM

The following actions to reduce PORV system failures have been completed or are pending:

1. The turbine runback feature has been eliminated from C-E operating plants.

2. The motor operators for the PORV block valves and the pilot solenoids for the PORVs have been provided with emergency power supplies to permit them to function upon the loss of all non-emergency power.

3. Ultrasonic flowmeters are being installed on the PORV discharge piping to provide a direct measurement of steam flow and therefore, of PORV position, with indication and alarm in the control room.

4. Operator training programs have been initiated to provide the operator with a more comprehensive understanding of plant operation under emergency conditions. Guidelines and detailed emergency operating procedures have been developed to aid the operator to cope with a spectrum of emergency conditions. This includes the conditioning of the operator to recognize and respond promptly to PORV failure to prevent escalation of the failure to a small break LOCA.

8. ANALYSIS AND RESULTS OF FAILURE REDUCTION PROGRAM

An analysis was performed to provide an estimate of the reliability of the PORV system as well as an estimate of the improvement in reliability expected as a result of the various actions taken or to be taken as noted in Section 7. Appendix A presents a description of the reliability analysis and the results obtained. This section provides a discussion of the analysis and results.

Table A-1 gives challenge frequencies for the PORVs and demand failure rates used in the analysis for various aspects of PORV and block valve operation. The frequency of challenges to the PORVs is based on the C-E operating plants' experience presented in Section (4.1). The PORV demand failure (failure-to-close) rate is based on the B&W operating experience described in Section 4.2. The reasons for using the B&W data as a basis are that:

1. The C-E PORV system design basis and other NSSS features as discussed in Section 3.0 tended to keep PORV actuations to a minimum, so that only a small statistical data base for PORV actuations on the C-E NSSS was available.

2. B&W operating plants had experienced a relatively large number of PORV actuations, and in addition, their operating plants are equipped, with one exception, with the same type of PORVs from the same supplier as are C-E operating plants.

3. Westinghouse operating plant experience was not included due to the fact that, in general, they used a different type of PORV from different vendors than did C-E and B&W.

The specific value of the B&W PORV demand failure rate used in the Appendix A analysis was 0.02 failures-to-close per opening. If the C-E plant experience (38 challenges with zero failures) was statistically combined with the B&W data, the demand failure rate would be reduced by about 20% to 0.016.

A value of 0.155 was used for the probability of failure of the operator to isolate the failed-open PORV. This value is based on data in WASH 1400[5], and is taken as the mean between the operator's normal stress level and severe stress level failure probabilities.

Table A-2 provides the estimated frequency of an unisolated failed-open PORV, (i.e. small break LOCA due to a failed-open PORV) for a C-E plant to which various features have been incorporated. It shows the progressive reduction in the recurrence frequency of a small break LOCA due to a failed-open PORV as the various methods for PORV system failure reduction noted in Section 7 are implemented. Case 1 is the reference case prior to elimination of the turbine runback feature. This case takes no credit for operator action to isolate the failed-open PORV on the assumption that the available instrumentation did not provide clear, positive valve position indication to the operator. Case 2 assumes elimination of the turbine runback feature, with no credit for operator action. Case 3 is similar to Case 2, except credit is taken for operator action on the basis that appropriate instrumentation has been added to give the operator clear, positive indication of PORV position. Cases 4 and 5 assume that provision for automatic closure of the block valve upon failure of the PORV to reclose has been incorporated. Case 4 assumes a control grade design which involves reliable components but has only a single isolation valve and hence is not single failure proof. Case 5 assumes a safety grade design with series isolation valves to provide single failure protection for closure.

The estimates in Table A-2 show that the elimination of the turbine runback feature and taking credit for operator action (based on positive valve position indication and alarms) serves to reduce the estimated recurrence frequency of a small break LOCA due to PORV failure by a factor of about 14.5 (or about 18 for a PORV demand failure rate of .016). The estimated recurrence frequency for a small break LOCA due to a PORV failure is 1.8×10^{-3} per reactor-year (or about 1.4×10^{-3} per reactor-year for a PORV failure demand rate of .016), which is well within the 90% confidence range of a small break LOCA due to a pipe break, 10^{-2} to 10^{-4} per reactor-year, as estimated by WASH-1400. Two factors which would further reduce the recurrence frequency of a small break LOCA due to PORV failure from the value before the TMI-2 accident have not been quantified. One is the improvement in operator capability and reduction in the probability of operator error due to new intensive operator training programs, and the updating of plant emergency procedures based on guidelines which consider the realistic response of the plant to transients and accidents. The second is the provision of emergency power to the PORV block valves to allow PORV isolation, if necessary, after loss of non-emergency power. These factors provide some additional confidence regarding the conservatism of the analytical results.

Table A-2 also shows that provision of control grade automatic block valve closure upon PORV failure to close would reduce the recurrence frequency of a small break LOCA due to PORV failure nearly to the lower limit of the range of 10^{-2} - 10^{-4} per reactor-year estimated for the small break LOCA due to pipe rupture by WASH-1400. The provision of a safety-grade, single-failure-proof design for automatic block valve closure by the addition of redundant isolation valves reduces the recurrence frequency to a negligable value.

9. SUMMARY AND CONCLUSIONS

The C-E operating plants after approximately 29 reactor-years of operation have experienced no PORV failures during power operation. The elimination of the turbine runback feature and the provision of a direct reliable means for indicating PORV position to the operator provided significant improvements in system reliability. The recurrence frequency of a small break LOCA due to PORV failure has been reduced by an estimated factor of about 15 to a value of about 1.8×10^{-3} per reactor-year. This recurrence frequency is well within the 90% confidence range of the recurrence frequencies of 10^{-2} to 10^{-4} per reactor-year for a LOCA due to a small pipe rupture estimated in WASH-1400. Improved operator training programs and emergency procedures, as well as the provision of emergency power to the PORVs and to their block valves, though not quantified, has reduced the small break LOCA recurrence frequency even further. The incorporation of the feature of automatic block valve closure upon PORV failure would further increase PORV system reliability.

10. REFERENCES

1. NUREG-0635 - Generic Evaluation of Feedwater Transients and Small-Break-Loss-of-Coolant Accidents in C-E Designated Operating Plants, January 1980.

2. NUREG-0737 Clarification of TMI Action Plan Requirements, Nov. 1980

3. NUREG-0560 - Generic Assessment of Feedwater Transients in Pressurized Water Reactor Designed by the Babcock and Wilcox Company, May, 1979.

4. NUREG-0611 - Generic Evaluation of Feedwater Transients and Small-Break-Loss-of-Coolant Accidents in Westinghouse Designed Operating Plants, January, 1980.

5. WASH-1400 - Reactor Safety Study, October, 1978 Appendix III, Table III 6-1.

TABLE 1

C-E PRIMARY SAFETY VALVE AND PO[illegible]ATA

A. PRIMARY SAFETY VALVES DATA

Plant	Valve Vendor	Valve Type	Number per plant	Setpoint psig	*Rated Minimum capacity lb/hr	*Maximum Actual capacity, lb/hr
Ft. Calhoun	Crosby	HB-BP-86	2	2530	216,000	240,000
				2485	212,000	236.000
Palisades	Dresser	31739A	3	2565	230,000	256,000
				2525	230,000	256,000
				2485	230,000	256,000
St. Lucie 1	Crosby	HB-BP-86	3	2485	212,000	236,000
Maine Yankee	Dresser	31709KA	3	2535	218,000	243,000
				2510	216,000	240,000
				2485	214,000	238,000
Calvert Cliffs 1 and 2	Dresser	31739A	2	2550	304,000	334,000
				2485	296,000	329,000
Millstone 2	Dresser	31739A	2	2485	296,000	329,000

*Capacity indicated corresponds to 3% accumulation above set pressure

B. PORV DATA

Plant	Valve Vendor	Valve Type	Number per plant	Setpoint psig	*Relieving Capacity lb/hr
Ft. Calhoun	Dresser	31533VX	2	2385	111,000
Palisades	Dresser	31533VX	2	2385	155,000
St. Lucie 1	Dresser	31533VX-30	2	2385	159,000
Maine Yankee	Dresser	31533VX	2	2385	150,000
Calvert Cliffs 1 and 2	Dresser	31533VX-30	2	2385	159,000
Millstone 2	Dresser	31533VX-30	2	2400	148,000

*Rated value at 0% accumulation, provided by vendor

TABLE 2

Summary of Events Involving PORV Operation

PLANT	DATE	PLANT CONDITIONS	INITIATING EVENT	DESCRIPTION
Consumers Power* Palisades	Sept. 8, 1971	Mode 3	Technician deenergized RPS for maintenance	PORV opened when RPS deenergized.
Baltimore Gas & Elec. Calvert Cliffs-1	July 6, 1979	Mode 5	Test of PORV	During operational test of PORV valve failed to fully close. Adjusted pilot valve stroke
2	August 20, 1980	100%	MSIV Closure	PORVs cycled on high pressure
Florida Power & Light St. Lucie -1	Feb. 21, 1977	100%	100% load rejection	PORV cycled during test when reactor tripped on high pressure.
Omaha Public Power Dist. Fort Calhoun	May 28, 1978	80%	Turbine control valve closed	PORV's cycled when plant tripped on high pressure.
Fort Calhoun	Dec. 20, 1978	Mode 5	Troubleshooting pressure recorder	PORV's opened when technician pulled recorder fuses.
Northeast Utilities Millstone-2	Aug. 10, 1979	Mode 5	Troubleshooting	PORV opened on loss of AC to emergency bus.
Maine Yankee Atomic Power Company	No PORV Operation Events			
Maine Yankee				

*Palisades has operated since 1972 with PORV block valve shut.

TABLE 3

Summary of Events Resulting
In Potential Challenge to PORV

PLANT	DATE	PLANT CONDITIONS	INITIATING EVENT	DESCRIPTION
Consumers Power Palisades (Note 1)	Mar. 19, 1973	85%	Circuit Noise	Spurious high pressure trip
	Aug. 31, 1976	100%	MSIV shutting	High pressure trip due to MSIV shutting.
	Nov. 26, 1976	15%	Generator Synchronization	Spurious high pressure trip while bringing generator on line.
	May 22, 1978	100%	Closure of both MSIV	High pressure reactor trip.
Baltimore Gas & Elec. Calvert Cliffs -1	July 8, 1975	100%	Turbine runback	High pressure trip due to turbine runback. Unable to verify PORV operation due to loss of plant computer.
	Jan. 26, 1975	20%	Power reduction with manual pressurizer spray control	High pressure reactor trip.
Northeast Utilities Millstone -2	Apr. 13, 1976	80%	Turbine runback	High pressure reactor trip.
	Apr. 23, 1976	100%	Turbine runback	High pressure reactor trip.
	May 10, 1976	100%	Turbine runback	High pressure reactor trip.
	May 24, 1976	100%	Turbine runback	High pressure reactor trip.
	May 25, 1976	100%	Turbine runback	High pressure reactor trip.
	June 8, 1976	100%	Turbine runback	High pressure reactor trip.
	June 10, 1976	100%	Turbine runback	High pressure reactor trip.
	June 19, 1976	100%	Turbine runback	High pressure reactor trip.
	June 21, 1976	100%	Turbine runback	High pressure reactor trip.
	Aug. 13, 1976	100%	Turbine runback	High pressure reactor trip.

Note 1 - Palisades has operated since 1972 with PORV blocking valve shut.

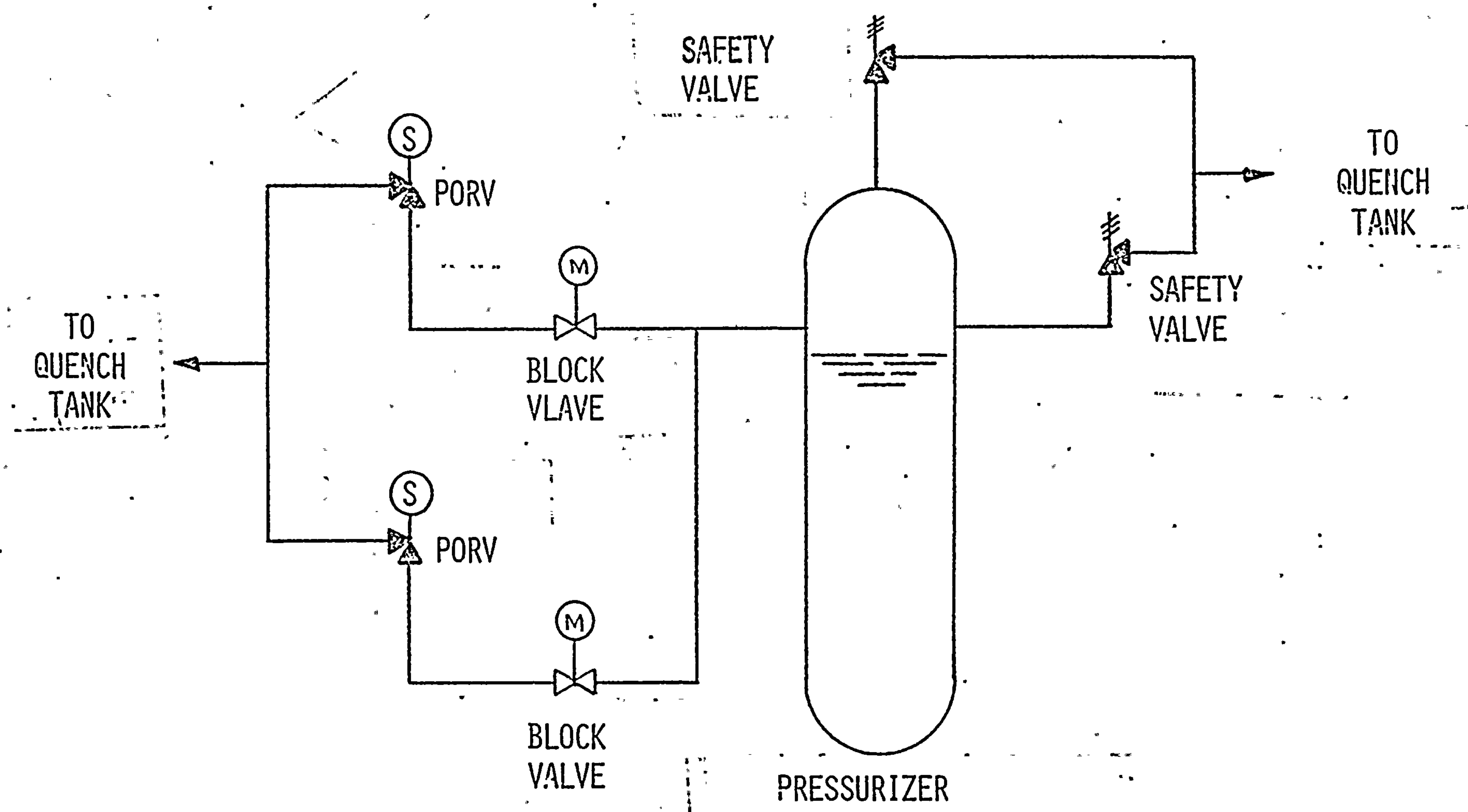


FIGURE 1
TYPICAL PRIMARY SYSTEM
OVERPRESSURE PROTECTION

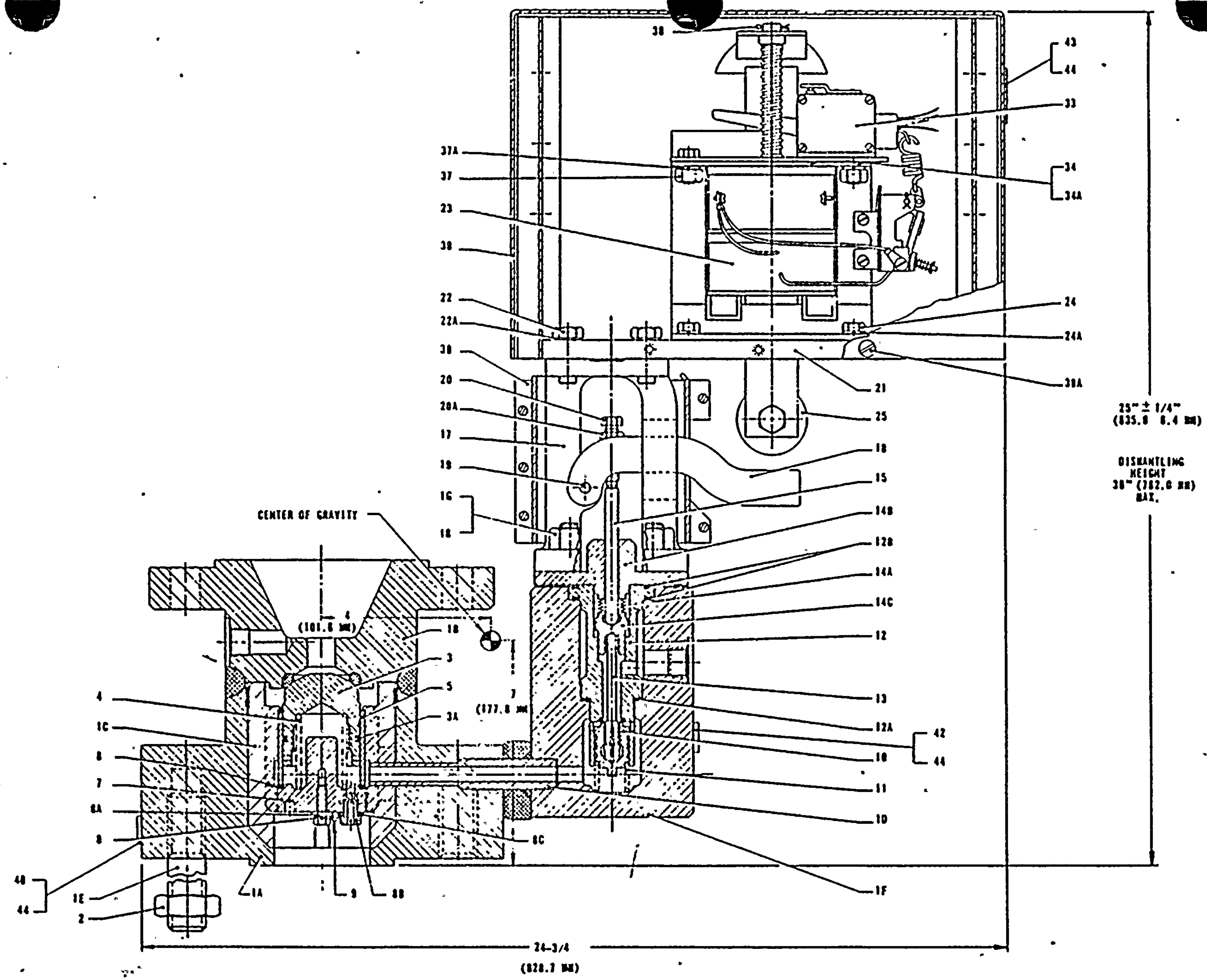


FIGURE 2 - TYPICAL ELECTROMATIC RELIEF VALVE

REF. NO.	QTY.	NOMENCLATURE
1	1	MAIN BASE-PILOT BASE ASSEM. (WELDED, INTEGRAL ASSEM.)
1A	1	INLET FLANGE
1B	1	OUTLET FLANGE
1C	1	CAGE
1D	1	TUBE INSERT
1E	8	MAIN BASE INLET STUD
1F	1	PILOT BASE
1G	4	PILOT BASE STUD
2	8	INLET STUD NUT
3	1	MAIN DISC
3A	1	PISTON RING
4	1	MAIN DISC SPRING
5	1	GUIDE
6	1	GUIDE GASKET
7	1	GUIDE RETAINER PLUG
8	1	RETAINER PLUG CAP SCREW
8A	1	CAP SCREW LOCKWASHER
8B	1	LOCK SCREW
8C	1	LOCK SCREW LOCKWASHER
9	1	SEAL WIRE
10	1	PILOT DISC
11	1	PILOT DISC SPRING
12	1	SEAT BUSHING
12A	1	LOWER GASKET
12B	2	UPPER GASKET
13	1	LOWER SPINDLE
14	1	BELLOWS ASSEM. (WELDED, INTEGRAL ASSEM.)
14A	1	BELLOWS
14B		FLANGE
14C	1	PISTON
15	1	UPPER SPINDLE
16	4	PILOT STUD NUT
17	1	SOLENOID BRACKET
18	1	LEVER
19	1	LEVER PIN ASSEM.
19A	1	SHOULDER SCREW
19B	1	NUT

FIGURE 2 - TYPICAL ELECTROMATIC RELIEF VALVE

REF. NO.	QTY.	NOMENCLATURE
19C	1	BRACKET BUSHING
19D	2	LEVER BUSHING
19E	1	COTTER PIN
20	1	ADJUSTING SCREW
20A	1	LOCKNUT
21	1	BRACKET PLATE
22	4	BRACKET PLATE CAP SCREW
22A	4	LOCKWASHER
23	1	SOLENOID
24	4	SOLENOID CAP SCREW
24A	4	LOCKWASHER
25	1	PLUNGER HEAD
26	1	LEFT HAND SPRING GUIDE
27	1	RIGHT HAND SPRING GUIDE
28	2	PLUNGER SPRING
29	2	PLAIN SPRING WASHER
30	2	SPRING COTTER PIN
31	2	GUIDE BRACKET
32	1	GUIDE BRACKET BOLT
32A	1	LOCKWASHER
32B	1	NUT
33	1	SWITCH
34	2	SWITCH MACHINE SCREW
34A	2	LOCKWASHER
35	3	SPRING GUIDE CAP SCREW
36	1	SPECIAL SPRING GUIDE SCREW
37	4	SPRING GUIDE NUT
37A	4	LOCKWASHER
38	1	BRACKET COVER ASSEM.
38A	1	LEFT HAND COVER
38B	1	RIGHT HAND COVER
38C	5	MACHINE SCREW
38D	5	LOCKWASHER
38E	5	NUT
39	1	SOLENOID COVER
39A	6	MACHINE SCREW
40	1	NAMEPLATE
41	1	TAG PLATE
42	1	CAUTION PLATE
43	1	SOLENOID NAMEPLATE
44	10	NAMEPLATE SCREW

FIGURE 2 - TYPICAL ELECTROMATIC RELIEF VALVE

Sheet 3 of 3

APPENDIX A

C-E ANALYSIS OF

REFERENCE PLANT (SL2) FAULT TREE

FOR

POWER OPERATED RELIEF VALVE

LOSS OF COOLANT ACCIDENT

TABLE OF CONTENTS

1.0 PURPOSE

This report presents the results of a reliability analysis for loss of reactor coolant through the power operated relief valves.

2.0 SCOPE

The reliability analysis considers the performance of the safety function element (SFE) strictly as defined in Sections 3 and 4, Safety Function Element Description and Analysis Assumptions. In this form, the analysis will not be applicable to all initiating events but presents a model which was determined to be most useful in terms of applicability and most amenable to later modification for application to special cases.

3.0 SAFETY FUNCTION ELEMENT DESCRIPTION

The safety function element, Relieving Reactor Coolant System Pressure through the Powered Operated Relief Valves (PORV), refers to the opening of the PORV due to high Reactor Coolant System pressure and reclosing these valves once the Reactor Coolant System pressure decreases below the valve setpoint. Included in this SFE are the opening and reclosing of the PORVs. Also included is the operator's capability to close the PORV block valve, from the control room, if the PORV fails to reclose.

A schematic of the PORV layout is shown in Figure A-1. There are two 50% flow capacity PORVs. Both PORVs receive a signal which causes them to open during a high Reactor Coolant System pressure transient. Once the Reactor Coolant System pressure decreases below the PORV setpoint, the PORVs reclose to preclude excessive loss of Reactor Coolant System inventory. However, if either or both PORVs do not reclose the operator has the capability of terminating flow through the valve(s) by closing the block valve(s).

4.0 ANALYSIS ASSUMPTIONS

The following assumptions were made in performing the reliability analysis:

1. PORV loss of coolant incident is defined as the inability to terminate flow through both PORVs to preclude excessive loss of Reactor Coolant System inventory.

2. At the actuation of the PORVs, the operator's normal stress level changes to a level intermediate between normal and severe stress (average of normal and severe stress levels).

3. Both PORVs have identical setpoint.

4. Failed components are not repaired during this SFE.

5. High pressurizer pressure condition exists at the actuation of the PORVs.

6. The reactor is at power prior to actuation of the actuation of the PORVs.

7. The component availability data for PORV loss of coolant incident which was used is given in Table A-1.

5.0 RESULTS

The fault tree logic diagram for power operated relief valve (PORV) loss of coolant incident is shown in Figure A-2. The minimal cutsets consist of at least three components. Therefore, all three component events must occur in order for a PORV loss of coolant incident to occur.

Best estimate recurrence frequencies for the PORV loss of coolant incident were calculated for the following cases:

1. Turbine runback feature and no operator action
2. Without turbine runback feature and no operator action
3. Without turbine runback feature and with operator action
4. Without turbine runback feature and with automatic closure of block valve
5. Without turbine runback feature and with automatic closure of series redundant block valves

The results are shown in Table A-2. Cases 4 and 5 assumed potential improvements to the current plant design.

6.0 LIST OF REFERENCES

Fault Tree Title: PORV LOSS OF COOLANT INCIDENT

Ref. No.	Description
1.	User's Manual and Output Guide for C-E Reliability Evaluation Code (CEREC), Rev. 1, W.S. Chow.
2.	Combustion Engineering Interim Data Base - Failure Rates for Nuclear Power Plant Components, D.J. Finnicum.
3.	IEEE STD500-1977, IEEE Guide to the Collection and Presentation of Electrical, Electronic, and Sensing Component Reliability for Nuclear Power Generating Stations.
4.	WASH 1400 (NUREG-75/014) Reactor Safety Study, An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants, Appendices III and IV, (Tables III-2-1 and III-6-1).
5.	Combustion Engineering Reliability Data System, Initiating Event Report (1-1-61 to 12-31-77), R.G. Sider.
6.	NPRDS 1977 Annual Reports of Cumulative ystem and Component Reliability, September, 1978.
7.	St. Lucie II SAR, Section(s) 5.5.12
8.	Post-TMI Evaluation Task 3 Follow-up Report, Pressurizer Systems and Emergency Power Supplies, Combustion Engineering, November, 1980.
9.	NUREG-0560, Staff Report on the Generic Assessment of Feedwater Transients in PWRs Designed by Babcock & Wilcox Company, U.S. NRC, May, 1979.
Drawings	St. Lucie II, Sequence of Events Auxiliary Diagrams St. Lucie II, Reactor Coolant System P&I Diagram, E-13172-310-109, Rev. 03

TABLE A-[illegible]

COMPONENT AVAILABILITY DATA

FOR

PORV LOSS OF COOLANT INCIDENT

Component Identification	Description	Code	Frequency (1/yr.)	Ref.	Demand Failure Rate	Ref.
Power Operated Relief Valve	Opens on Demand (With Turbine Runback)	PORV100D	6.60E-01	8		
		PORV200D	6.60E-01	8		
	Opens on Demand (Without Turbine Runback)	PORV100D	2.78E-01	8		
		PORV200D	2.78E-01	8		
	Opens Spuriously	PORV10S	2.80E-03	*		
		PORV20S	2.80E-03	*		
	Fails to Reclose	PORV1FTR			2.00E-02	9
		PORV2FTR			2.00E-02	9
Block Valve IA	Mech. Malf.	BVIAMM			6.59E-05	2
	Valve Motor Fails	BVIAMT			2.02E-04	2
	Valve Breaker Fails to close	BVIABR			1.00E-06	3
	Automatic Signal not Received	BVIAAS			1.20E-02	4
	Operator Fails to Close Valve	BV14030P			1.55E-01	**
Block Valve IIA	Mech. Malf.	BVIIAMM			6.59E-05	2
	Valve Motor Fails	BVIIAMT			2.02E-04	2
	Valve Breaker Fails to close	BVIIABR			1.00E-06	3
	Automatic Signal not Received	BVIIAAS			1.20E-02	4
	Operator Fails to Close Valve	BV14050P			1.55E-01	**

TABLE A-1 (continued)

COMPONENT AVAILABILITY DATA

FOR

PORV LOSS OF COOLANT INCIDENT

Component Identification	Description	Code	Frequency (1/yr.)	Ref.	Demand Failure Rate	Ref.
Block Valve IB	Mech. Malf.	BVIBMM			6.59E-05	2
	Valve Motor Fails	BVIBMT			2.02E-04	2
	Valve Breaker Fails to Close	BVIBBR			1.00E-06	3
	Automatic Signal not Received	BVIBAS			1.20E-02	4
Block Valve IIB	Mech. Malf.	BVIIBMM			6.59E-05	2
	Valve Motor Fails	BVIIBMT			2.02E-04	2
	Valve Breaker Fails to Close	BVIIBBR			1.00E-06	3
	Automatic Signal not Received	BVIIBAS			1.20E-02	4

* Best Estimate Using 246.2 Possible Reactor Years

** Values Were Obtained from Data in Ref. 4

Table A-2

Recurrence Frequencies for PORV Loss of Coolant Incident

CASE NO.	DESCRIPTION	FREQUENCY (1/YR.)
1	Turbine runback feature and no operator action	2.6E-02
2	Without turbine runback feature and no operator action	1.1E-02
3	Without turbine runback feature and with operator action	1.8E-03
4	Without turbine runback feature and with automatic closure of block valve	1.4E-04
5	Without turbine runback feature and with automatic closure of series redundant block valves	1.7E-06

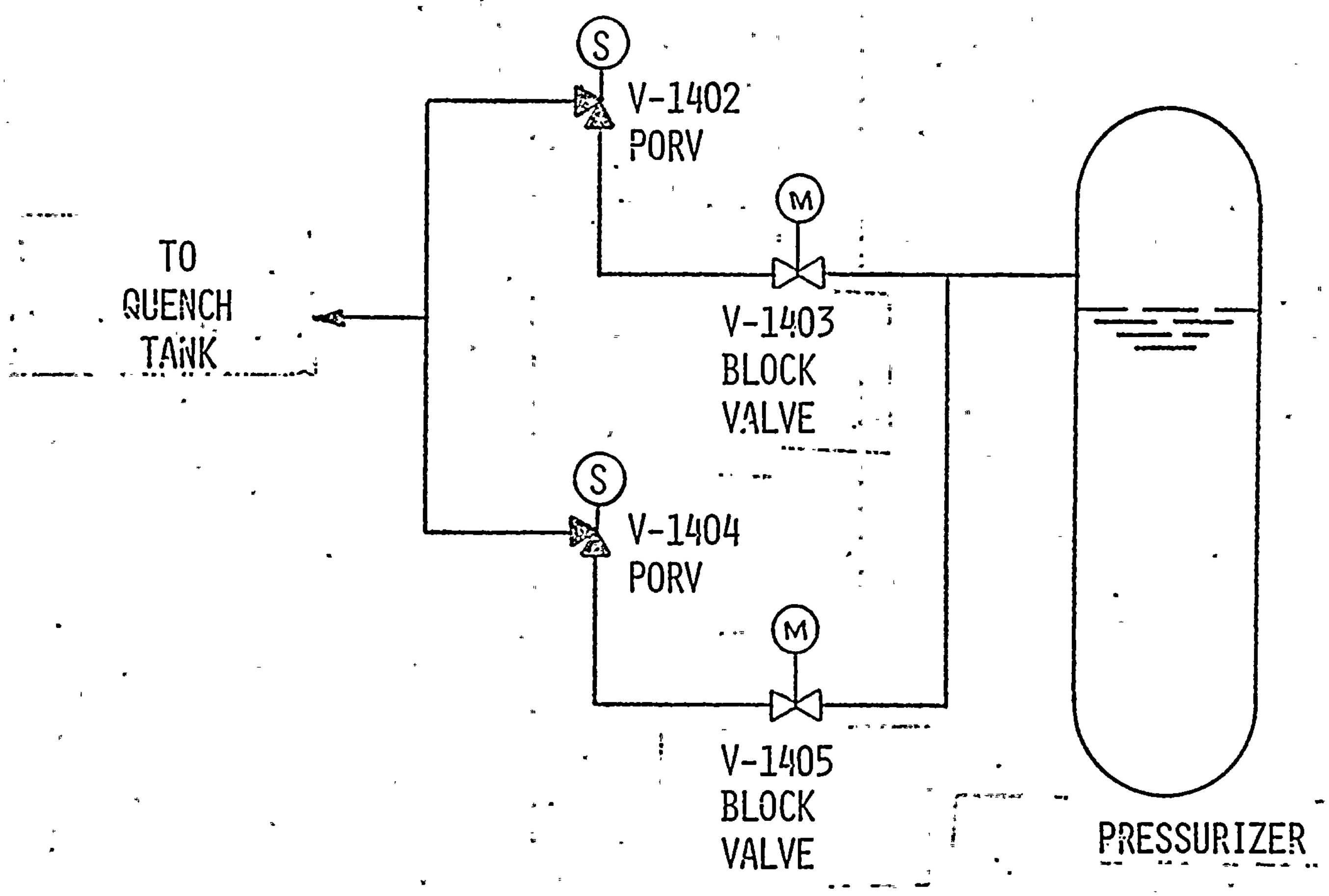


FIGURE A-1
POWER OPERATED RELIEF VALVE
SCHEMATIC

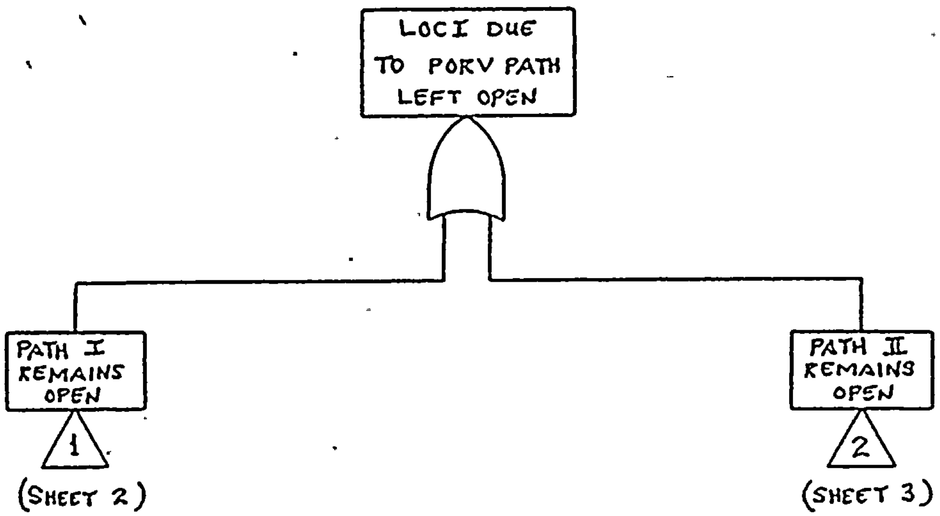


Figure A-2

Fault Tree Logic Diagram for

Power Operated Relief Valve Loss of

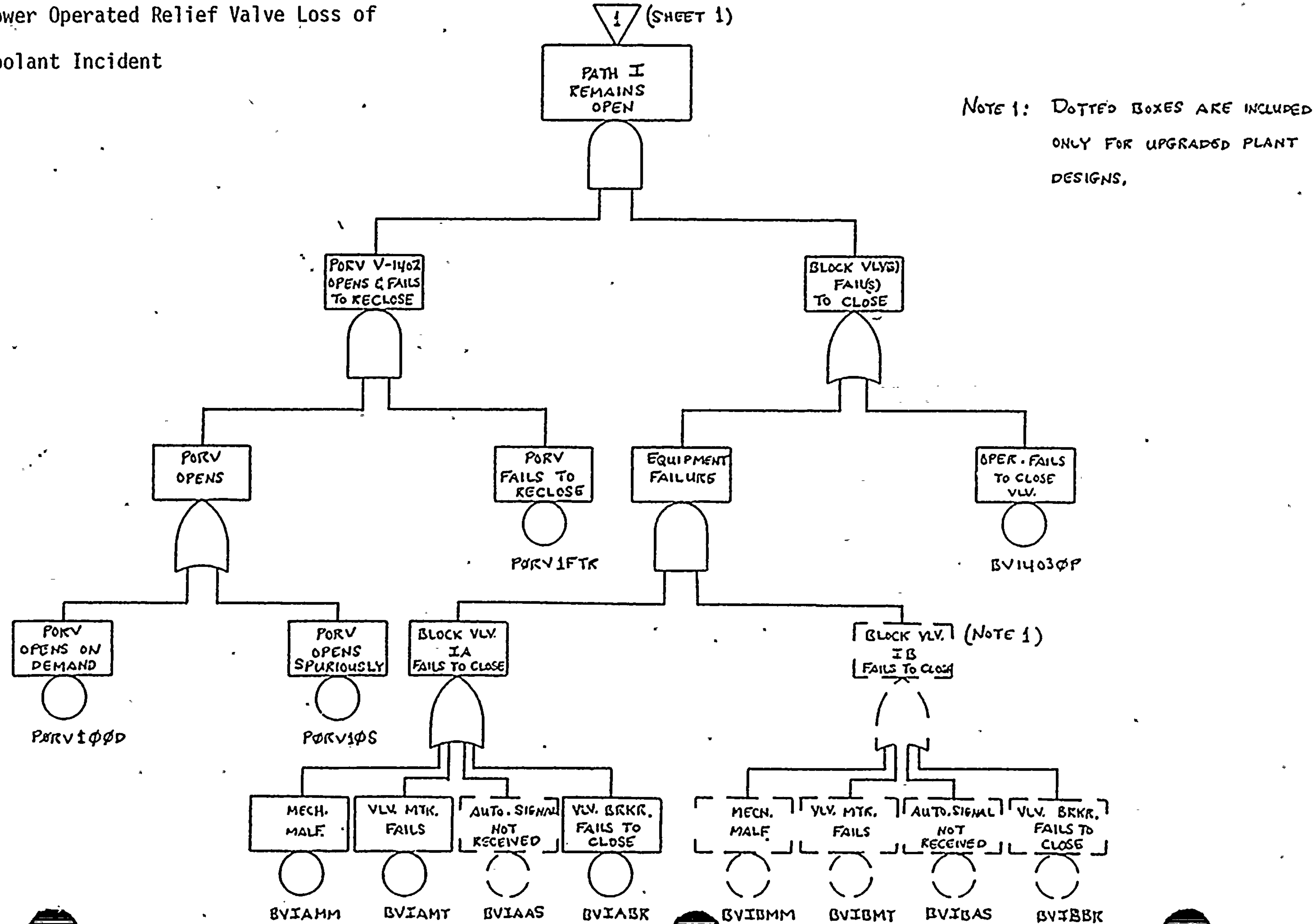
Coolant Incident

SHEET 1 OF 3

Fault Tree Logic Diagram for

Power Operated Relief Valve Loss of

Coolant Incident



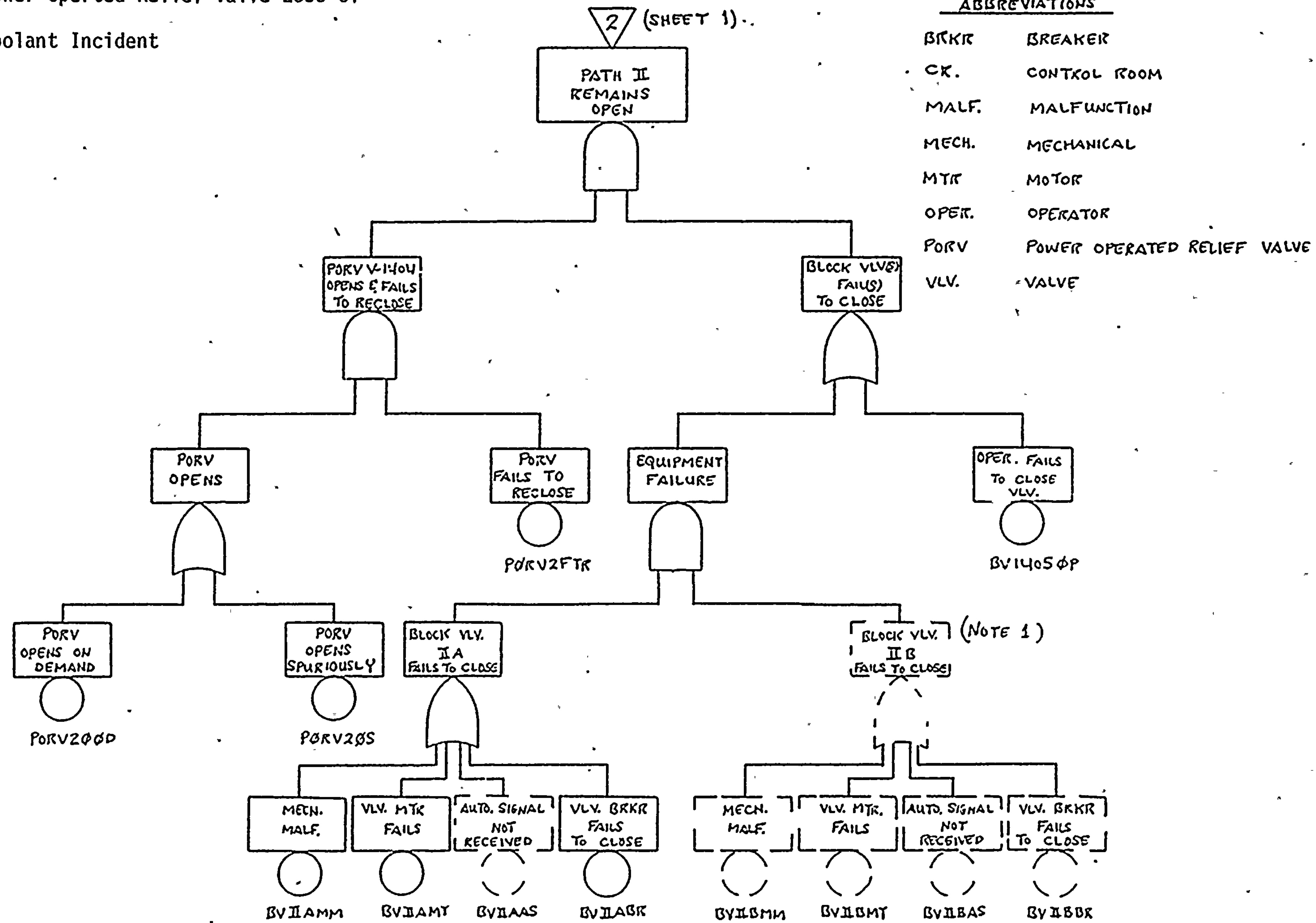
SHE OF 3

Figure A-2

Fault Tree Logic Diagram for

Power Operted Relief Valve Loss of

Coolant Incident



SHEET 3 OF 3

Figure A-2

ENCLOSURE 4

RESPONSE TO NUREG 0660
ITEM II.K.3.17
REPORT ON OUTAGES OF ECC SYSTEMS

This report details the outages of ECC Systems since licensing of St. Lucie Unit 1 on March 1, 1976. The report was generated on the following bases:

A. Only those outages which resulted in less than the minimum required ECCS capacity for the plant mode are included.

B. Failure of a component or removal from service of a component which does not place the plant into Technical Specification action statement (e.g.;removal of third pump from service in the component cooling water system) were not considered because the plant still has the minimum systems.

The report lists outages other than scheduled preventative maintenence chronologically by system. Scheduled preventative maintenance is indicated by notes in the listing for each system.

It should be noted that with the 18 month fuel cycle currently in effect the "annual" preventative maintenance and overhauls performed during refueling will be extended to a eighteen month cycle. Additionally, St. Lucie mechanical and control systems have been designed such that most periodic testing can be done without taking the systems out of service. With one exception, those tests which require taking a system out of service are performed in operating modes which do not require the system to be operable.

Cumulative reactor availability for St. Lucie Unit 1 was 26,356 hours as of October 31, 1980. No ECC system train had an outage rate of over 1% for this period using the bases specified above. The approximate percentages are noted with the data for each system.

HIGH PRESSURE SAFETY INJECTION SYSTEM

OUTAGE DATES	LENGTH (HOURS	CAUSE	COMPONENT(S)	ACTION TAKEN
4/26/76	16	Leaking Seal	A HPSI Pump	Repair
2/8/77	1	Dirty CCW Flowmeter	A HPSI Pump	Clean CCW Flowmeters
5/20/77	1	Leaking CCW Union	A HPSI Pump	Remake Coupling

NOTES: (a) B and C HPSI Pumps are redundant when properly aligned mechanically and electrically. For preventative maintenance they are not removed from service simultaneously. A HPSI pump when removed from service for preventative maintenance places one HPSI header out of service.

(b) Annual Preventative Maintenance
Mechanical - scheduled in December Time: 6 hours
Electrical - scheduled in September Time: 8 hours

(c) Semiannual Preventative Maintenance
Scheduled in June and December Time: 3 hours

(d) Quarterly Preventative Maintenance
Scheduled in March, June, Sept & December Time: 3 1/2 hours

(e) Total annual A header preventative maintenance outage time excluding periods when system is not required is 34 hours per year.

(f) The cumulative out of service percentage is .67.

LOW PRESSURE SAFETY INJECTION SYSTEM

OUTAGE DATES	LENGTH (HOURS)	CAUSE	COMPONENT(S)	ACTION TAKEN
2/5/77	3	Dirty CCW Flowmeters	1A LPSI Pump	Clean CCW Flowmeter
2/9/77	2	Dirty CCW Flowmeters	1B LPSI Pump	Clean CCW Flowmeter
5/10-11/77	30	Cracked Weld	1A2 SIT	Cooldown Parts (LER 77-29)
9/25/77	1	Check Timers	1A LPSI Pump	Adjust ECCS Timers
9/25/77	1	Check Timers	1B LPSI Pump	Adjust ECCS Timers
11/21-22/78	38	Valve Operator Failed	MV-07-1B	Repairs (LER 78-44)
2/22/79	1/2	Leaking Fill and Drain Valve	1B1 SIT	Refill and Repressurize, Maintenance (LER 79-007)
2/23-24/79	34	Allignment Check	1B LPSI Pump	Check Pump Alignment
2/22/79	12	Annual PM/Maint.	1B LPSI Pump	
9/17/79	2	Level Trans Drift	1B1 SIT	Repair (LER 79-29)
2/26-27/80	16	Annual PM/Maint	1B LPSI Pump	
12/3-5/80	46	Seal Leakoff High	1A LPSI Pump	Repair

NOTES:

(a) Annual preventative maintenance scheduled for February — Time: 8 hours

(b) Semi-annual preventative maintenance scheduled for March and September — Time: 5 hours

(c) The 2/22/79 and 2/26-27/80 preventative maintenance dates are included in the above list because the outage provided a convenient time to perform some additional discretionary maintenance.

(d) Total annual preventative maintenance time, including maintenance during periods LPSI is not required, is 18 hours per year per train.

(e) The cumulative out of service time for train A was 0.4% and train B ia 0.7%. Total plant operation with less than required LPSI trains is 1.1%.

CONTAINMENT SPRAY

OUTAGE DATES	LENGTH (HOURS)	CAUSE	COMPONENT(S)	ACTION TAKEN
2/11/77	2	Dirty CCW Flowmeter	1A CS Pump	Clean Flowmeter
2/11/77	2	Dirty CCW Flowmeter	1B CS Pump	Clean Flowmeter
8/17/77	1	Valve Not Fully Open	1A Header Valve	Open Valve (LER 77-34)
9/8/77	6	PC/M 246-77	1B CS Pump	Complete PC/M
9/8/77	6	PC/M 246-77	1A CS Pump	Complete PC/M
9/25/77	1	Adjust Safeguard Timer	1A CS Pump	Adjust
9/25/77	1	Adjust Safeguard Timer	1B CS Pump	Adjust

NOTE: (a) Annual preventative maintenance is scheduled in March. Time: 7 hours

(b) To avoid inavertant spray down of containment and introduction of sodium hydroxide to primary systems, the respective train is taken out of service for the monthly pump run. Time: 1/2 hour per train per month

(c) Cumulative time out of service for A header is 0.17% and for B header is 0.16%.

CONTAINMENT COOLERS

NOTE: To date maintenance and preventative maintenance has been performed in modes 4, 5 or 6 when the coolers are not required. The plant intends to continue this policy. Three coolers operate at all times in operation. This is indicative of the reliability of these components.

COMPONENT COOLING WATER SYSTEM

OUTAGE DATES	LENGTH (HOURS	CAUSE	COMPONENT(S)	ACTION TAKEN
8/30/77	.98	Leaking Mechanical Seal/ Operator Error	1B CCW Pump	Repair (LER 77-31)
10/2/80	.75	Battery Charger/ Elect. Configuration	1C and 1B CCW Pump	Repair/Realign (LER 80-61)
12/1-3/80	24	Leaking Line	1B CCW Heat Exchanger	Repair (LER 80-67)

NOTE: Annual, semi-annual and quarterly preventative maintenance is performed with either the plant in a mode not requiring both CCW loops or the extra "C" CCW pump is employed to maintain independent CCW loops.

INTAKE COOLING WATER SYSTEM

OUTAGE DATES	LENGTH (HOURS	CAUSE	COMPONENT(S)	ACTION TAKEN
8/26/77	3	Plugged Seal Line/ Electrical Lineup	1B ICW Pump	Repair/Realign Electrical Plant (LER 77-35)

NOTE: Annual or semi-annual preventative maintenance is performed with either the plant in a mode not requiring both systems or the extra "C" pump is employed to maintain CCW requirements.

CHARGING AND VOLUME CONTROL SYSTEM (CVCS)

OOS

OUTAGE DATES	LENGTH (HOURS)	CAUSE	COMPONENT(S)	ACTION TAKEN
2/20/77	8	Seal Lube Pump Failure	A & B Charging Pumps	Replace (LER 77-11)
2/26/77	13	Seal Lube Pump Failure	A Pump (B still OOS)	Replace (LER 77-14)
10/2/78	1 1/2	Leaking Relief and Electrical Interlock	B Pump (A OOS for Maintenance)	Repair (LER 78-40)

NOTES: (a) Semi-annual and quarterly preventative maintenance is scheduled such that only one pump is out of service at a time. Only two of three pumps are required.

(b) Outages of individual charging pumps have occurred and resulted in generation of LERs. However, these events have not resulted in less than two charging pumps being in service at a time.

AUXILIARY FEEDWATER SYSTEM

OUTAGE DATES	LENGTH (HOURS)	CAUSE	COMPONENT(S)	ACTION TAKEN
4/8/76	48	Faulty Governor	1C AFW Pump	Replace (LER 76-11)
4/13/76	10	Improper Wiring	1C AFW Pump	Rewire (LER 76-13)
5/21/76	[illegible]	Corroded Contacts	1C AFW Pump	Replace/Seal (LER 76-22)
7/9/76	4	Moisture in Control Circuit	1C AFW Pump	Dry (LER 76-35)
7/17/76	4	Moisture in Control Circuit	1C AFW Pump	Dry & Seal (LER 78-35)
12/8/76	12	Corrosion in Latch Mech.	1C AFW Pump	Dry/Seal (LER 78-47)
2/17/77	40	Moisture in Terminal Box	1C AFW Pump	See Note (e) (LER 77-10)
8/11/77	1	Fail to Start - Cause Unknown	1C AFW Pump	Op-Check (LER 77-33)
2/9-10/78	28	Partially Shorted Winding	MV-09-11 (C AFW Pump Flow Control)	Check Motor (LER 78-6)
2/10/78	4	Partially Shorted Winding	MV-09-11	Replace (LER 76-6)
6/14/79	1	Steam Inlet Failed to Open	MV-08-3 (C AFW Pump Steam Inlet)	Op Check (LER 79-20)

NOTES:

(a) Annual preventative maintenance has been done during Mode 6 (refueling). Plant intends to continue this policy.

(b) Semi-annual preventative maintenance for A, B & C pumps scheduled in April and October. Time: 3 Hours

(c) Semi-annual proventative maintenance for C pump turbine scheduled in April and October. Time: 4 Hours

(d) Quarterly preventative maintenance for C pump turbine throttle scheduled in January, February, July and October. Time: 1 Hour

(e) Following the 2/17/77 incident on C AFW pump, steam line drains were rerouted, and electrical controllers were sealed. The absence or subsequent moisture and corrosion problems indicates that the change resolved this problem.

(f) Total annual preventative maintenance performed in modes requiring system is 30 hours per year.

(g) The cumulative time with less than the required AFW pumps is 0.6%.

DIESEL GENERAT

OUTAGE DATES	LENGTH (HOURS)	CAUSE	COMPONENTS	ACTION TAKEN
5/18/76	8	Clogged Filter (Air Start)	1A DG	Clean (LER 76-21)
6/2/76	1/2	Adjust Timer	1A DG Breaker	Return to Service
6/2/76	1/2	Adjust Timer	1B DG Breaker	Return to Service
5/10/76	3	DG will not stop from CR	1B DG	Repair Switch
1/11/77	1	Breaker Test Light Out	1B DG Breaker	Replace
1/18/77	69	Turbocharger Failure	1B DG	Replace (LER 77-2)
1/19/77	1 1/2	Stuck Linkage	1A DG	Lubricate (LER 77-3)
1/20/77	8	Check Phase Balance	1B DG	Check, Return to Service
3/1/77	8	Operator Error	1A DG	Reset Trip (LER 77-15)
9/20/77	65	Turbocharger Failure	1A DG	Repair (LER 77-42)
2/27-28/78	2	Clean Switchgear	1A DG Switchgear	Return to Service
3/10/78	1	Install PC/M	1A DG	
3/13/78	2	Install PC/M	1B DG	
9/5/78	3 1/2	Breaker Failure to Close	1A DG Breaker	Repair (LER 78-36
10/16/79	2	Failure of Voltage Reg.	1A DG	Repair (LER 79-32)
9/3/80	6	Leaking Relief Valve	1A DG	Replace (LER 80-55)
10/1/80	12.	Leaking Relief Valve	1A DG	Vent Sys (LER 80-56)

NOTES:

(a) Annual preventative maintenance has been performed during Mode 6 (refueling). Intent is to continue this practice.

(b) Semi-annual preventative maintenance scheduled for January and July — Time: 3 Hours

(c) Monthly preventative maintenance scheduled monthly — Time: 1 Hour

(d) Total time required for semi-annual and monthly preventative maintenance is 18 hours.

(e) The cumulative total out of service time for A DG is 0.7% and B DG is 0.6%. Total plant time is 1.3%.

EMERGENCY SAFEGUARD FEATURES ACTUATION SYSTEM

A number of instanceshave occurred where single or multiple ESFAS channels have drifted out of specification or been found to be incorrectly set. These instances are listed below:

4/21/76	Two containment pressure and two steam generator pressure bistables drifted out of specification. Bistables were conservatively reset. (LER 76-17)
5/5/77	One steam generator pressure channel out of specification. Incorrect setting was due to not setting following maintenance. (LER 77-25).
8/1/77	Two refueling water storage tank (RWT) Levels out of tolerance due to setpoint drift. Instruments reset. (LER 77-32).
10/28/77	Channel A RWT level setpoint drifted. Bistable trip unit replaced. (LER 77-48).
1/3/78	Channel A RWT level setpoint drifted. Cause unknown. Channel reset. (LER 78-1).
2/19/80	Channel D RWT level setpoint drifted low. Channel was reset. (LER 80-11).

ENCLOSURE 5

FLORIDA POWER & LIGHT COMPANY
ST LUCIE PLANT UNIT 1
CONTROL ROOM HABITABILITY REPORT

The St Lucie Unit 1 control room has been specifically designed to assure that the control room operators will be adequately protected against the effects of accidental release of toxic and radioactive gases and that the plant can be safely operated or shut down under design basis accident conditions.

The St Lucie Unit 1 control room habitability systems are designed to:

a) Limit control room personnel doses to within GDC 19 and SRP 6.4 guidelines,

b) maintain CO_2 levels below one percent and O_2 levels at a minimum of 17 percent at all times,

c) maintain the ambient temperature required for personnel comfort and equipment operation at all times,

d) withstand design basis earthquake loads without loss of function, and

e) permit personnel occupancy during a chlorine, or other toxic chemical, release accident.

The location of the control room within the Reactor Auxiliary Building is shown in Figure 1. The St. Lucie Unit 1 control room ventilation and air conditioning system has undergone extensive modifications, the results of which are shown on the diagram provided as Figure 2.

The St Lucie Unit 1 control room habitability systems have been designed assuming that ten persons would be present in the control room during the accident. By assuming ten occupants, not only are the operators on the shift at the time of the accident accounted for, but also additional personnel such as health physicists.

A potable water supply of 1.0 gallon per man per day is provided in a number of plastic containers stored in the control room. The total amount of potable water stored exceeds 100 gallons. This water requirement conservatively allows for evaporated moisture and moisture losses in urine and feces. By Reference 1, 3.0 quarts per day per man is required at 75 F drybulb, while one gallon per day per man is the total recommended for drinking, food preparation, personal hygiene, and medical requirements.

To provide means for disposing of garbage, trash and human waste, the equivalent of sanitation kit III as recommended by the Office of Civil Defense for fallout shelters is provided and includes the following:

Paper, toiler tissue	5 rolls
Plastic commode seat	1
Heavy sanitary pads	1 dozen
Regular sanitary pads	1 dozen
Polyethylene gloves	1 pair
Tie wires, bag closures	1
Cups and lids	35
Can opener (manual)	1
Commode chemical	1

Polyethylene bag liners	1
Instruction sheet	1
Fiberboard boxes	2
Fiber drum	1

The above is recommended for a two week stay by 25 occupants in a survival shelter and is therefore conservative for this application. A supply of food is stored in the control room which is sufficient to maintain habitability for ten men for a week.

The control room contains portable fire extinguishing equipment to permit the timely extinguishing of control room fires. Storage provisions for bottled air or chemox canisters are provided in the control room for six hours of occupancy.

Toxic Chemical Release

A review of the locations and distances of industrial, military, and transportation facilities and routes in the vicinity of the St Lucie site is included in Sections 2.2.1 and 2.2.2 of the St Lucie Unit 2 FSAR (see Attachment I). This review has been performed using the guidance from Standard Review Plan Sections 2.2.1-2.2.2 Rev. 1 and since both Units 1 and 2 occupy the same site, is applicable for the identification of potential hazards to the St Lucie Unit 1 control room personnel.

Because of the close proximity of the St Lucie Unit 1 control room outside air intake to the St Lucie Unit 2 air intake, the evaluation of offsite toxic chemical sources contained in St Lucie Unit 2 FSAR Section 2.2.3 is also applicable to the St Lucie Unit 1 control room (see Attachment I). Since the issuance of the St Lucie Unit 2 FSAR it has been learned that chlorine, used for the city water system and the sewage treatment facility, is stored in tanks located near the northwest corner of the St Lucie site. Because of the close proximity to the site an analysis of effects of a chlorine release accident on the St Lucie Unit 1 control room habitability has been performed. The results are given in Table 1.

The short separation distances between the air intakes and the locations at which toxic chemicals are stored onsite require that a separate analysis be performed for the St Lucie Unit 1 control room for the chemicals: ammonium hydroxide, carbon dioxide and cyclohexylamine. The arguments used in the St Lucie Unit 2 FSAR to eliminate the other chemicals stored onsite remain valid for the St Lucie Unit 1 control room as well (see Attachment I). In addition to the above listed chemicals which will be common to both St Lucie Units 1 and 2, chlorine is and will be used for the circulating treatment at St Lucie Unit 1 until the installation of a hypochlorite generator.

An analysis of the effects of an accidental release of the three above mentioned toxic chemicals and of onsite and offsite stored chlorine has been performed. The locations of these chemicals and of the St Lucie Unit 1 outside air intakes are shown in Figure 1. The results of that analysis are provided in Table 1 and are summarized below.

Ammonium hydroxide is stored onsite in two 55 gallon drums at 30 percent concentration by weight. The concentration at the outside air intakes of the control room is calculated assuming all the ammonia in the solution becomes airborne instantaneously following a postulated rupture of the container. This very conservative assumption results in concentrations at the outside air intakes well is excess of those which can be actually expected.

In the case of carbon dioxide, complete vaporization is assumed immediately following accidental releases. The airborne transport of the puff is modeled using the instantaneous release diffusion model presented in Regulatory Guide 1.78. Since the control room is located a short distance from the release point and the amount of chemical is small, the model is adjusted to allow for additional dispersion in the vertical direction by assuming uniform mixing between the ground and the elevation of the fresh air inlet (a 19 meter elevation from ground level is used).

The concentration of cyclohexylamine in the feed tank in the Turbine Building is 10 percent. The chemical is delivered to the site in 55 gallon drums in the concentrated form and stored in a 20 ft by 12 ft storage room. Conservatively it is assumed that a 55 gallon drum fails. The evaporation rate of cyclohexylamine is calculated using Equation 2.1-18 in Reference 2. The transport of vapor is modeled by the short term, continuous release diffusion equation presented in Regulatory Guide 1.4.

It is conservatively assumed that the centerline of the plume remains incident on the control room outside air intakes during the entire time it takes the liquid to evaporate. A ground level release is assumed. However, credit is taken for additional dispersion in the vertical direction by assuming uniform mixing between the ground and the elevation of the outside air intake.

The offsite chlorine is stored in two 150 lb cylinders near the sewage treatment facility and one 150 lb cylinder near the city water storage tanks. The airborne transport of a puff release of 25 percent of the closest tank contents is modeled using the instantaneous release diffusion model presented in Regulatory Guide 1.78. As with the carbon dioxide analysis, the model is adjusted to allow for additional dispersion in the vertical direction.

Chlorine used in the treatment of the circulating water is stored onsite in one-ton cylinders. In order to provide control room occupants protection against an accidental chlorine release, seismic Category I chlorine detectors have been installed at the control room outside air intakes. An analysis has been performed to evaluate the concentration in the control room based on the failure of a one-ton chlorine tank and the instantaneous release of 25 percent of its contents. The guidance given in Regulatory Guide 1.78 has been followed in modeling the diffusion of the chlorine cloud. The results of the analysis indicate that a maximum chlorine concentration of 25 ppm occurs in the control room 30 minutes following the detection of the gas. The toxicity limit is reached in 9 minutes. These time intervals are

greater than the Regulatory Guide 1.78 requirement that the minimum time from detection to 15 ppm be at least two minutes, consequently, the control room occupants will have sufficient time to don their self-contained breathing apparatus. Table 2 lists the assumptions and parameters used in this analysis.

The models described above are used to calculate concentrations of toxic chemicals at the control room outside air intakes. As indicated in Table 1, the concentration of ammonia and chlorine at the outside air intake of the control room exceed the toxicity limits. Since carbon dioxide and cyclohexylamine concentrations at the outside air intake are below the toxicity limits, the concentrations inside the control room are not required to be calculated.

The concentrations inside the control room are calculated based on the following equation:

$$C(t) = e^{-vt} \int_{0}^{t} v e^{vt'} X(t')\, dt'$$

where:

$C(t)$ = chemical concentration inside the control room at time t

$X(t')$ = chemical concentration outside the air intake at time t'

v = control room air exchange rate of 0.52 per hour which is based on normal air intake rate of 920 cfm

For the chemicals analyzed, except chlorine, it is found that the concentration remains well below the toxicity limit under the assumptions that the control room is not isolated and no action is taken by the operators following the accident. In the case of chlorine stored onsite, there is sufficient time for the control room occupants to don breathing apparatus. Therefore, an accidental release of toxic chemicals stored on and off site poses no threat to the control room operators.

Radiological Release

In the event of a postulated accident (LOCA) there could be three major sources of radiation exposure to the control room personnel: 1) direct radiation exposure from radioactive material outside of the control room, 2) submersion exposure from radioactive material within the control room, and 3) inhalation exposure from radioactive material within the control room. An analysis has been performed to insure that the St Lucie Unit 1 control room personnel do not receive a combined dose from any accidental release of radioactivity which exceeds the limit of GDC 19 of 10CFR50 and the dose guidelines of SRP 6.4.

The source of the largest potential direct dose to the control room personnel is the control room emergency filtration system which is located in a room adjacent to the control room, separated from it by a one ft thick concrete wall. Using conservative assumptions, this system will contribute less than 1 rem over a 30 day period due to buildup of radioactive material on its filters.

The next largest potential direct dose is from the external atmosphere surrounding the control room. This source arises from assuming a 0.5 percent per day leakage rate of the containment atmosphere to the external atmosphere for the first day following a LOCA, and a 0.25 percent per day leakage rate for subsequent days. The control room is shielded from this source by at least two ft and as much as four ft of concrete. The time integrated dose calculated using an infinite cloud model, which overestimates the answer, amounts to less than 0.2 rem.

The third largest potential direct dose arises from the radioactive material released to the containment atmosphere following a LOCA. This source combines the dose from the material remaining in the atmosphere as well as that which plates out inside the containment. The magnitude of the initial source was determined assuming that 100 percent of the noble gas inventory and 50 percent of the halogen inventory of the core is released immediately following a LOCA. Further, a halogen plateout factor of 0.50 is used. The activity at subsequent times was determined considering radioactive decay of the isotopes. The control room is shielded from these sources by at least the three ft thick concrete shield building wall, the two inch thick steel containment wall, and the two ft thick control room wall. The source was modeled as a cylindrical volume source, and no credit was taken for any additional shielding from structures in the interior of either the Shield Building or the Reactor Auxiliary Building. The total dose then calculated amounts to less than 0.1 rem.

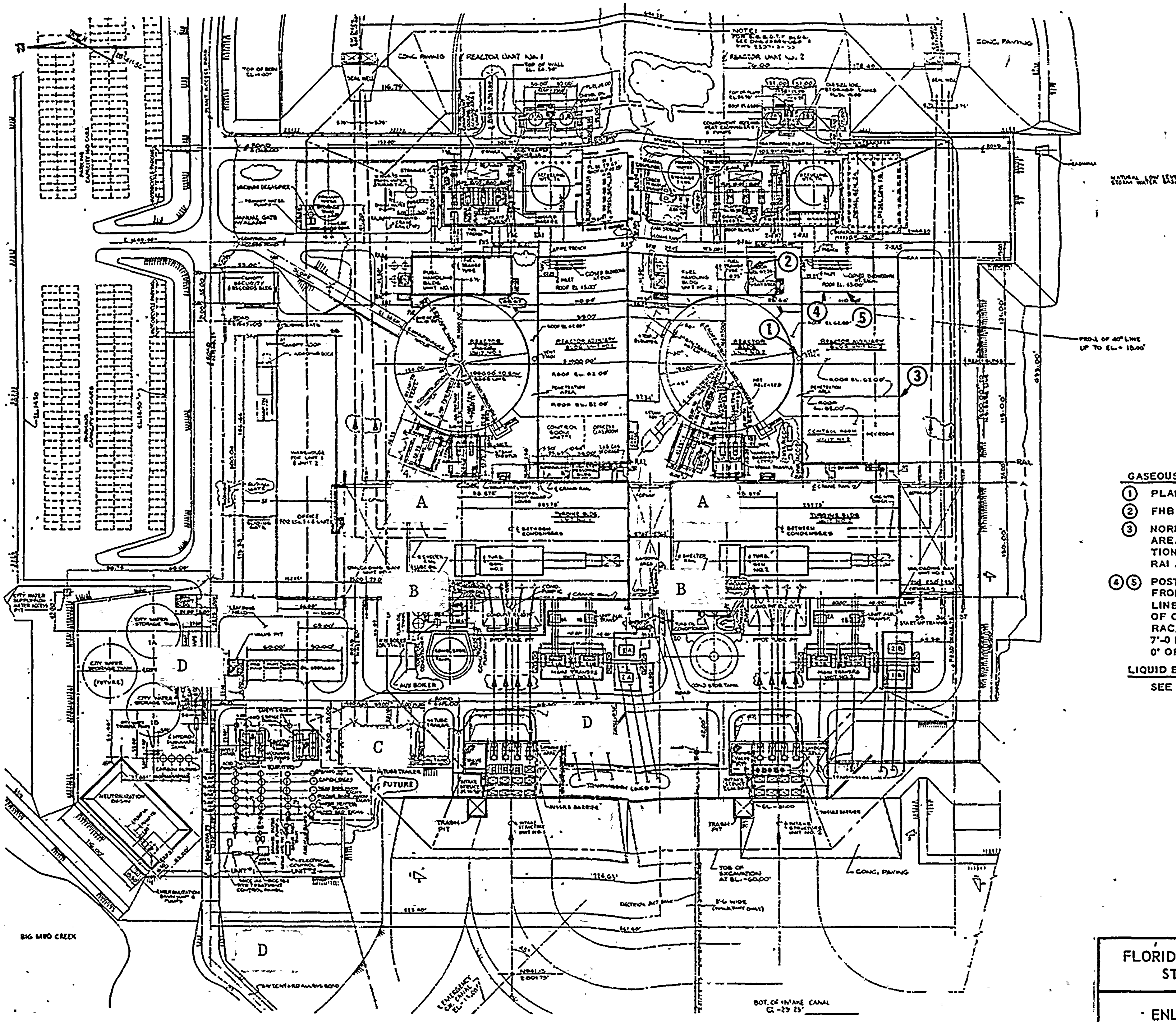
The remaining source of direct radiation, the Shield Building ventilation emergency filtration units, the atmosphere outside the control room but inside the Reactor Auxiliary Building, and the containment sump water, contribute a total dose to the control room of less than 0.005 rem for 30 days post-LOCA. The Shield Building ventilation emergency filtration units are separated from the control room by at least four to six ft of concrete shielding. The atmosphere inside the Reactor Auxiliary Building becomes radioactive from the relatively slow leakage of contamination from the containment. This low leak rate combined with concurrent radioactive decay, results in a low dose to the control room. The final source, the radioactive water in the containment sump, is separated from the control room by at least nine ft of concrete (measured perpendicularly through the shielding). As a consequence, its dose contribution is negligible under the most conservative shielding assumptions.

The analysis of the dose which the control room personnel would recieve following a postulated accident (LOCA) via the air introduced into the control room has been performed using the methodology of Murphy and Campe. The St Lucie Unit 1 control room is designed to be maintained at a slightly positive pressure after a LOCA. The inhalation and submersion doses to its occupants are proportional to the makeup air intake rate, necessary to maintain the control room envelope pressurized at a differential pressure of at least 1/8 inch water gauge. The assumptions and parameters used to determine these doses are provided in Table 3, while the doses to control room personnel are reported in Table 4. The analysis shows that the control room design meets the requirements of GDC 19 and SRP 6.4 without the use of self-contained breathing apparatus, bottled air, or potassium iodide.

REFERENCES

1) Department of Defense, Office of Civil Defense, Shelter Design and Analysis, Volume 3, Chapter 9.

2. Toxic Vapor Concentrations in the Control Room Following a Postulated Accidental Release, J Wing, U S Nuclear Regulatory Commission, NUREG-0570, June 1979.



A. Cyclohexylamine
B. Ammonium Hydroxide
C. Acetylene & CO_2
D. Chlorine

GASEOUS EFFLUENT RELEASE POINTS

① PLANT VENT STACK – HEIGHT +202'-8

② FHB VENT STACK – HEIGHT +109'-6

③ NORMAL HVAC RELEASES FROM HOT AREAS IN RAB – CENTERLINE ELEVATION +38'-0, 3'-6 WEST OF COLUMN RAI AND 0' OFF COLUMN 2RA5.

④⑤ POSTACCIDENT HVAC RELEASE FROM ECCS AREA IN RAB – CENTERLINE ELEVATION +57'-2, 5'-10 NORTH OF COLUMN 2RA3 AND 0' OFF COLUMN RAC. CENTERLINE ELEVATION +57'-2, 7'-0 NORTH OF COLUMN 2RA2 AND 0' OFF COLUMN RAC.

LIQUID EFFLUENT RELEASE POINT

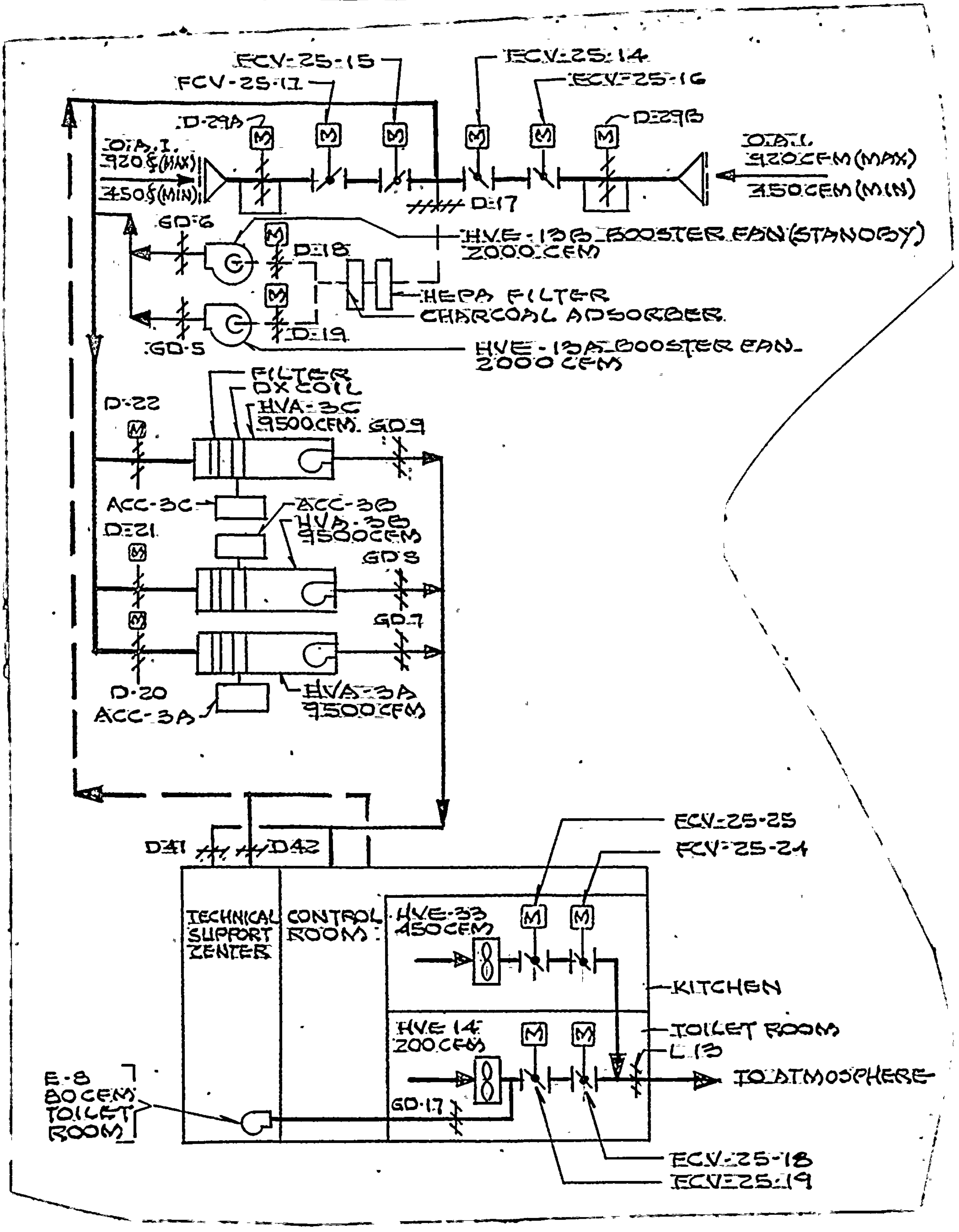
SEE FIGURE 1.2-1

REF DWG: 2998-G-059 (REV. 3)

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT

ENLARGED SITE PLOT PLAN

FIGURE 1



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

CONTROL ROOM
AIR FLOW DIAGRAM

FIGURE 2

TABLE 1
Toxic Chemical Evaluation

Toxic Chemical	Toxicity Limit (ppm)	Distance From Control Room* (feet)	Quantity Released	Peak Conc. (ppm) At OAI	Peak Conc. (ppm) Inside Control Rm	Time to Reach Toxicity Limit (minutes)	Note
Ammonium Hydroxide	500	210	55 gal;30% by weight	7.69(+4)	1.32(+2)		1
Carbon Dioxide	10000	450	360 SCF	4.40(+3)	-		1,2
Cyclohexylamine	20	100	55 gal;100% by weight	3.45	-		2,3
Chlorine (offsite)	15	580	38 lb	1.58(+3)	4.86		
Chlorine (onsite)	15	328	500 lb	-	25	9	

* See Figure 1 for locations

Notes: 1) The concentration was determined assuming that the toxic chemical becomes instantaneously airborne following the rupture of the container.

2) Not a design basis event, because the calculated concentration at the outside air intake of the control room is less than the toxicity limit.

3) The concentration was determined assuming that the toxic chemical evaporates following the rupture of the container.

TABLE 2

Regulatory Guide 1.78 Onsite Chlorine Release Evaluation
Calculation Assumptions

Chlorine container size	1 ton
Distance to outside air intake ducts	100 meters
Control room filter efficiency for chlorine	99%
Meteorological conditions	Pasquill F & 1m/sec
Control room chlorine detectors sensitivity	5 ppm
Control room volume	106, 920 ft^3
Normal Control room air intake rate	920 cfm
Time for chlorine activation detection signal*	6 seconds

* Measured from time when chlorine concentration outside control room intakes is 5 ppm, to the time of isolation.

TABLE 3

Assumption and Parameters

Control Room Volume	
Control Room	62,550 ft^3
Technical Support Center	44,370 ft^3
TOTAL	106,920 ft^3
Outside Air Intake Rate	
Normal	920 cfm
Emergency	450 cfm
Emergency Recirculation Rate Through Charcoal Adsorbers	1550 cfm
Emergency Filtration System Charcoal Adsorber Efficiency;	
Elemental Iodine	95%
Oganic Iodine	95%
Particulate Iodine	99%
Noble Gas	0%
Unfiltered Infiltration Rate	3 cfm
Atmospheric Diffusion Factor	
0-8 hours	4.86(-4) sec/m^3
8-24 hours	4.17(-4) sec/m^3
1-4 days	2.80(-4) sec/m^3
4-30 days	1.59(-4) sec/m^3
Occupancy Factor	
0-1 day	100%
1-4 days	60%
4-30 days	40%

Finite Cloud Approximation Used to Estimate Whole Body Gamma Dose

TABLE 4

Radiological Doses to Control Room Personnel

	Dose (Rem)	GDC 19/SRP 6.4 Limit (Rem)
Whole Body		
Direct	1.3	
Submersion	0.6	
TOTAL	1.9	5.0
Skin		
Submersion	18.0	30.0
Thyroid		
Inhalation	18.0	30.0

ATTACHMENT 1

2.2 NEARBY INDUSTRIAL, TRANSPORTATION, AND MILITARY FACILITIES

2.2.1 LOCATION AND ROUTES

The St Lucie site is located on Hutchinson Island approximately six miles southeast of Ft. Pierce, Florida. Within five miles of the St Lucie site are: (a) six primary, secondary and light duty highways, (b) one rail line, (c) two airways, (d) intracoastal shipping lanes, and (e) sand mining operations (see Figure 2.2-1). Available data indicate that no other facilities exist within either a five mile radius or, in terms of significant facilities, a 10 mile radius of the plant (e.g., oil and gas pipelines, military bases, chemical plants, drilling operations, etc.)[1-11]

2.2.2 DESCRIPTIONS

2.2.2.1 Description of Facilities

There are no significant facilities within the plant vicinity that produce hazardous materials. For a description of nearby facilities, refer to Subsection 2.1.3.3.3.

2.2.2.1.1 Transportation Facilities

Roads

One primary highway (US 1), three secondary highways (SR A1A, SR 712, SR 707) and two light duty roads (Walton Road, Easy Street), are within five miles of the plant. As indicated by Table 2.2-1, average daily traffic volumes on these routes during peak seasonal times (fall and winter) during 1977-1978 ranged from 1,016 to 19,535. Increases in traffic volumes could be as high as six percent annually[12,13]. The shortest linear distance from the center of the Reactor Building to each of these highway corridors is listed below:

a) US 1 - 4.8 miles WSW

b) SR A1A - 0.2 mile E

c) SR 712 - 3.8 miles NW

d) SR 707 - 1.8 miles WSW

e) Walton Road - 3.4 miles SW

f) Easy Street - 3.1 miles NW

Rail

Paralleling the western shore of the Indian River (2.0 miles west southwest of the Reactor Building) is a Florida East Coast Railway. Descriptive statistics concerning this facility are provided below[14]:

a) Average daily number of trains - 21

b) Average train size - 50 to 55 cars

c) Maximum train size - No limit

d) Number of passenger trains - None

e) Commodities transported - Rock, autos, building materials, perishables, piggyback shipments (FAK/Freight of all kinds) and any hazardous materials meeting the tariff regulations of the Interstate Commerce Commission (ICC)

f) Tonnage shipped annually past the site - 7,959,098 tons

Airways

Two airways are located approximately two miles to the east of the plant: V259 and V3E. The two airways are used extensively by both IFR traffic (instrument flight rules; primarily commercial) and VFR traffic (visual flight rules; primarily private)[15].

Waterways

Commercial shipping lanes are located east and west of the plant. The Intracoastal Waterway is located 1.2 miles to the west of the plant. The St Lucie County portion of the Intracoastal Waterway (a north-south transportation route extending the length of the east coast) passes through the Indian River. Atlantic Ocean shipping lanes are about 10 to 15 nautical miles east of the plant; with north bound traffic lanes located farther east than southbound lanes[16].

2.2.2.1.2 Quarrying/Mining Operations

A small sand mining operation (employing two people) is located along the western shore of the Indian River approximately four miles northwest of the plant site. No explosives are employed by these operations[1,13].

2.2.2.2 Description of Products and Materials

2.2.2.2.1 Railroads

The Florida East Coast Railway may transport any hazardous material complying with ICC tariff regulations past St Lucie Unit 2. The principle explosive substance transported is liquid petroleum gas (maximum tank size, 33,000 gallons): the principle toxic substance transported is chlorine (maximum tank size, 90 tons)[17]. Such materials may be included on all trains.

Within the past 10 years, two minor rail accidents (both derailments) have occurred within five miles of the plant: (a) May 15, at milepost (MP) 248.2- "wrung journal" (i.e., broken axle) on car FEC 12295, (b) August 26, 1974 at MP 257.1 - brake rod broke, derailing car NW 292587. Neither incident involved hazardous materials and total recorded damage (involving only equipment and track) approximated $762.00[14].

2.2.2.2.2 Truck Carriers

No data were available on truck traffic or truck shipments within five miles of St Lucie Unit 2, although existing records indicate that no truck related accidents involving hazardous materials have occurred within the area[19].

Since there was very little information on truck traffic in the vicinity of the plant, the Applicant performed a survey[20] to get an indication of the amount and type of truck traffic on the roads within a five mile radius of the site. The survey was performed between January 30, and February 6, 1979 and consisted of collected information on US 1, SR A1A, SR 707, SR 712, and Walton Road. The survey consisted of the following:

a) A collection of existing information from the state and the county regarding traffic counts and accident characteristics.

b) Twenty-four hour truck classification counts taken at 13 locations. Each truck was classified by the number of axles and whether it was marked as carrying hazardous material. The type of hazardous material was also noted.

c) A roadside interview on US 1 of trucks marked as carrying material was conducted on US 1 for a total of 18.5 hours over a two day period. Information on the type and amount of hazardous material being carried by these vehicles was collected. The interview station was located on US 1 because it is the most heavily traveled roadway in the five mile area and has the majority of truck traffic of the roads in question.

d) Automatic traffic recorder counts were obtained at 10 locations for a seven day period.

e) Contacting propane gas supply companies in the area to determine if deliveries were made within the five mile radius; the type of material being transported; and size and capacity of these trucks.

The survey locations are shown on Figure 2.2-2. The results of the survey are given in Tables 2.2-2, 2.2-3 and 2.2-5 and shown on Figures 2.2-3 and 2.2-4.

The average daily traffic count and the average weekday traffic count determined from the survey are given in Table 2.2-2 and shown on Figure 2.2-3. The count data are consistently higher than the volume figures obtained from the State of Florida. The difference is apparently caused by the increase in seasonal activity on all routes in the area during this time of year (January-February), and in traffic volumes on SR A1A due mainly to construction activities at St Lucie Unit 2.

The truck volume and the volume of trucks displaying hazardous material placards are also represented in Table 2.2-2 and Figure 2.2-3 and are based on the trucks observed during the classification counts.

In summary, the trucks observed during the study period comprised from 1.3 to 6.4 percent of the total traffic. These values are comparable to the normal truck percentages, which are usually five percent[21]. Trucks carrying hazardous material comprised from 0 to 16.7 percent of the total truck traffic as indicated on Table 2.2-2.

A summary of the trucks marked hazardous and interviewed on US 1 is given in Table 2.2-3. Though this is a limited sample, it does give an indication of the type and amount of hazardous material transported within the five mile radius.

On SR A1A, the majority of the truck traffic services the St Lucie site. Table 2.2-4 is a description of the type, size and frequency of truck shipments of compressed gases and process chemicals to the St Lucie site.

In addition to interviews and classification counts, four propane gas companies known to make deliveries in the area were contacted to determine the type and amount of material being transported and the size and capacity of their trucks; the results are as follows:

a) Tropigas, Stuart - transports liquid propane gas in 2,000 gallon tanks on two axle and three axle trucks into the area of the five mile radius. They deliver once a month on SR 707 as far north as the plant site and they deliver once a month to "Venture Sales" located on SR A1A in the area of Nettles Island.

b) Tropigas, Fort Pierce - is similar to a) above. However, they deliver once a month during summer months and twice a month during winter months to locations along SR 712 and SR 707 as far south as the plant site.

c) Tri-County Gas, Inc, Stuart - transports liquid propane gas in quantities up to 2,150 gallons on three axle trucks into the area of the five mile radius. They presently service St Lucie Unit 2 once or twice a week for welding operations. They also service "American Resort" in the vicinity of Nettles Island once a month and deliver along SR 707 and SR 712 once a month.

d) Econ-O-Gas, Inc, Stuart - makes no deliveries into the site area.

Information on truck accidents is presented in Table 2.2-5 and Figure 2.2-4. Between January 1, 1973 and December 31, 1977, 19 accidents involving trucks occurred on US 1 and one accident occurred on SR A1A within the five mile radius. Between January 1, 1973 and December 31, 1976, one accident occurred on SR 707. Within the last five years there were no accidents within the St Lucie County involving hazardous material[20].

2.2.2.2.3 Waterborne Commerce

As is indicated in Table 2.2-6, 21 different types of commodities are regularly shipped past the site via the Intracoastal Waterway[22]. During 1975-1977, residual fuel oil constituted 56 percent of all shipments (by weight/tons). Other major types of commodities shipped past the site during this same period included nonmetallic mineral products (10.3 percent) and sugar (9.0 percent). Although no data are available concerning shipping in the Atlantic Ocean, the U.S. Coast Guard estimates that 40 to 50 ships pass the site each day. Approximately half of this traffic (i.e., 25 to 30 ships) is estimated to carry petroleum products[16].

2.2.2.2.4 Onsite Products and Materials

Compressed gases and process chemicals located on the St Lucie site for operation and maintenance purposes (and stored in standard industrial high pressure cylinders) include the following:

<u>Compressed gases</u>

a) Acetylene - approximately 25 bottles, (360 scf)

b) Oxygen - approximately 25 bottles, (360 scf)

c) CO_2 - approximately 80 bottles, (360 scf)

d) N_2 - 40,000 scf tube trailer
40 bottles, (360 scf)
Liquid Dwyer (several hundred gallons)

e) Hydrogen - 40,000 scf tube trailer
80 bottles, (260 scf)

<u>Process Chemicals</u>

a) Cyclohexylamine - two 55 gallon drums

b) Ammonium hydroxide - two 55 gallon drums

c) Hydrazine - eleven 55 gallon drums

d) Potassium Dichromate - 200 pounds

e) Sodium Hydroxide - 3800 gallons

f) Sulfuric Acid - 3000 gallons

<u>Other gases limited to a small number of bottles</u>

a) Argon

b) Methane

c) Propane

d) Laboratory specialty gases

2.2.2.3 Pipelines

No pipelines are located within five miles of the plant[3,4].

2.2.2.4 Waterways

The Intracoastal Waterway (10 foot channel depth) passes 1.2 miles west of the plant. All intake structures are located to the east of the plant and open into the Atlantic Ocean. Four types of vessels utilize the Intracoastal Waterway: (a) self propelled passenger and dry cargo vessels, (b) non-self propelled dry cargo vessels, (c) non-self propelled tankers, and (d) towboats and tugboats[22].

2.2.2.5 Airports and Airways

No major airports exist within 10 miles of the plant. Approximately nine miles west northwest of the site there is a private airport called Sunrise. Within nine and 50 miles of the plant, there are 26 airports[15]. Location data for all airports within 50 miles of the site are provided in Table 2.2-7. Based on available data, no airport within the 50 miles area records operations at or beyond levels of $500d^2$* (within 10 miles) or $1000d^2$ (within 10 to 50 miles)[23,24,25,26]. The estimated number of IFR flights occurring in airways V295 and V3E (calculated within a 20 mile radius of the site) is 250,000 annually; VFR traffic in this same area is estimated to equal 500,000 flights annually[27].

2.2.2.6 Projections of Industrial Growth

Between 1980 and the year 2000, light manufacturing activity in St Lucie County is projected to increase by approximately 111 percent[28]. In 1980, approximately seven percent of total earnings** or $13.1 million is expected to be derived from manufacturing activities. By the year 2000, earnings derived from light manufacturing ($27.7 million) are projected to approximate two percent of total earnings. During each of these benchmark years, approximately 60 percent of manufacturing-based earnings is expected to be derived from two industrial sectors: foods and kindred products, (primarily citrus) and chemicals and allied products (primarily fertilizers). Throughout this time frame, relatively small amounts of manufacturing activity are expected to occur in four additional industrial sectors: printing and publishing; metal fabrication; machinery manufacturing; and electrical equipment manufacturing.

Such increases in manufacturing activity as may occur in St Lucie County are expected to be contained in relatively high intensity nuclei located along major highways - especially I 95 and US I[29]. Given such a development strategy, increases in manufacturing activity within five miles of St Lucie Unit 2 may be anticipated primarily along US 1 in the vicinity of Port St Lucie (i.e., approximately five miles west of the plant).

* Refers to distance (d) in miles from the site.
** Earnings include income derived from wages, salaries, proprietary and miscellaneous income.

2.2.3 EVALUATION OF POTENTIAL ACCIDENTS

There are no design basis events external to the plant that have a probability of occurrence of 10^{-7} per year or greater and have potential consequences serious enough to affect the safety of the plant to the extent that 10CFR100 guidelines could be exceeded.

2.2.3.1 Explosions

2.2.3.1.1 Transportation of Explosives and/or Flammables on the Atlantic Ocean and Intracoastal Waterway

The Atlantic Ocean shipping lanes are about 10 to 15 nautical miles east of the plant (refer to Subsection 2.2.2.1.1). Hence, with a distance of 10 miles, no ship or barge explosion can affect the plant structures.

Due to the Intracoastal Waterway channel depth of 10 ft, the size of barges passing the plant site is limited. The waterway depth is nominally assumed to be capable of handling nine feet draft vessels which transport a maximum load of about 16,000 bbl[30]. However in actual practice, transporters are limited to loads of about 7,000 bbls per trip on barges of no more than a six feet draft because the Intracoastal Waterway is not dredged often[30]. As indicated in Table 2.2-6, the commodities of concern regarding explosions are gasoline and petroleum.

Gasoline is used as an example for calculating explosion overpressures. According to Robert F Benedict, the upper limit of flammability for gasoline is 7.9 percent[31]. The highest limit of flammability for the gasoline family stated by the Bureau of Mines is 10.5 percent for cyclopropane[32]. Mr. Benedict has stated that although the density of gasoline vapor at the highest limits of flammability is unavailable, the combination of a 10.5 percent limit of flammability and a gasoline vapor density of 0.245 lbm/ft^3 (which corresponds to the vapor density of heptane) at this limit is conservative.

The free volume of a 16,000 bbl barge is 16,000 bbl x 42 gal/bbl x 0.1337 ft^3/gal = 89,846 ft^3. Using a conservative 10.5 percent gasoline-air mixture (i.e., 0.105 of volume) at a vapor density of 0.245 lbm/ft^3, there are 2311 lbm of gasoline in a 16,000 bbl barge. Assuming an extremely conservative upper bound of mass equivalency at 240 percent, 2311 lbm of gasoline vapor yields a detonation equivalent to 5547 lbm TNT.

From Equation 1 of Regulatory Guide 1.91, "Evaluations of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plants," February 1978 (R1), the calculated safe overpressure distance is

$$R \geq k\, W^{1/3} \geq 45\ (5547)^{1/3} \geq 797 \text{ ft.}$$

where:

R = distance in feet from an exploding charge

W = pounds of TNT

K = constant = 45

Therefore a one psi peak positive overpressure will not occur at a distance greater than 797 ft. for a 16,000 bbl barge of gasoline vapors. Since the Intracoastal Waterway shipping channel is over 6000 ft away from any safety related structures, no damage occurs from a barge explosion.

Explosion generated missiles are also considered as follows:

In order to determine the distance through which generated missiles can travel, consider the exploding volume of 89,846 ft^3 as a sphere (radius = 27.78 ft^3) with energy equipartitioned through the exploding volume. The explosive energy (E_e) assuming 5547 lbm TNT is

$$E_e = 5547 \text{ lbm} \times 500 \frac{\text{kcal}}{\text{lbm}} \times 3.968 \frac{\text{Btu}}{\text{kcal}} = 1.1 \times 10^7 \text{ Btu}$$

and the energy density (E) is

$$E = \frac{E_e}{\text{free volume}} = \frac{1.1 \times 10^7 \text{ Btu}}{89{,}846 \text{ ft}^3} = 122.5 \text{ Btu/ft}^3$$

In a deflagration type of explosion the maximum energy density imparted to potential missiles cannot exceed the energy of the explosion. Hence, the kinetic energy (KE) of a potential missile cannot exceed

$$KE(\text{Btu}) = E(\text{Btu/ft}^3) \times M_m(\text{lbm}) \div \rho_m \text{ (lbm/ft}^3)$$

where M_m and ρ_m are the mass and density of the potential missile. With a kinetic energy (KE) = $1/2\ M_m V^2$, where V is the missile speed, the maximum range (R_{max}) of a missile is

$$R_{max} = \frac{V^2}{g} = \frac{2\ KE}{M_m} \times \frac{1}{g} = \frac{2E\ M_m}{\rho_m M_m} \frac{1}{g} = \frac{2\ E}{\rho_m g}$$

Using steel as an example with a density of 489 lbm/ft^3, the maximum range calculated of a potential missile is

$$R_{max} = \frac{2 \times 122.5 \frac{\text{Btu}}{\text{ft}^3} \times 778 \frac{\text{ft-lbf}}{\text{Btu}} \times 32.2 \frac{\text{lbm-ft}}{\text{lbf-sec}^2}}{32.2 \text{ ft/sec}^2 \times 489 \text{ lbm/ft}^3} = 389.8 \text{ ft}$$

The above equation does not include consideration of air resistance or energy lost in rotation, which would decrease the range of any generated missile. Thus there is no hazard from a barge explosion due to missiles.

2.2.3.1.2 Transportation of Explosives and/or Flammables by Truck or State Road A1A

A review of the truck traffic reveals that the governing explosive and/or flammable event would arise on SR A1A which passes about 750 ft east of the diesel oil storage tanks due to a liquified propane truck accident. Based on the limited amount of hazardous truck movements past the site, the probability of having a potential accident whose consequence can result in radionuclide releases in excess of 10CFR100 guidelines is less than 10^{-7} per year as described below.

Based on accident data for a five year period (January 1, 1973 through December 31, 1977) provided by the Florida Department of Transportation, there has been only one truck accident within five miles of the site on SR A1A in a total of 2,600,000 truck vehicle miles traveled[20]. Therefore, the probability of any type of truck accident is calculated to be 3.8×10^{-7} truck accidents per vehicle mile. This site specific probability is much smaller than the 1.3×10^{-6} truck accidents per vehicle mile probability for a tank truck accident in the "minor" severity category predicted in WASH-1238[35]. Therefore the WASH-1238 probability is used and gives a conservative estimate of the frequency of truck accidents in the site vicinity.

To calculate the probability that hazardous flammable liquids explode due to a spill, it is necessary to determine the conditional probability of a spill and the conditional probability of an explosion occurring due to a spill. Although there have been no accidents within St Lucie County involving hazardous material within a period of 1973-1978[20], the probability of a spill as a result of an accident is estimated at 0.02[36], since two percent of accidents involve a tank truck with sufficient impact to cause rupture of tank. The probability of an explosion due to a spill as determined by the Department of Transportation's Office of Hazardous Materials is 0.0113[37].

Thus, the probability(Pe) associated with an in-transit explosion of a truck is $1.3 \times 10^{-6} \times 0.02 \times 0.0113 = 3.6 \times 10^{-10}$ explosions per vehicle mile.

The number of vehicle miles per year for the transport of hazardous material in the one mile stretch of SR A1A incident to the site can be estimated. The annual number of liquified propane gas truck deliveries on SR A1A in the vicinity of the site is 27 shipment/yr (as described in Subsection 2.2.2.2.1). Assuming all these trucks travel through the one mile stretch of road incident to the site, 27 vehicle mile/yr can be estimated. Using the probability of 3.6×10^{-10} explosions/vehicle mile, the probability of an explosion is 9.72×10^{-9} per year, in the one mile stretch of SR A1A closest to the site.

Since the probability is less than 10^{-7} per year, an explosion of a truck carrying hazardous material is not a design basis event.

2.2.3.1.3 Transportation of Explosives and/or Flammables on the Florida East Coast Railway

The Florida East Coast Railway runs about two miles west southwest of the plant site (refer to Subsection 2.2.2.1.1). Since the rail line can approach the safety related structures no closer than the distances computed in Figure 1 of Regulatory Guide 1.91(R1), no further consideration need be given to the effects of blast in plant design. This two mile distance is greater than the ranges of fragments of the train accident in Laurel, Mississippi[38]. The range of the "rocketing" car in the Laurel, Mississippi accident was 1100 ft while small fragments had a maximum range of 1600 ft.

Thus there are no hazards from "rocketing" rail cars or their fragments for St Lucie Unit 2 safety related structures.

2.2.3.2 Design Basis Toxic Chemical Events

2.2.3.2.1 Introduction

The accidental release of toxic chemicals may affect control room habitability. Based on information presented in Subsection 2.2.2, the potential sources are analyzed in detail to determine the threat to the control room operators.

Table 2.2-8 contains a list of each of the toxic chemical stored or transported in the vicinity of the plant. Those specific events which are found to have a probability of less than 10^{-7} per year are considered not to be design basis events.

2.2.3.2.2 Assumptions and Methodology

Based on information presented in Subsection 2.2.2, Table 2.2-8 includes a list of hazardous chemical sources which are considered in evaluation of potential accidents. Consideration is limited to those chemicals which are present within a distance of five miles from the control room air intakes. Chemicals stored or situated at distances greater than five miles from the facility are not considered because, if a release occurs at such a distance, wind speed and atmospheric dispersion will dilute and disperse the incoming plume to such a degree that there will be sufficient time for the control room operators to take appropriate action, if any is required. In addition, the probability of a plume remaining within a given sector for a long period of time is quite small.

Facilities located within five miles of the plant do not store, use or produce large quantities of hazardous substances. However, some quantities are stored on site as indicated in Table 2.2-8. There are no toxic chemicals transported by waterborne commerce and road in significant quantities that may affect the safety of the plant following accidental releases. Consequently such sources are not evaluated. There are no pipelines located within five miles of the plant and so this source is also not considered. The amounts of toxic chemicals transported by the Florida East Coast Railway (FECR) are greater than those specified in Regulatory Guide 1.78. "Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release", June 1974 (RO). Therefore releases of toxic chemicals due to railroad accidents are considered in the analysis.

In order for the control room operators to become exposed to one of the toxic chemicals listed in Table 2.2-8, the following chain of events must occur. First, the container in which a given chemical is enclosed must somehow fail and release its contents. Second, the chemical must be sufficiently volatile to become airborne. Third, at the time of release, the direction of the wind must be such as to transport the airborne material from the point of release to the control room outside air intakes. The airborne material has to be sufficiently stable in air not to condense on the ground, or burn or explode, or otherwise lose its toxicity prior to reaching the outside air intake. The quantity of the chemical which becomes airborne has to be sufficiently large and dispersion in air sufficiently low, for the concentration of the toxic agent to build up to toxic levels in the control room atmosphere before the operators can take protective action.

Chemicals that are nonvolatile solids or liquids, or that spontaneously combust in air do not pose a threat to control room habitability. Consideration of these factors leads to the elimination of the following chemical sources from toxic hazard evaluation. Solutions of sodium hydroxide and potassium dichromate are eliminated because, while the solvent may evaporate, the solute is nonvolatile. Sulfuric acid is eliminated due to its low volatility. It is an oily liquid with a vapor pressure of only 0.0008 mm Hg at 25 C and its evaporation rate is negligible under ambient atmospheric conditions. Similarly hydrazine stored on site is eliminated because its partial vapor pressure in the solution of 30 percent concentration is only 3.7 mm Hg under ambient conditions.

Table 2.2-8 also indicates that many chemicals are eliminated because their potential for ignition constitutes a greater hazard than their toxicity. When a flammable or explosive substance is released, it is highly likely that its vapor will explode or burn before reaching the control room. Therefore, the only chemicals considered to present a potential danger to control room operators, are those whose toxicity limits are lower than their lower limits of flammability. This leads to the elimination of chemicals such as hydrogen, acetylene, natural gas, propane, butane, and other flammable hydrocarbons.

These toxic chemicals in Table 2.2-8, which are not eliminated on the basis of criteria discussed above are shown to pose no threat to control room habitability by a detailed assessment of their atmospheric transport and potential for infiltrating into the control room atmosphere. The atmospheric dispersion condition is conservatively assumed to be stability Class F and 1.0 m/sec wind speed.

Ammonium hydroxide is stored onsite in two 55 gallon drums at 30 percent concentration by weight. The concentration at the outside air intakes of the control room is calculated assuming all the ammonia in the solution becomes airborne instantaneously following a postulated rupture of the container. This very conservative assumption results in concentration at the outside air intakes well in excess of that which can be actually expected.

In case of carbon dioxide, complete vaporization is assumed immediately following accidental release. The airborne transport of the puff is modeled using the instantaneous release diffusion model presented in Regulatory Guide 1.78 (RO). Since the control room is located at a short distance from the release point and the amount of chemical is small, the model is adjusted to allow for additional dispersion in the vertical direction by assuming uniform mixing between the ground and the elevation of the fresh air inlet (a 23 meter elevation from ground level is used).

The concentration of cyclohexylamine in the feed tank in the Turbine Building is 10 percent. The chemical is delivered to the site in 55 gallon drums in the concentrated form and stored in a 20 ft by 12 ft storage room. Conservatively it is assumed that a 55 gallon drum fails. The evaporation rate of cyclohexylamine is calculated using equation 2.1-18 in NUREG

0570[39]. The transport of vapor is modeled by the short term, continuous release diffusion equation presented in Regulatory Guide 1.4 "Assumptions Used for Evaluating the Potential Radiological Consequences of a LOCA for PWR", June 1974 (R2). It is conservatively assumed that the centerline of the plume remains incident on the control room outside air intakes during the entire time it takes the liquid to evaporate. A ground level release is assumed. However, the credit for additional dispersion in the vertical direction by assuming uniform mixing between the ground and the elevation of the outside air intake is taken.

The models described above are used to calculate concentrations of toxic chemicals at the control room outside air intakes. As indicated in Table 2.2-8, the concentration of ammonia at the outside air intake of the control room is the only chemical expected to exceed the toxicity limit following the rupture of a 55 gallon drum containing 30 percent ammonia by weight. Since carbon dioxide and cyclohexylamine concentrations at the outside air intakes are below the toxicity limits, the concentrations inside the control room are not required to be calculated.

The ammonium hydroxide concentations inside the control room as calculated based on the following equation (see Appendix 15B):

$$C(t) = e^{-vt} \int_0^t v e^{vt'} X(t')\, dt'$$

where:

C (t) = chemical concentration inside the control room at time t

X (t') = chemical concentration outside the air intake at time t'

v = control room air enchange rate of 0.46 per hour which is based on normal air intake rate of 750 cfm

Based on the above equation, the concentration of ammonia inside the control room remains well below the toxicity limit under the assumptions that the control room is not isolated and no action is taken by the operators to do so following the accident. Therefore, ammonium hydroxide stored onsite poses no threat to the control room operators.

Chlorine is the principal toxic substance transported by the FECR (2.0 miles west south west of the plant). Since the quantity, pershipment, of chlorine (90 tons) shipped past the site is greater than the adjusted quantity given in Table C-2 of Regulatory Guide 1.78 (RO), the shipments are considered in the hazardous chemical analysis.

There have been a few minor railroad accidents within five miles from the plant in the past 10 years which resulted in small damages. Based on information presented in Regulatory Guide 1.78 (RO), releases in the amount of 30 tons or more of chlorine at the railroad require consideration in an evaluation of the control room habitability. A release of such magnitude is assumed to be equivalent to the total loss of a railroad car with a capacity of 90 tons or less, i.e., the accidental release of the entire contents of chlorine from a tank car is assumed to be an initiating event for a design basis accident.

The probability of such an event is given by the following equation:

$$P_{il} = P \times N_i \times M_l \times \sum_{j=1}^{n} D_j \times F_{jl}$$

where:

Pil = annual probability of design basis event under atmospheric stability Class l involving the i-th chemical.

P = probability of a design basis accident for a mobile source

N_i = annual numbers of trips involving the i-th chemical.

M_l = annual probability of an atmospheric stability class.

D_j = the length of road, rail or river in sector j.

F_{jl} = wind frequency from sector j to outside air intake of the control room for stability Class l.

n = number of wind direction sectors.

Based on the number of trains, movements, cars per train (see Subsection 2.2.2.1.1) and the length of the track near the site (approximately 9.2 miles), there are approximately 3.52×10^6 railroad car-miles traveled per year within five miles of the site. Within the past 10 years, two rail accidents have occurred within five miles of the plant. Therefore the frequency of an accident at the site is 5.68×10^{-8} per mile. National statistics indicate that 33×10^{-9} events[40] per mile result in total loss of chlorine contents from a car. The national statistics also indicate that the frequency of rail road accident[35] is 8.1×10^{-7} per mile. Therefore the frequency of an event that results in the total loss of the contents from a car is 4.1×10^{-2} ($=33 \times 10^{-9}/8.1 \times 10^{-7}$) per accident. Assuming the same relationship is applicable in the vicinity of the plant, the probability of a design basis accident P, is 2.33×10^{-9} ($=5.68 \times 10^{-8} \times 4.1 \times 10^{-2}$) per mile of travel distance. An average number of 14 cars carrying chlorine are shipped per month by the FECR[41]. Therefore the annual number of trips, Ni, involving chlorine are 168.

Table 2.3-81 indicates that the atmospheric stability frequency, M_l, is 4.16 and 1.13 percent for stability Classes F and G, respectively. Although atmospheric stability classes A through E are considered, the control room habitability is not affected under such meteorological conditions. The length of each segment of railroad, D_j, within each sector is shown on Figure 2.2-5. The required wind direction from each segment of railroad towards the outside air intakes of the control room is also shown on Figure 2.2-5. The wind frequency F_{jl}, for stability Class F and G is obtained from Table 2.3-34. The probability P_{il}, of a design basis event under stability Class F in each segment is then:

Segment 1-2: $P_i = 2.33 \times 10^{-9} \times 168 \times 0.0416 \times 0.0617 \times 2.66 = 2.67 \times 10^{-9}$ /yr

Segment 2-3: $P_i = 2.33 \times 10^{-9} \times 168 \times 0.0416 \times 0.1122 \times 1.14 = 2.08 \times 10^{-9}$ /yr

Segment 3-4: $P_i = 2.33 \times 10^{-9} \times 168 \times 0.0416 \times 0.0813 \times 0.823 = 1.09 \times 10^{-9}$ /yr

Segment 4-5: $P_i = 2.33 \times 10^{-9} \times 168 \times 0.0416 \times 0.0631 \times 0.835 = 8.58 \times 10^{-10}$ /yr

Segment 5-6: $P_i = 2.33 \times 10^{-9} \times 168 \times 0.0416 \times 0.0449 \times 1.2 = 8.77 \times 10^{-10}$ /yr

Segment 6-7: $P_i = 2.33 \times 10^{-9} \times 168 \times 0.0416 \times 0.0757 \times 2.51 = 3.09 \times 10^{-9}$ /yr

Total 1.07×10^{-8} /yr

The probability of an event under stability Class F for the entire hazardous travel distance of 9.2 miles is the sum of the values calculated above and is 1.07×10^{-8} per year. Similarly, the probability of an event under stability Class G is calculated to be 3.27×10^{-9} per year. Therefore an overall probability of an event that may affect control room habitability is 1.4×10^{-8} per year. Since the probability is less than 10^{-7} per year, the release of chlorine due to a railroad accident is not a design basis event.

2.2.3.2.3 Results

The accidental releases of chemicals stored on site and transported in the vicinity of the plant are found not to present undue risk to control room operators. Therefore, no detailed analysis is required in Section 6.4.

2.2.3.3 Fires in the Vicinity of the Site

There are no industrial and chemical plants or storage facilities, or pipelines containing oil or gas adjacent to St Lucie Unit 2. The potential hazard from fires offsite are negligible because no flammable mass of appreciable size exists in the area.

In the unlikely event that a barge spills oil or gasoline accidentally on the Intracoastal Waterway, the spill would not only have to travel to the Hutchinson Island shoreline (approximately 3000 ft) but would have to travel across 3000 ft of Big Mud Creek, basically a stagnant body of water. The ultimate heat sink barrier will stop the flow of water from Big Mud Creek to the intake structure. Therefore, it is highly improbable that such a fire could affect the St Lucie site.

2.2.3.4 Collisions with Intake Structure and Liquid Spills

Because the plant cooling water intake structure is located in a commercially non-navigable area offshore in the Atlantic Ocean, no reasonable hazard exists from barges or ships that pass the site and no corrosive liquids or oils accidentally released could enter the intake structure.

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TABLE 2.2-1

AVERAGE DAILY AUTO AND TRUCK TRAFFIC COUNTS
DURING PEAK (FALL AND WINTER) SEASON
1977-1978

HIGHWAY NO.	HIGHWAY SEGMENT	TRAFFIC VOLUME
U S 1	712 to Walton Road	19,535
SR 712	U S 1 to SR 707	6,406
SR A1A	Ft. Pierce to Martin Co.	2,731
SR 707	Walton Road to Martin Co.	2,072
	SR 712 to Walton Road	1,016
Walton Road	-	NA
Easy Street	-	NA

SOURCE: St. Lucie County. Traffic Corridors Input Data, 1978.

NOTE: Separate truck counts do not exist for the area within five miles of the plant.

TABLE 2.2-2

RESULTS OF TRUCK TRAFFIC SURVEY

Station[1] Number	Average Daily Traffic[2]	Average Weekday Traffic[2]	Number of Trucks Counted	Number of Trucks Classified as Carrying Hazardous Material	Percentage of Trucks of Weekday Traffic (%)	Percentage of Trucks Carrying Hazardous Material (%)
1	6,895	7,262	129	4	1.8	3.1
2	- [5]	- [5]	210	6	6.1[3]	3.2
3	- [5]	- [5]	187	6	5.5[3]	3.2
4	2,511	2,574	50	1	1.9	2.0
5A	1,543	1,537	24	4	1.6	16.7
5B	1,871	1,953	26	4	1.3	15.4
5C	1,687	1,711	27	0	1.6	0
6A	5,163	5,289	340	8	6.4	2.4
6B	- [5]	- [5]	859	43	4.5[4]	5.0
7	17,641	19,170	1,187	32	6.2	2.7
8A&B	1,305	1,380	19	0	1.4	0
9	3,035	3,418	- [6]	- [6]	- [6]	- [6]

Notes:

[1] See Figure 2.2-2 for locations of stations.
[2] All counts are non-directional.
[3] Based on Average Weekday Traffic recorded at Station No. 9.
[4] Based on Average Weekday Traffic recorded at Station No. 7A.
[5] As indicated on Figure 2.2-2 - automatic traffic recordings were made at this location.
[6] As indicated on Figure 2.2-2 - classification counts and roadside interviews were not made at this location.

Source: Champagne Associates, Vehicle Classification and Product Containment Study, February, 1979.

TABLE 2.2-3

HAZARDOUS MATERIALS FROM TRUCKS

INTERVIEWED ON US 1

JANUARY 30 AND 31, 1979

Propane	Total	5 Vehicles
	Maximum size (gallons)	2604
	Average size (gallons)	1400
Gasoline	Total	18 Vehicles
	Maximum size (gallons)	8300
	Average size (gallons)	3100
Non-Flammable Gas:		
(A)	Total	1 Vehicle
	Size (ft^3)	12000
(B)	Total	1 Vehicle
	SIze (lbm)	230
(C)	Total size (gallons)	1 Vehicle
	Size (gallons)	100
Oxygen	Total	1 Vehicle
	Size (ft^3)	10,000
Batteries	Total	2 Vehicles
	Maximum transported	400
	Average transported	300
Diesel Oil	Total	9 Vehicles
	Maximum size (gallons)	8100
	Average size (gallons)	1260
Bottled Gas	Total	1 Vehicle
	size (gallons)	50
Chlorine	Total	2 1/2 Vehicles
	Maximum (gallons)	2400
	Average (gallons)	970

TABLE 2.2-3 (Cont'd)

Muriatic Acid	Total	1/2 Vehicle
	Size (gallons)	500
Combustible	Total	1 Vehicle
	Size (total)	6000
Corrosives	Total	1 Vehicle
	Size (gallons)	3600
	Total Number of Vehicles:	44

Source: Champagne Associates, Vehicle Classification and Product Containment Study, March, 1979

TABLE 2.2-4

TRUCK DELIVERIES (COMPRESSED GASES, PROCESS CHEMICALS) TO ST LUCIE UNITS 1 AND 2

MATERIAL	SHIPMENT FREQUENCY	SHIPMENT METHOD	QUANTITY SHIPPED
Acetylene	weekly	5 ton open truck	5-10 cylinders
Oxygen	weekly	5 ton open truck	5-10 cylinders
CO_2	semi-annually	5 ton open truck	80 cylinders
N_2 - trailers	bi-monthly	Tube trailer	40,000 scf ea. load
N_2 - bottles	3 times/year	5 ton open truck	20-30 bottles
N_2 - liquid	monthly	Liquid N_2 tanker	1100 gal.
Argon	3 times/year	5 ton open truck	1-2 cylinders
Methane	3 times/year	5 ton open truck	1-2 cylinders
Propane	3 times/year	5 ton open truck	1-2 cylinders
Specialty gases	3 times/year	5 ton open truck	1-2 cylinders
Cyclohexylamine	semi-annually	Closed semi-trailer	2 drums
Ammonium Hydroxide	semi-annually	Closed semi-trailer	2 drums
Hydrazine	bi-monthly	Closed semi-trailer	6 drums
Potassium Dichromate	semi-annually	UPS	200 lbs
Sodium Hydroxide	monthly	Tank truck	3,800 gallons
Sulfuric Acid	monthly	Tank truck	3,000 gallons
Chlorine	monthly	Open tractor trailer	4 tons
Hydrogen-trailer	bi-monthly	Tube trailer	75,000 scf
Hydrogen-bottles	bi-monthly	5 ton open truck	20-30 bottles

TABLE 2.2-5

TRUCK ACCIDENTS WITHIN FIVE MILES OF ST LUCIE UNIT 2
1973-1977

Location*	Date of Accident	Number of Vehicles Involved	Type of Truck** Involved	Type of Damage	Amount of Property Damage
A	1973	2	SU	PDO	$10,000
B	1973	1	T-T	I	$ 2,000
C	1973	2	SU	PDO	$ 150
D	1973	2	T-T	I	$ 2,700
E	1973	2	SU	PDO	$ 150
F	1973	1	SU	I	$ 2,100
G	1974	3	T-T	PDO	$ 300
H	1974	2	T-T	F	$ 1,900
I	1974	2	SU	PDO	$ 175
J	1974	2	SU	I	$ 2,300
K	1977	2	T-T	PDO	$ 900
L	1977	3	SU	PDO	$ 2,500
M	1977	2	T-T	PDO	$ 800
N	1977	2	SU	PDO	$ 350
O	1976	2	SU	PDO	$ 300
P	1973	2	T-T	PDO	$ 800
Q	1973	2	SU	PDO	$ 1,600
R	1974	1	SU	PDO	$ 100
S	1976	2	T-T	I	$ 9,000
T	1973	1	SU	PDO	$ 5,000
U	1977	2	SU	PDO	$ 900

*See Figure 2.2-4.
**No accident involved more than one truck.

Legend:

SU - Single Unit Truck
T-T - Tractor Trailer
I - Injury
F - Fatality
PDO - Property Damage Only

Source: Champagne Associates, Vehicle Classification and Product Containment Study, March, 1979

TABLE 2.2-6

FLORIDA POWER & LIGHT COMPANY
COMMODITY MOVEMENTS - 1975, 1976, & 1977
DOMESTIC (ONLY) WATERBORNE COMMERCE
PASSING THE APPROXIMATE LOCATION OF THE
ST LUCIE UNIT 2

Type of Commodity	Type of Vessel	Short Tons per Calendar Year 1975	1976	1977	Totals
Ships and Boats	Passenger & Dry Cargo - Self-propelled	748.0	4,030.0	12,832.0	17,610.0
Fresh Fish, except Shellfish	"		15.0	22.0	37.0
Ice	"		11.0	5.0	16.0
Misc Products of Manufacturing	"			50.0	50.0
Furniture and Fixtures	Dry Cargo - Non-Self-propelled	28.0			28.0
Misc Non-metallic Mineral Prod.	"	19,615.0	9,372.0	7,203.0	36,190.0
Iron and Steel Bars, Rods, Angles, Shapes and Sections, Including Sheet Piling	"	480.0	75.0	650.0	1,205.0
Iron and Steel Pipe and Tube	"	1,395.0		300.0	1,695.0
Fabricated Metal Products except Ordnance, Machinery, and Transportation Equipment	"	3,801.0	2,940.0	5,225.0	11,966.0
Machinery except Electrical	"	6,753.0	2,955.0	7,056.0	16,764.0
Electrical Machinery, Equipment and Supplies	"	1,190.0	3,050.0	1,610.0	5,850.0
Aircraft and Parts	"	33.0		280.0	313.0
Ships and Boats	"	28.0		275.0	303.0
Misc Shipments not Identifiable by Commodity	"	300.0	615.0	2,255.0	3,170.0
Sugar	"		2,700.0	29,050.0	31,750.0
Aluminum and Aluminum Alloys, Unworked	"		4,000.0	5,000.0	9,000.0

TABLE 2.2-6 (Cont'd)

Type of Commodity	Type of Vessel	Short Tons per Calendar Year 1975	1976	1977	Totals
Basic Textile Products, except Textile Fibers	Dry Cargo - Non-Self-propelled			50.0	50.0
Timber, Posts, Poles, Piling, and other Wood in the Rough	"			100.0	100.0
Iron and Steel Scrap	"			2,170.0	2,170.0
Sodium Hydroxide - (Caustic Soda)	Tanker - Non-Self-propelled	3,987.0	4,046.0	4,098.0	12,131.0
Gasoline, including Natural Gas	"	100.0			100.0
Residual Fuel Oil	"	48,273.0	91,269.0	57,251.0	196,793.0
Asphalt, Tar, and Pitches	"		2,812.0		2,812.0
	TOTALS	86,731.0	127,890.0	135,482.0	350,103.0

Source: Department of the Army, Lower Mississippi Valley Division Corps of Engineers, Waterborne Commerce Statistics Center, January 10, 1979.

TABLE 2.2-7

AIRPORTS WITHIN 9-50 MILES OF ST LUCIE UNIT 2

AIRFIELD	DISTANCE AND DIRECTION FROM SITE (STATUTE MILES)*
Civil-Public Use	
Valkaria	45 NNW
Sebastian	34 NNW
Vero Beach	22 NNW
St Lucie Co.	12 NW
Witham	13 S
Palm Beach Gardens	38 S
Palm Beach International	48 SSE
Okeechobee Co.	37 W
Circle T Ranch	25 SSW
Palm Beach Co., Glades	48 SW
Private	
Fellsmere	36 NW
Broocke	27 NNW
New Hibiscus	25 WNW
Indian River	22 WNW
Nelson	11 NW
Hawgwild	11 NW
Peacock Ranch	13 WSW
Naked Lady Ranch	16 SSW
Tropical Plantation	18 S
Chem	42 SSW
Evans	21 SW

TABLE 2.2-7 (Cont'd)

Mulgrew Ranch	40 WSW
Sunset	41 W
Indian Hammock	38 WNW
Sunrise	9 WNW
Palm Beach Ranch Groves	37 S
Heliports	
Sikorsky (Private)	32 SSW

SOURCE: United States Department of Commerce, National Oceanic and Atmospheric Administration. Sectional Aeronautical Chart, Miami, September 1978.

* To nearest mile, measured from Sectional Aeronautical Chart, Miami.

TABLE 2.2-8

TOXIC CHEMICAL EVALUATION

Source and Type of Toxic Chem	Toxicity Limit/(Ref) (ppm)	Distance From Main Control Room (Feet)	Distance From Main Control Room (Direction)	Quantity Released	Peak Concentration (ppm) OAI of Control Rm	Peak Concentration (ppm) Inside Control Rm	Remarks
Onsite Storage:							
Acetylene		600	NW	360 SCF			Note 4
Ammonium Hydroxide	500/(4)	260	NW	55 gal; 30 by weight	4.84×10^4	105	Note 2
Carbon Dioxide	1.0×10^4/(3)	600	NW	360 SCF	2.08×10^3	*	Note 1
Cyclohexylamine	20/(6)	180	NW	55 gal; 100% conc	1.61×10^1	*	Note 10
Hydrazine-Amerzinc Chem Feed System	5/(5)	180	NW	400 gal; 35% conc by weight	-	-	Note 9
Hydrazine-Iodine Removal	5/(5)	60	SW	550 gal; 5% conc by weight	-	-	Note 9
Hydrogen		640	NW	260 SCF	-	-	Notes 4,5
N_2-gas		590	NW	360 SCF	-	-	Note 5
N_2-liquid		620	NW	1100 gal	-	-	Note 5
Potassium Dichromate-TCCWS		60	W	100 gal	-	-	Notes 3,6
Potassium Dichromate-CCWS		220	ENE	50 gal	-	-	Notes 3,6
Sodium Hydroxide		690	NW	10,000 gal; 50% conc by weight	-	-	Notes 3,6
Sulfuric Acid	3.0 $\frac{mg}{m^3}$/(5)	700	NW	10,000 gal; 60° Baume	-	-	Note 3

* Concentration inside control room not determined since concentration at outside air intake is below toxicity limit

TABLE 2.2-8 (Cont'd)

Source and Type of Toxic Chem	Toxicity Limit/(Ref) (ppm)	Distance From Main Control Room (Feet)	(Direction)	Quantity Released	Peak Concentration (ppm) OAI of Control Rm	Inside Control Rm	Remarks
Road:							
Chlorine	15/(7)	4.8 miles	WSW	2,430 gal(2.8×10^4 lbs)	-	-	Note 7
Combustibles		4.8 miles	WSW	6,000 gal	-	-	Note 4
Rail:							
Chlorine	15/(7)	2.0 miles	E	90 tons	-	-	Note 8
Liquid Petroleum Gas		2.0 miles	E	33,000 gal	-	-	Notes 4,5
River:							
Gasoline		1.2 miles	W	16,000 bbl	-	-	Notes 4,5
Sodium Hydroxide		1.2 miles	W	16,000 bbl	-	-	Notes 3,6

Notes

1) Not a design basis event because the calculated concentration at the outside air intake of the control room is less than the toxicity limit.

2) The concentration at the outside air intake of the control room calculated assuming all the toxic chemicals in the solution becomes airborne instantaneously following the rupture of the container.

3) Not volatile

4) Primarily a fire hazard

5) Simple asphyxiant. Toxic effect occurs at 33 percent volume in air.

6) Toxic agent is a solid under ambient conditions.

7) Quantity of toxic chemical, at the given distance, is less than the maximum specified in Regulatory Guide 1.78 (RO), Table C-2.

8) Not a design basis event. The probability of occurrence, as indicated in Section 2.2.3 is less than 10^{-7} per year.

9) The partial vapor pressure of the toxic chemical in the solution is less than 10 mm Hg. The event eliminated based on guidelines provided in Regulatory Guide 1.78 (RO).

10) Concentration at the outside air intakes based on guidelines provided in Section 2.2 Reference (1).

TABLE 2.2-8 (Cont'd)

References for Toxicity Limits

1) Criteria for a Recommended Standard - Occupational Exposure to Acetylene, DHEW/PBU/NIOSH-76/195.

2) Sax, N Irving. Dangerous Properties of Industrial Materials, Third Edition, Reinhold Book Corp., New York, 1968.

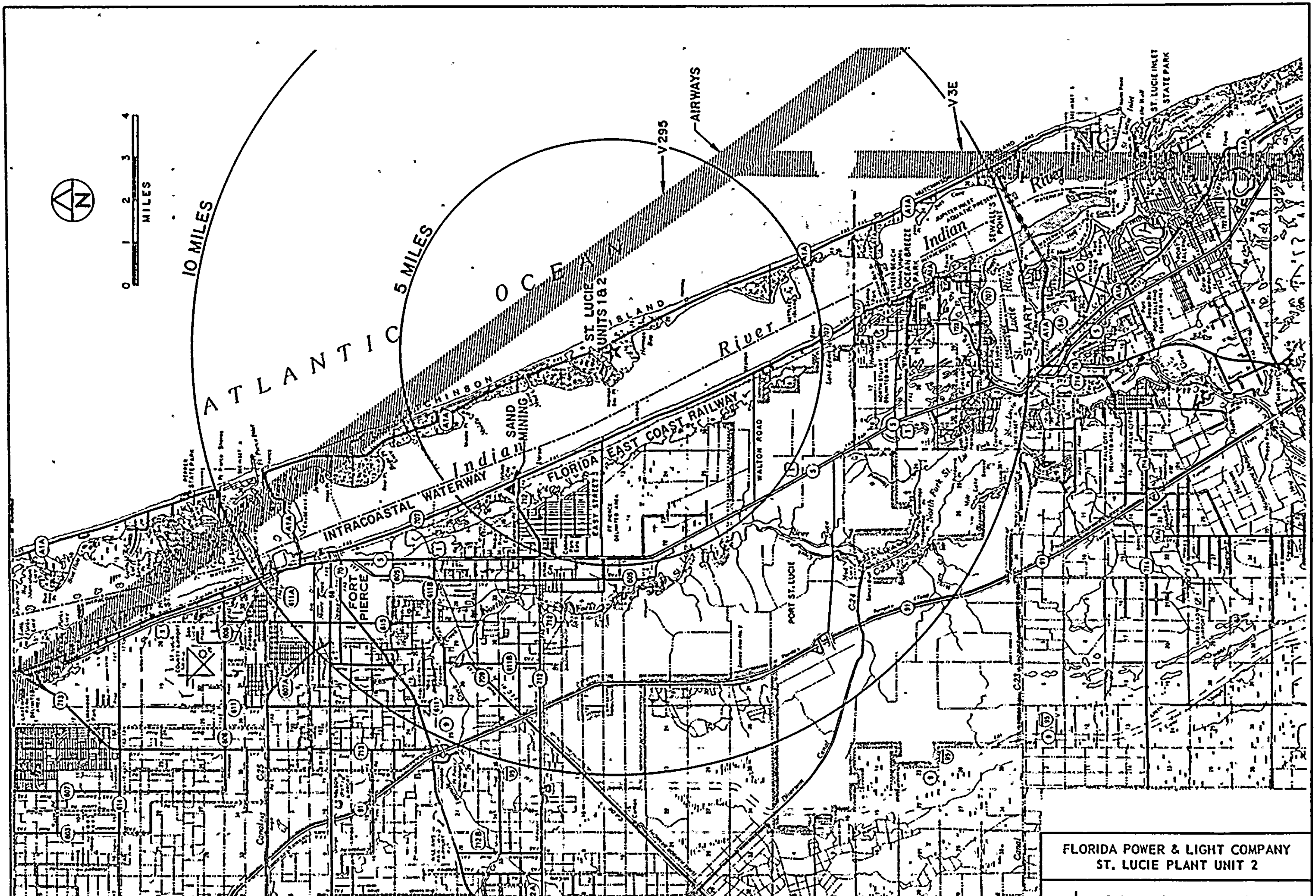
3) Regulatory Guide 1.78, "Assumptions for Evaluating the Habitability of Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release."

4) Criteria for a Recommended Standard - Occupational Exposure to Ammonia. DHEW/PUB/NIOSH 74-136. NTIS-PB-246 699.

5) Patty, Frank, A. Industrial Hygiene and Toxicity, Vol II - Toxicity (2nd Edition Revised), Interscience Publishing Co. New York, 1963.

6) Karel Verschueren. Handbook of Environmental Data on Organic Chemicals, Van Nostrand Rheinhold Company, New York.

7) Criteria for a Recommended Standard - Occupational Exposure to Chlorine. DHEW/PUB/NIOSH 76-170.



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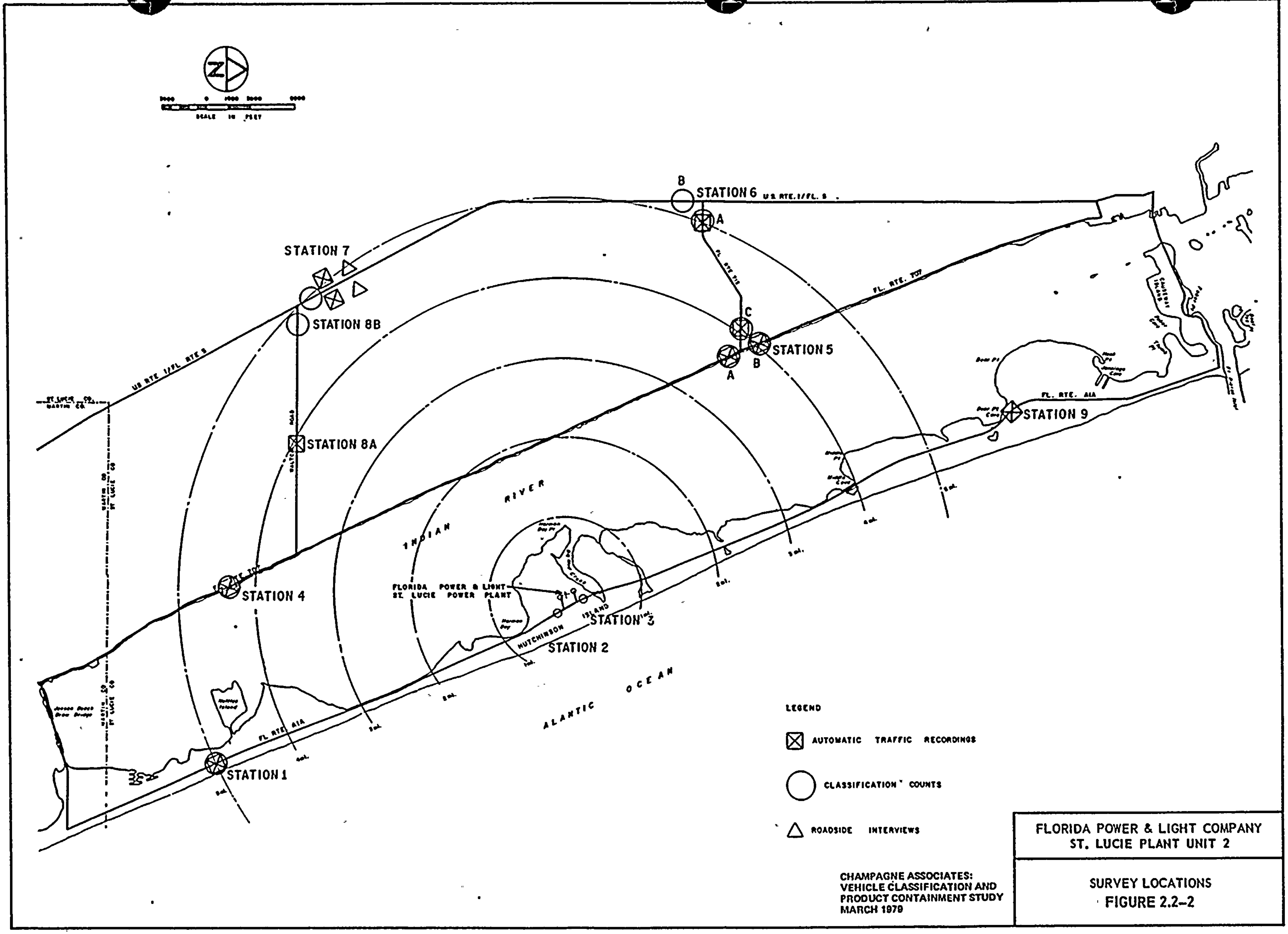
ST. LUCIE PLANT UNIT 2

NEARBY INDUSTRIAL AND TRANSPORTATION FACILITIES
FIGURE



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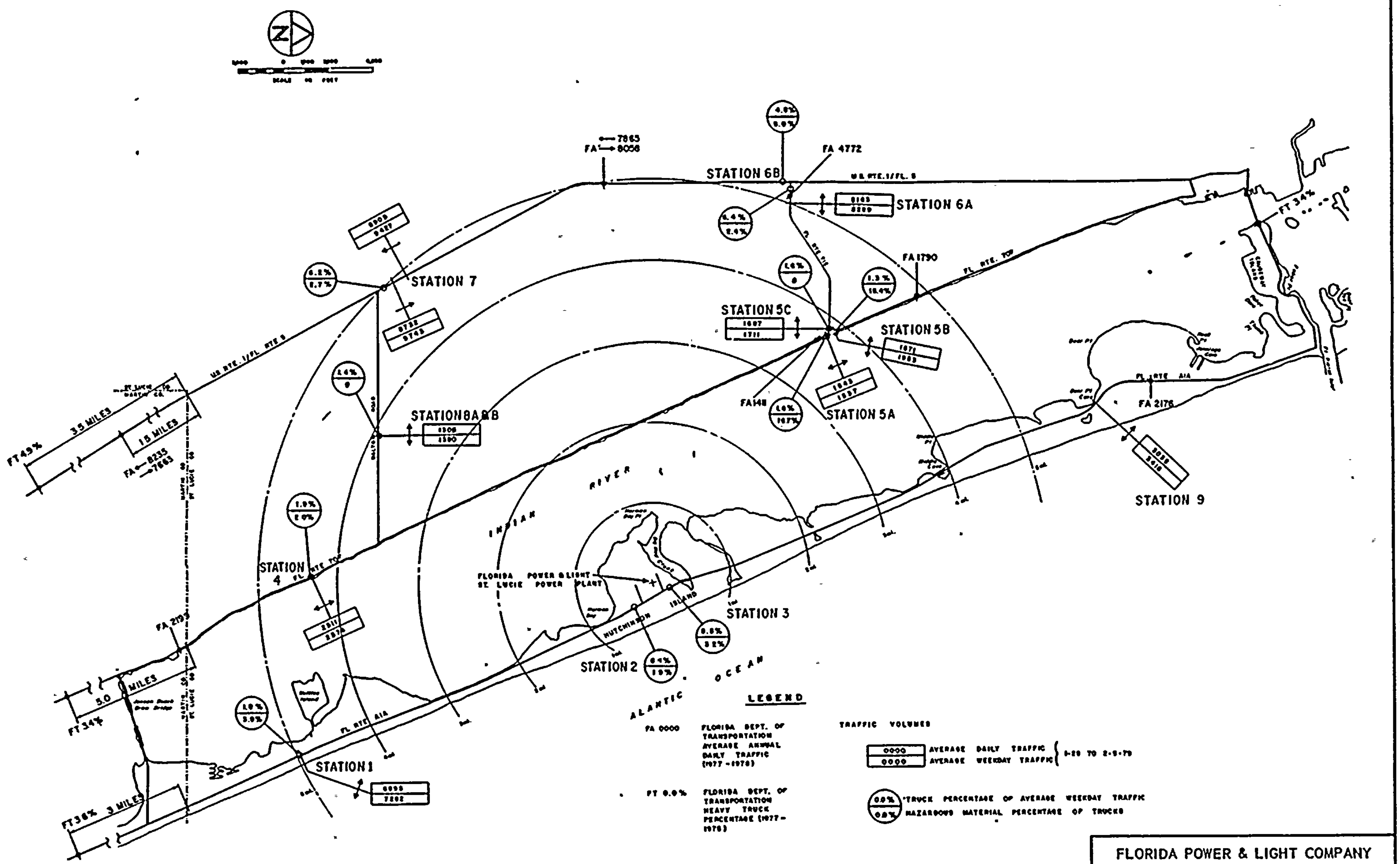
SURVEY LOCATIONS
FIGURE 2.2-2



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VEHICLE CLASSIFICATION AND PRODUCT CONTAINMENT STUDY
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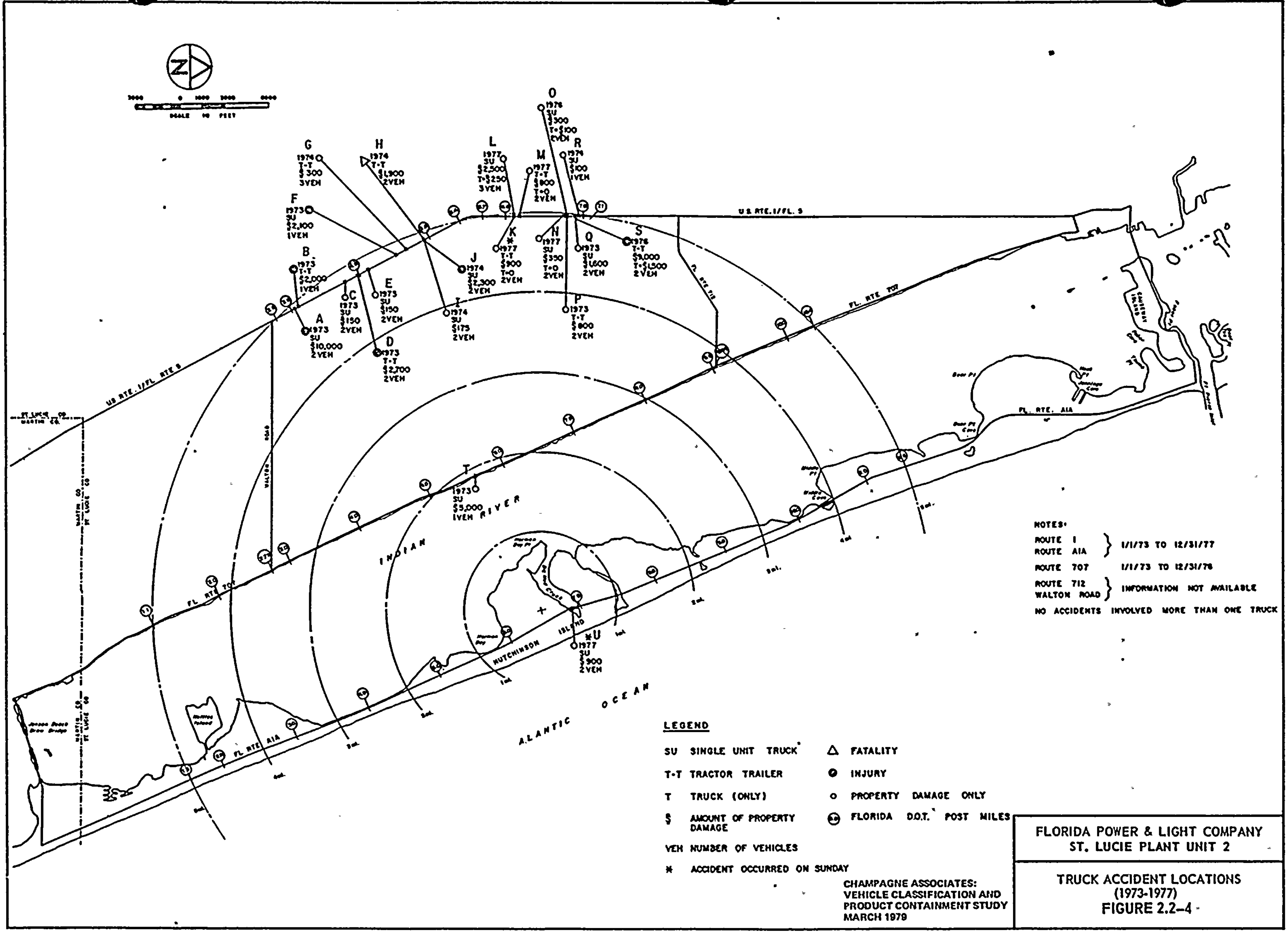
TRAFFIC VOLUMES
TRUCK PERCENTAGES
FIGURE 2.2-3



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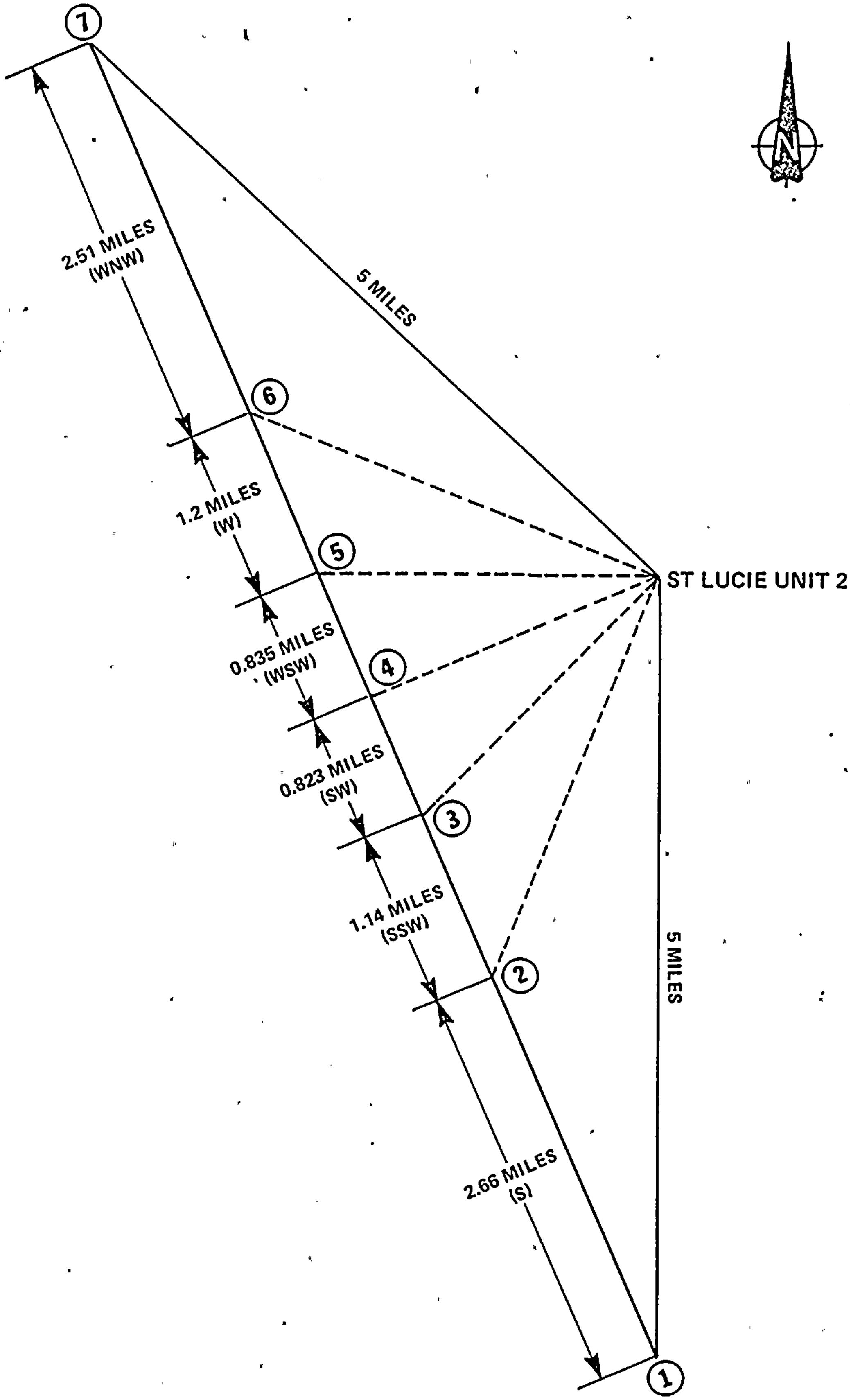
TRUCK ACCIDENT LOCATIONS
(1973-1977)
FIGURE 2.2-4



NOTE: THE WIND DIRECTION FROM THE SEGMENT OF THE RAILROAD TO THE OUTSIDE AIR INTAKES OF THE CONTROL ROOM INCLUDED IN PARANTHESIS.

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ST. LUCIE PLANT UNIT 2

SEGMENTS OF THE FLORIDA EAST COAST RAILWAY USED IN TOXIC CHEMICAL ANALYSIS
FIGURE 2.2-5