Commonwealth Edison
One First National Plaza, Chicago, Illinois
Address Reply to: Post Office Box 767
Chicago, Illinois 60690

September 18, 1981

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U.S. NUCLEAR REGULATORY COMMISSION

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: Byr[illegible] Station Units 1 and 2
Braidwood Station Units 1 and 2
Responses to FSAR Questions
NRC Docket Nos. 50-454, 50-455,
50-456 and 50-457

Dear Mr. Denton:

This is to provide advance copies of answers to questions from the NRC staff regarding the Byron/Braidwood FSAR.

Attachment A to this letter contains responses to questions on several sections of the FSAR. Some voluntary changes to various sections of the FSAR and answers provided previously are also included.

All of this material will be incorporated into the Byron/Braidwood FSAR in the next amendment. Fifteen (15) copies are provided now for your early review and approval. One (1) signed original and fifty-nine (59) copies of this letter are provided.

Please address questions regarding these matters to this office.

Very truly yours,

T. R. Tramm

T. R. Tramm
Nuclear Licensing Administrator

Attachment

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ATTACHMENT A

List of Byron/Braidwood FSAR Questions Addressed

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Also included are voluntary text changes to the FSAR

QUESTION 010.52

"Your response to Questions 010.14 and 010.34 concerning our request for a reliability analysis for the auxiliary feedwater direct diesel driven pump is not acceptable. As indicated in NUREG-75/023 Supplement 1, dated August, 1975, we require that you provide us with evidence of the reliablity of this pump to assure that its reliability is at least consistent with the reliability of the emergency diesel generators.

"It is our position that the direct diesel drive system for the auxiliary feedwater pump meet those aspects of Regulatory Guide 1.9 'Selection, Design & Qualification of Diesel-Generator Units Used As Standby (Onsite) Electric Power Systems at Nuclear Power Plants,' as are applicable to a diesel-pump unit. We recognize that Regulatory Guide 1.9 and its referenced IEEE Standards are designed for diesel-generator units but that many of its requirements can be adapted to a non-electrical output device. Clearly such requirements as starting, load acceptance, vibration, overspeed, automatic control, and site testing are applicable to a diesel-pump unit as well as a diesel-generator unit.

"Provide a comparison analysis of the reliability of similar features between the emergency diesel-generator and the auxiliary feedwater diesel driven pump. Include comparative reliabilities of the following subsystems: starting, combustion air, exhaust, flywheel, fuel oil, lubricating oil, cooling, governor, control, protection, surveillance and cubicle environment. The comparative analysis shall be based on the applicant's or other's experience with similar equipment or subsystems. Where similarities between proposed existing equipment and subsystems are poor, the applicant shall justify his reliability assessment based on the specific differences between the subsystems. Test data comparisons of existing duplicate or nearest similar diesel drive arrangements should be included."

RESPONSE

The question is addressed in the answer to Question 010.53. The auxiliary feedwater system reliability study provides data in Subsection 3.2.2.1 and in Appendix A, Sections A.I and A.II.2, of the data base that shows the reliability of the diesel driven auxiliary feedwater pump exceeds the WASH-1400 reliability for diesel generators.

QUESTION 010.53

"Provide a response to our March 10, 1980 letter concerning your auxiliary feedwater system (AFS) design. This response should include the following:

1. A detailed point-by-point review of your AFS design against Standard Review Plan Section 10.4.9 and Branch Technical Position ASB 10-1.

2. A reliability evaluation similar to that performed for operating plants (refer to Enclosure 1 of the March 10, 1980 letter) and discussed in NUREG-0611.

3. A point-by-point review of your AFS design, technical specifications and operating procedures against the generic short term and long term requirements discussed in the March 10, 1980 letter.

4. An evaluation of the design basis for the AFS flow requirements and verification that your AFS will meet these requirements (refer to Enclosure 2 of the March 10, 1980 letter).

"We note that your present AFS design provides two safety grade auxiliary feedwater pumps. We wish to point out to you that previous reliability studies for two pump auxiliary feedwater systems have indicated that installation of a third automatically started pump powered from a redundant emergency bus significantly improves AFS reliability. It is our position that you achieve a system reliability comparable to other recently approved operating Westinghouse plants with three safety grade auxiliary feedwater pumps."

RESPONSE

The following response to Question 10.53 addresses the above points. The reliability analysis incorporates a non-safety grade startup feedwater pump as part of the Auxiliary Feedwater System which increases the reliability of the Byron/Braidwood system to a level commensurate with the operating Westinghouse plants referenced in the NRC generic letter dated March 10, 1980. The response to items 1, 3, and 4 above consider only the two safety-grade pump trains installed at Byron/Braidwood stations.

Response to Question 010.53

Preface

This provides a response to the March 10, 1980 letter from D. F. Ross to All Pending Operating License Applicants of Nuclear Steam Supply Systems Designed by Westinghouse and Combustion Engineering. The response is organized into four parts, each part dealing with a request for information contained in the above letter. The ordering of the parts is modified to place the reliability analysis as Part I and the evaluation of the AFW System to Standard Review Plan 10.4.9 as Part II. The reason for this is that the reliability analysis contains a system description of the Byron/Braidwood AFS that allows the reader an understanding of the design and operation of the system and, hence, makes the discussions in the remaining parts of the response more meaningful. We advise the reader to understand this section of Part I before going on to other sections.

Parts I, II, and III respond to Enclosure I of the March 10, 1980 letter. Part IV responds to Enclosure 2 of that letter.

The Byron/Braidwood Stations use the Westinghouse Nuclear Steam Supply System (NSSS).

Response to Question Q10.53

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Part I: Reliability Evaluation of the AFW System

The Torrey Pines Technology Report GA-C16444, "Byron Units 1 and 2, Braidwood Units 1 and 2, Auxiliary Feedwater System Reliability Analysis," dated August 1981, follows.

GA-C16444

Byron Units 1 & 2
Braidwood Units 1 & 2
Auxiliary Feedwater System
Reliability Analysis

FINAL REPORT

PREPARED BY
TORREY PINES TECHNOLOGY
FOR

Commonwealth Edison

AUGUST 1981

TORREY
PINES
TECHNOLOGY
A Division of General Atomic Company

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FOREWORD

GA-C16444

This report was prepared by Torrey Pines Technology Company, a division of General Atomic Company for Commonwealth Edison Company under Purchase Order 25476 (TPT P.oject Number 2966.054). The issuing of this report completes the work assigned this project. The report authors are:

Torrey Pines Technology

W. Hannaman
D. Ligon
T. Taniguchi

Commonwealth Edison

L. Bowen
T. Weis

BYRON/BRAIDWOOD NUCLEAR GENERATING STATONS

AFS RELIABILITY ANALYSIS

GA-C16444

1.0 Introduction

1.1 Objectives

The objectives of this study were to:

- Perform a reliability analysis of the Auxiliary Feedwater Systems (AFS) of the Byron/Braidwood (B/B) plant.

- Meet the requirements of the NRC generic letter dated March 10, 1980. The NRC letter requires a reliability analysis of AFS similar to the analysis described in NUREG 0611 (Ref. 1) for other Westinghouse units.

1.2 Background

The reliability characteristics of the four Byron/Braidwood Auxiliary Feedwater Systems (AFSs) were evaluated in response to Item II.E.1.1 of NUREG-0737 and in response to the NRC generic letter of March 10, 1980. NUREG-0611 provides the basis of comparison for the Byron/Braidwood AFS reliability with Westinghouse designed operating plants. While the relative reliability of the AFSs can be ascertained from this qualitative comparison, a detailed quantitative reliability analysis of the system is required to identify the major contributors to AFS unreliability. Both qualitative and quantitative reliability results are presented in this study.

The Byron Unit 1 and 2 and the Braidwood Unit 1 and 2 AFSs are identical systems. The system evaluated in this study is the Byron Unit 1 AFS. The only difference identified between Byron Station and Braidwood Station is the reliability of the offsite power systems. The Braidwood offsite power system is considered to be more reliable for the reasons stated in section 4.1.2. Hence, to be conservative and to simplify the analysis a Byron AFS was chosen to be evaluated.

The Byron/Braidwood AFSs are redundant and diverse. The AFSs consist of three (3) trains; safety Trains A and B, and a non-safety Train C. Each train can supply 100% of the flow required for residual heat removal. Each pump has a different power supply. Train A and C pumps are electrically dependent. The Train B pump is electrically independent.

Trains A and B have two independent water supplies, the condensate storage tank and the essential service water system. The Train C water supply is the condenser hotwell. Trains A and B are automatically actuated. Train C is manually actuated. Trains A and B flows enter the steam generators (SGs) via the tempering flow lines. Train C flow enters the steam generators via either the main feedwater lines or the tempering flow lines. Trains A and B are tested on a monthly basis and following each maintenance outage. Train C is operated non-periodically on each plant start up. Hence, considerable difference exists between the trains. Two potential common cause failure areas are the steam generators, where 2 of the 4 are required for successful cooldown, and the automatic start logic controls.

The remainder of this report describes the Byron/Braidwood AFS reliability analysis assumptions, methodology and results.

1.3 Scope of Study

This study presents a qualitative comparison of the Byron/Braidwood AFS design to the operating Westinghouse designed plants using the methodology of NUREG-0611.

This study also presents a quantitative analysis of the AFS. Specifically, it includes error bounds on the results; it incorporates common cause failures and it provides a treatment of operator error.

This quantitative method of analysis consists of two approaches. The detailed analysis was performed using Reliability Block Diagrams (RBD). The uncertainty analysis was performed using fault trees (FT). The results of each approach were compared to uncover data inconsistencies, modeling differences and unreasonable assumptions.

1.4 Criteria and Assumptions

The following analytical criteria, definitions and assumptions have been made:

A. The top event for this study is taken from NUREG-0611 which states: "The time interval of interest for all transient events considered is the availability of the auxiliary feedwater system during the period of time to boil the steam generator dry."

B. The 20 to 30 minutes boil dry time assumed in NUREG-0611 is used in this study.

C. The following initiating events were used in this study as required by NUREG-0611 and are assumed to occur on one unit only:

Event A: Loss of main feedwater (LMFW) with reactor trip (LMFW/RT)

Event B: LMFW coincident with loss of offsite power (LMFW/LOOP)

Event C: LMFW coincident with loss of all AC power except for any derived from batteries (LMFW/LOAC)

D. Availability Criterion: Given that one of the postulated demand events occurs, unit AFS availability is defined as a successful systems start up (at least one Train) for the RBD model. The fault tree model considers also the system failure to start and failure to recover within the steam generator boil dry time of 20 to 30 minutes.

E. Availability of AFS Power Sources: The following conditions are met with respect to the postulated demand events and the resulting AFS success.

1. LMFW: All AC and DC power available.
2. LMFW/LOOP: Diesel Generator 1A is available for Train A
3. LMFW/LOAC: DC and battery-backed AC available for Train B

F. The failure rate data base used for quantification was taken primarily from NUREG-0611. Additional data were taken from Reference 2 and from the data analysis presented in Appendix A.

G. Degraded Failures: A partially successful performance of any active or passive component was not considered. Each component and each operator action was assumed to be either successful or failed.

H. AFS Actuation and Control: For automatic operation during emergency shutdown conditions the Engineered Safety Feature (ESF) signal is initiated by any steam generator low-low level, safety injection and/or loss of offsite power. This starts AFS Trains A and B. The AFS Trains A, B, and C can also be actuated manually.

2.0 Summary of the AFS Reliability Study

The two objectives of this study were to evaluate the reliability of the Byron/Braidwood Auxiliary Feedwater Systems for three initiating events and to meet the requirements of the NRC generic letter dated March 10, 1980. The results of these evaluations are presented as findings, recommendations, and future guidance. Findings stem from insights gained through the modeling and calculations which indicate that the Byron/Braidwood AFS are well designed from the reliability viewpoint considering the specific site conditions. Recommendations arise from areas where a simple clarification can impact AFS reliability under certain conditions. The future guidance section is intended to supply information to future studies in the probabilistic risk assessment area.

2.1 NUREG-0611 Method of Analysis (Qualitative)

The qualitative analysis of the AFS was achieved according to the qualitative criteria in NUREG-0611. The principal aim of the NUREG was to evaluate the variability of auxiliary feedwater system designs rather than evaluating the variability in data to be applied to a specific design. Details of this comparison are presented in Section 4.2. the results are summarized in Figure 2.1.

2.2 Method of Quantitative Analysis

The quantitative analysis was achieved using two methods. The first method used hand calculations based on point estimated reliability data. The second method used probability distributions in a fault tree code.

Insights into the impact of AFS reliability were first determined with event trees on a qualitative basis. Reliability Block Diagrams (RBDs) were used to study components in the AFS and provide first cut hand calculations to determine important contributors to unreliability. Next, a fault tree (FT) was developed and processed for the minimal cut sets. Then the STADIC computer code was used to calculate the probabilities and uncertainties for the fault trees under different initiating event conditions. Finally, the computer and hand calculations were compared, revealing differences in modeling approaches, data applications and the impact of variability in data on the AFS reliability. Details of these calculations are presented in Section 4.3.

The statistical independent estimate of AFS unreliability is on the order of 2×10^{-4}/yr for all the initiating events. The unreliability calculation including common cause (CC) failures is on the order of 2×10^{-3}/yr. These results are summarized in Table 2.1.

The comparison of AFS unavailability contribution for each of the initiating events is presented in Figure 2.2.

2.3 Comparison of Quantitative Methods

The two assessment methods give consistent results considering the assumption differences in the logic of both RBDs and fault trees. The two methods can be made identical for simply defined systems. However, in this analysis both methods were employed to complement each other. The RBD and FT numerical unavailability contributions were in close agreement considering the type of calculations and the differences in assumptions. The mean of the distribution was generally within a factor of 2 of the point estimate value.

For independent calculations the fault tree calculations were slightly lower because of the consideration of manual backup to the automatic equipment, adjustment of initiating event recovery for 20 to 30 minutes, and the breakdown of dominant equipment faults into various modes where the failure rates of these modes are treated probabilistically.

In the calculation of common cause factors a simplification was made for the hand calculations in which the point estimate of Beta, a factor used in common-cause failure methodology (Ref. 9 and App.F), was 0.03, whereas in the fault tree model each Beta was given its own value ranging from 0.1 to 0.001. The Beta factor contributions were not corrected for the 20 minute startup time which would lower the contribution to system unreliability. The impact of this approach was to obtain slightly higher unreliability for the common mode considerations in the fault tree modeling than in the RBD model. However, the dominant contribution to the unreliability remained the same for both methods as described in Section 4.3.

2.4 Findings

- The Byron/Braidwood three train design is assessed to have high, medium and high reliability (see Figure 2.1) based on the qualitative criteria described in NUREG 0611 for the three initiating events, (LMFW, LMFW/LOOP, and LMFW/LOAC). (Section 4.2)

- The testing capabilities and procedures exceed NUREG 0611 criteria. This allows for full flow periodic testing into the SG's at power operation and following maintenance outage. Train availability is maintained during testing. (Section 4.2)

- The diesel driven pump appears to have better start reliability than diesel generators and the overall diesel driven pump reliability is comparable to turbine driven pump reliability based on a review of the current Trojan experience. (Appendix A.I)

- The Byron/Braidwood Trains A and B are partly diverse and, therefore, resistive to some types of common mode failures. Train C is diverse from Trains A and B, but is dependent on off-site power availability. (Appendix F)

- The Byron/Braidwood off-site power systems outage frequency is significantly lower than the NUREG 0611 value of 0.2 to 0.3/yr. The outage frequency for a four line plant such as Byron are expected to be on the order of 0.02/yr from actual experience. This is a factor of 10 improvement over the average values in NUREG 0611. This in turn lowers the LOAC power probability to 5×10^{-4}/year which was used in this study. (Appendix A.III)

- Automatic switching to the essential water service system on low suction pressure for Trains A and B does not significantly improve the AFS quantitative reliability. (Section 4.3)

- The major contributors to AFS unreliability in each train were the pump startup and its controls, and Diesel-Generator 1A startup for Train A under the loss of offsite power initiating event. The major common cause contributor affecting all trains is the initiating ESF signal failure to close the steam generator blowdown valves, which results in an auxiliary feedwater bypass. Most electrical control and logic failures can be corrected by operator action within 20 minutes. Mechanical faults in the electric motors, or the diesel drive are the major contributors to unreliability beyond 20 minutes according to the experience data base currently available. Recovery from these events was included in the event tree repair models for restoring main feedwater.

  Manual backup to the automatic start signals improved the AFS reliability. The failure probability was reduced by about an order of magnitude.

  Train maintenance was also a contributor to the unreliability, but not a major contributor.

2.5 Recommendations for the AFS Operation

- Supply the Train C auxiliary feedwater pump electric power from the bus fed by off-site power from the System Auxiliary Transformer (SAT) to eliminate bus transfer unreliability.

- Consider manual instead of automatic actuations of the Essential Service Water System (ESW). Spurious operation could introduce untreated water into the steam generators. Manual operation of ESW would prevent spurious automatic actuations.

2.6 Future Guidance

- In future plant level reliability, availability, or probabilistic risk assessment studies careful consideration should be given to all diamond (undeveloped event) contributions in the fault tree model which contribute to AFS unreliability. For example, the details of the ESW system, and the steam generator blowdown system were listed as unreliability contributors, but were not modeled from drawings as the main parts of the AFS were. However, results indicate that some of these components are dominant contributors to the AFS failure.

Table 2.1

AFS Unreliability Estimates

A. Point Estimates from RBD

Freq. Per Year	Trans. Event	(20 min. recovery not included) Method	Hardware	Test	Maintenance	Human Error	Unavail Per Demand	Unavail Per Year
3	LMFW	RBDs Indep Point Est.	5.7E-5 +	2.1E-7	8.6E-6 +	2.9E-9 =	6.6E-5	2.0E-4
.02	LOSP	RBDs Indep Point Est.	8.E-4	2.8E-6	3.5E-4	3.2E-8	1.2E-3	2.4E-5
$5x10^{-4}$	LOAC	RBDs Indep Point Est.	1.7E-2	5.3E-5	6.0E-3	3.6E-7	2.3E-2	1.2E-5
								$\sum$ 2.4E-4
3	LMFW	RBD CC Point Est. β~.03	3.1E-4	1.1E-6	1.5E-5	5.1E-9	3.3E-4	9.9E-4
.02	LOOP	RBDs CC Point Est. β~.03	1.0E-3	3.7E-5	3.5E-4	3.8E-8	1.4E-3	2.8E-5
$5x10^{-4}$	LOAC	RBDs CC Point Est. β~.03	1.7E-2	6.6E-3	6.0E-3	5.8E-7	2.3E-2	1.2E-5
								$\sum$1.0E-3

Table 2.1 (cont.)

B. Mean Estimates From Fault Tree

Freq. Per Year	Trans. Event	Byron/Braidwood AFS Reliability Estimates Method		Unavail Per Demand*	Unavail Per Year
FAULT TREE EVALUATION					
3	LMFW	FT Indep	(includes allowance for 20 min startup)	$2x10^{-5}$	$6x10^{-5}$
.02	LOOP	FT Indep.		$1x10^{-3}$	$3x10^{-5}$
$5x10^{-4}$	LOAC	FT Indep.		$7x10^{-2}$	$4.3x10^{-5}$
					$\sum 1.3x10^{-4}$
3	LMFW	FT CC .2> β >.001	(Beta factors not corrected for 20 min startup)	$7x10^{-4}$	$2.4x10^{-3}$
.02	LOOP	FT CC .2> β >.001		$2x10^{-3}$	$5.2x10^{-5}$
$5x10^{-4}$	LOAC	FT CC .2> β >.001		$7x10^{-2}$	$4.3x10^{-5}$
					$\sum 2.5x10^{-3}$

*Hardware and human error considered explicitly for cases of omission. Human errors in maintenance or testing considered implicitly within the factor approach.

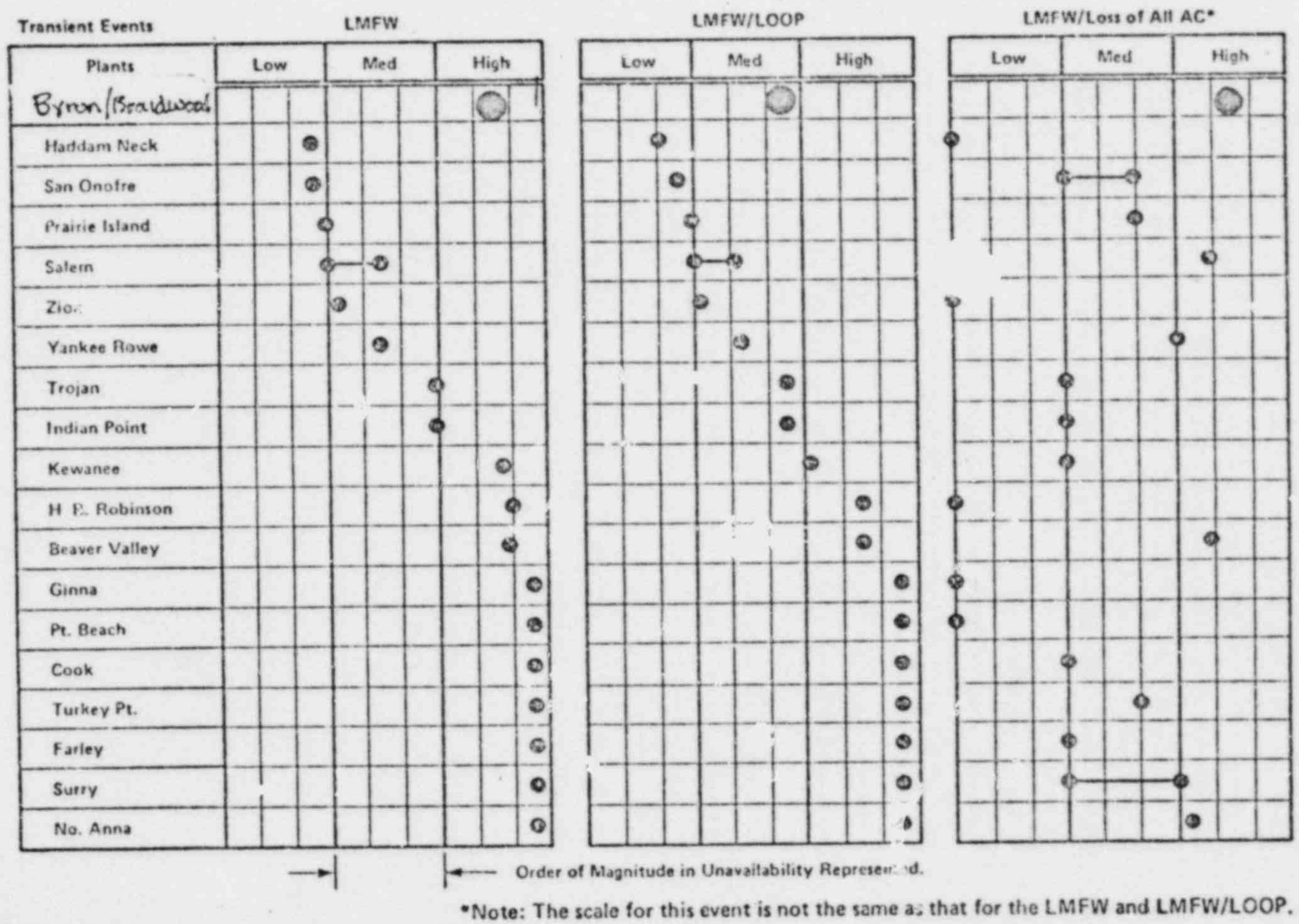

Figure 2.1 Comparison of This Byron/Braidwood Qualitative Reliability Assessment with Other Westinghouse AFS Using NUREG 0611

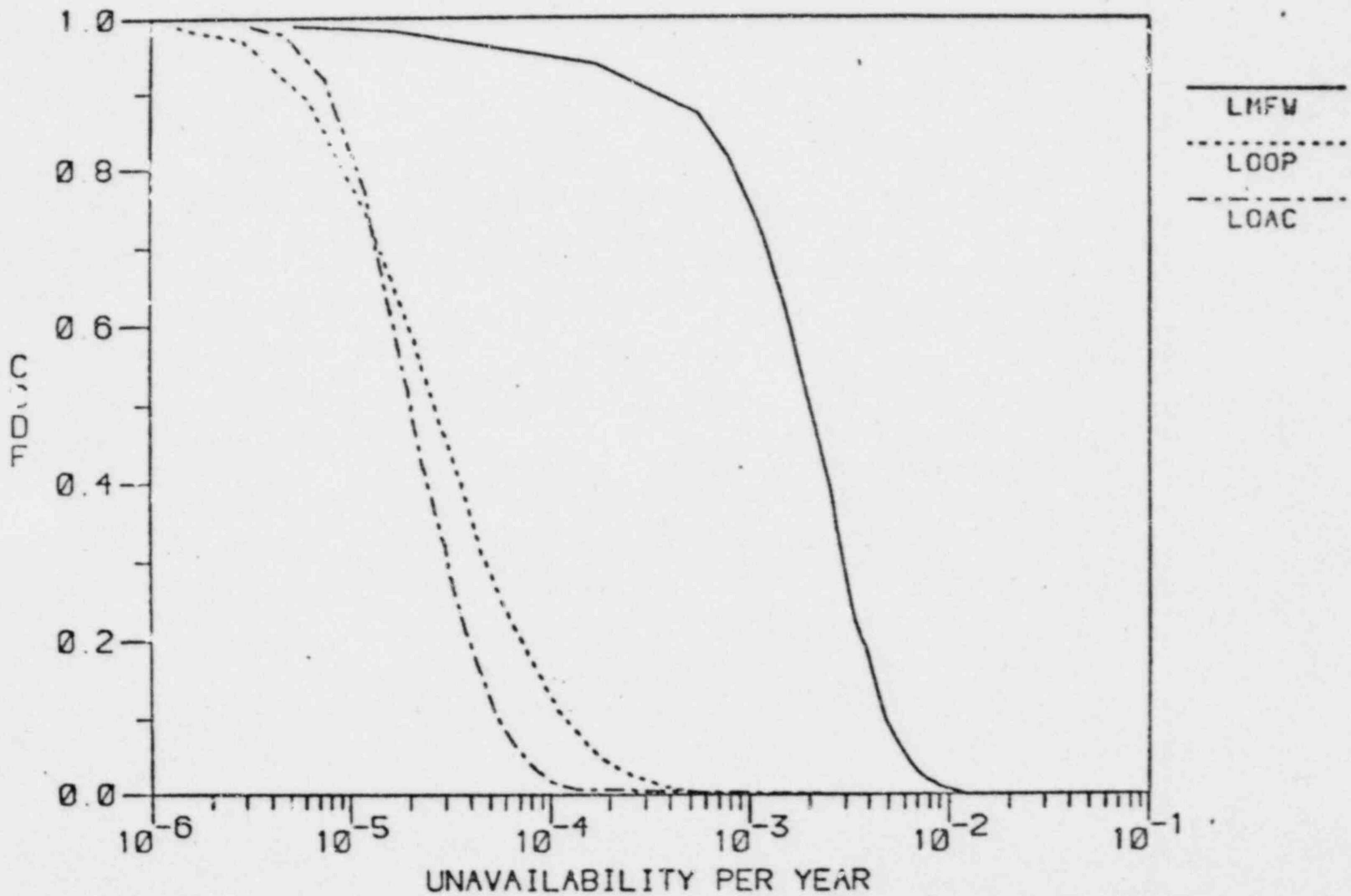

Figure 2.2 Comparison of AFS Unavailability Contribution from LMFW, LMFW/LOOP, LMFW/LOAC at the B/B Plant

3.0 Auxiliary Feedwater System Description

3.1 General AFS Information

The function of the Auxiliary Feedwater System is to provide residual heat removal when the main feedwater (MFW) system is unavailable. The AFS consists of three 100% trains. Each train has the capacity to supply the steam generators with sufficient feedwater to cool down the unit safely to 350°F, the temperature at which the low pressure residual heat removal system can be utilized. One of the trains is used during start up of the unit. A simplified drawing of the Bryon/Braidwood AFS is shown in Fig. 3.1.

Auxiliary feedwater is supplied by diverse means with two automatically initiated safety trains, Trains A and B, and one manually initiated non-safety train, Train C. Train A utilizes an ESF seismic Category I electric-motor driven pump powered from ESF Bus No. 141. On loss of offsite power this bus is supplied by Diesel Generator 1A. Train B utilizes an ESF seismic Category I diesel-engine driven pump. This diesel-engine pump is AC power independent. Trains A and B are located in the Auxiliary Building.

Train C utilizes the start up feedpump, a non-ESF electric-motor driven pump. This train is also used during unit startup and normal unit shutdown from less than 10% power. It is also used in the hot standby mode. Train C is located in the Turbine Building.

3.2 System Operation

Successful unit cooldown can be achieved by supplying feedwater to any two of the four available steam generators. Safety analysis has shown that 160 gpm delivered to each of three steam generators or 240 gpm delivered to each of two steam generators is sufficient for residual heat removal. Each of the ESF trains, Trains A or B, has two times the minimum capacity. The non-ESF train, Train C, has approximately four times minimum capacity. Hence, any one of the three trains can supply the needed cooling water to the secondary side of the steam generators.

The Train A pump motor drive is powered from ESF Bus 141. the Train A AFS regulating valves are powered by 125 V DC ESF Bus 111. The Train A AFS isolation valves are powered from 480 V AC ESF Bus 131. The AFS isolation valves are normally open and are not required to change position on AFS actuation.

The Train B pump diesel engine is supplied by its own 24 volt DC batteries. This diesel drive is self-contained and completely independent of AC power under emergency conditions. Train B AFS regulating valves are powered by 125 V DC ESF Bus 112 Train B AFS isolation valves are powered by 480 V AC ESF Bus 132. The AFS isolation valves are normally open and are not required to change position on AFS actuation.

The Train C pump motor is powered from a non-ESF bus. This train is unavailable on loss of offsite power.

The ESF system will automatically start Trains A and B on three signals: lo-lo level in the secondary side of the steam generator; a safety injection signal; and, a loss of offsite power signal. Manual start up capability from the control room backs up the automatic system start. Instrumentation and controls are also provided at the remote shutdown panel in the unlikely event that the control room must be evacuated. Train C is manually operated from the control room and provides a manual back up to Trains A and B.

The normal water supply for Trains A and B is the condensate storage tank. The alternate supply is the essential service water system. Train C takes a suction on the condenser hotwell via the condensate/condensate-booster pump.

The Train A and B pumps are protected against low suction pressure with a run inhibit signal. This condition is corrected by the automatic opening of essential service water (ESW) valves to the suction of the auxiliary feed pump. This occurs when a low suction pressure signal is received.[1] The automatic startup can be initiated within about 1 minute from the initiation of the Lo-Lo steam generator level signal. This is well within the limit of 30 minutes set by the inventory of secondary water already in the steam generators.

To prevent pump damage a recirculation system is provided on each AFS pump discharge which bypasses flow to the condensate storage tank or essential service water system for the ESF pumps and to the condenser hotwell for the non-ESF pump. Excessive recirculation flow is prevented by orifices in the piping, hence each of the pumps can supply the full feedwater requirement with full recirculation flow.

Monthly periodic tests are required for Trains A and B. Only one train is tested at a time, the other train remains in a normal line up. The air-operated discharge test valve is closed and the pump is warmed up on recirculation flow. An ESF logic start signal will automatically open the discharge valve. Hence, the train under test is always available to supply auxiliary feedwater to the steam generators. The conclusion of the test requires the train valves to be aligned to the steam generator for a full flow test. This portion of the periodic test will identify any plugged or inadvertently closed valves. The same test is required following each maintenance or repair activity.

Train C operation is tested by normal use during startup of the unit. It is always available except following loss of offsite power.

From the reliability viewpoint, the key components which contribute to the AFS unreliabilities are the electric and diesel drives and their controls, and start logic. Valves in the water pathway have been contributors to system failure in other AFS but the valve failure modes, the application of check valves, and the monthly full flow tests reduce their contributions. A failure of the condensate water supply is not a major contributor because of the partly diverse valve train from the tank and the backup water supply system. Blowdown valves in the steam generator are

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1. This action provides unprocessed water to the main steam generators. Section 4.3 shows that this action does not significantly improve AFS reliability, for the initiating events considered.

potentially dominant contributors of AFS failure. Maintenance outage of a train is also a contributor to AFS unreliability.

3.2.1 Water Pathway

3.2.1.1 Trains A and B

The normal water supply is the 500,000 gallon condensate storage tank. A minimum capacity of 200,000 gal is reserved for the Trains A and B. In addition to this normal water supply and the essential service water system, the condensate tank from the other unit can be manually valved in by changing the position of one valve.

The auxiliary feedwater pump discharge is routed to the steam generators via the tempering flow lines. From the steam generator the steam is normally discharged to the condenser through the steam dump system. Should the condenser be unavailable, the steam is vented to atmosphere through the secondary relief valves. Either path provides for successful operation of the AFS, since the AFS function is to remove heat from the steam generators until either a restart condition or the residual heat removal system condition is reached.

The auxiliary feedwater flow to the steam generators is manually controlled from the control room. The control valves are throttled as the desired steam generator level is reached and the decay heat load diminishes during a cool down cycle. Depending on which trains, pumps, and steam generators are available, the operator can line up the appropriate valves to establish AFS flow paths. Any train can feed any of the four steam generators. In the event that the control room must be evacuated the AFS valves and pumps can be operated at the remote shutdown panel.

When the AFS has cooled the plant down to 350°F, about 5 hours after the initiating event, the residual heat removal system can manually be placed in operation and the AFS trains are manually placed in standby condition.

3.2.1.2 Train C

The Train C water supply is the condenser hot well. The normal volume is approximately 100,000 gallons. Four 33% condensate/condensate booster pumps are available. Each condensate/condensate-booster pump is driven by one motor. With off-site power available, the running pumps will remain operating and recirculating to the condenser, thus providing NPSH (net positive suction head) to the Train C pump. This pump is manually controlled from the control room. An ESF logic start of Trains A and B will automatically close the main feedwater valves used by Train C. To start Train C, the steam generator (SG) feedwater valves must be manually reset with pushbuttons, one for Logic Train A and one for Logic Train B.

3.2.2 Auxiliary Feedwater Drives

3.2.2.1 Diesel Driven Auxiliary Feedwater Pump

This 1250 HP diesel-pump combination is designed to be independent of AC power under emergency conditions. It is automatically started by a self

contained 24 Volt D.C., battery-powered, start system. When any of three emergency signals (e.g. the Lo-Lo steam generator level, safety injection, or loss of AC power) are received the diesel is started. The pump provides 840 gpm at a 3350 feet head. This Detroit Diesel has its own 500 gallon supply of fuel, intake and exhaust air ducts, internal lube oil pump, and water jacket cooling pump. An axial vane fan driven from the gear box circulates air over the engine to provide cooling during operation. The heated air is exhausted through passive building vents. When AC power is available, backup pumps are available for oil pressure, water jacket cooling and room air cooling. These backup systems help reduce engine wear during testing periods by providing prestartup oil pressure to the bearings and providing backup engine cooling.

Experience with diesel driven auxiliary feedwater pumps has been gained in the Trojan power plant. Operational data from that plant provides insights into the type of failure modes experienced and an estimate of the frequency of their occurrence. These data, described in Appendix A, suggest that the failure to start on demand is $1x10^{-2}$/D whereas diesel generator failure to start on demand in WASH-1400 is $3x10^{-2}$/D. In about 1900 H of operation five failures have occurred after the early break-in faults were corrected. This yields a failure rate of about $3x10^{-3}$/H which is equivalent to the WASH-1400 estimate. Hence, this first-of-a-kind diesel driven auxiliary feedwater pump has achieved a reliability somewhat better than the Class 1E diesel generators found in nuclear power plants.

Review of the B/B diesel design specification indicates that two of the early problems with the Trojan system, overspeed trip and poor combustion on startup, have been resolved. The excessive trips on engine overspeed have been corrected by utilizing an independent supply of hydraulic fluid for the governor. This was a problem in early operation at Trojan when erratic governor operation caused several overspeed trips. The second problem, that of cold engine temperature, has also been corrected with the addition of a water jacket heater to maintain the coolant temperature as specified in the ASME code Section III, Class 3. This will improve fuel combustion on startup by maintaining temperature in the 100°F range.

3.2.2.2 ESF Motor Driven Auxiliary Feedwater Pump

The electric motor driven pump is used in most Pressurized Water Reactors (PWRs) as a diverse method of supplying auxiliary feedwater. In the B/B design the Train A pump is powered from the 4160 volt ESF Bus No. 141. This horizontal pump rated at 1250 HP provides 890 gpm at 3350 feet of head. This pump is functionally redundant to the diesel driven pump under all conditions except the loss of all AC power.

Experience with these pumps indicated that the failure to start probability, given that electrical power is available, is about an order of magnitude less than for the diesel driven pump. WASH-1400 uses 10^{-3}/D and 10^{-5}/H for failures to start and to run.

3.2.2.3 Non-ESF Motor Driven Auxiliary Feedwater Pump

The non-ESF electric motor driven pump is rated at 2000 HP and can provide 5300 gpm at 1200 feet of head. It is powered from a non-ESF 4160 V bus.

This electric motor driven pump has two requirements. Off-site power and at least one of the four condensate/condensate-booster pumps running to provide auxiliary feedwater to the four steam generators. The condensate/condensate-booster pumps also require off-site power. Commonwealth Edison's practice is to operate with 50% of the house load on the unit auxiliary transformers (UATs) and 50% on the system auxiliary transformer (SATs) (FSAR pg. 8.3-2). On a turbine-generator trip, two condensate/condensate-booster pumps will continue to operate on the SATs. The other condensate/condensate-booster pumps will continue to run while the bus breakers automatically transfer from UAT to SAT.

3.2.3 Train A and B Valves

3.2.3.1 Condensate Storage Tank Valves

The ESF AFW pumps are supplied from the Condensate Storage Tank via two separate lines, both of which are normally aligned to supply the ESF AFW system.

One line has a manual, locked open valve and a check valve in series. The other line has two manual, normally open valves in series. The lines combine downstream of these valves in the Turbine Building and this common line splits to supply the two ESF AFW pumps in the auxiliary building through a manual, normally open valve and a check valve in series in each pump supply line.

3.2.3.2 Auxiliary Feedwater Pump Suction Valves (Manual)

The manual suction valves to the Train A and B are normally open. In the case of low suction pressure the check valves in the suction line prevent backflow from the essential service water system to the condensate tank. The positioning of these valves are verified by the full flow testing procedure. Plugging or manually closing these valves are the most likely cause of loss of normal suction from the condensate storage tank.

3.2.3.3 Auxiliary Feedwater Supply Valves

Each train supplies all four steam generators. The Train A and Train B supply lines combine into a common header prior to entering the tempering flow lines at each steam generator. Each supply line has three valves; a check valve; a motor operated isolation valve; and a flow control valve. The control valves are air operated valves controlled from the control room or from the remote shutdown panel. The valves fail open on loss of air or loss of power to the solenoids.

3.2.3.4 Auxiliary Feedwater Backup Water Supply (Automatic)

The essential service water back up supply valves are normally closed, motor-operated valves. There are two valves in series to each pump suction. these valves are powered by ESF buses. A low pump suction pressure signal developed independently by each train in conjunction with a lo-lo SG level signal will automatically open these valves. They can also be opened from the control room or manually at the valve.

3.2.4 Train C Valves

The Train C flow path utilizes the normal main feedwater lineup. All the valves are to remain in operating position following a loss of main feedwater condition except for the main feedwater flow control valves. These valves must be reset and manually positioned for auxiliary feedwater flow from the Train C pump.

3.3 Inspection and Testing Requirements

The AFS trains are capable of being tested while the plant is in normal operation. A full flow test through the AFS valves allows the valve positions to be operationally tested. Discharge pressures and flow indications are provided locally and in the control room. Periodic testing will identify any "plugged" valve failures. During the first phase of the test procedure, the discharge test valves are closed and the auxiliary feedwater is recirculated back to the condensate storage tank. After the pump is tested, the discharge valves will be opened to allow full flow into the steam generators. These valves are designed to open on an ESF start signal for the AFS. Thus, the train is available during the test.

3.4 Instrumentation and Control

Control room instrumentation includes steam generator level indications, controls, hand switches, and position indicators for power operated valves.

The control start logic for the AFS, which is part of the Engineered Safety Features Actuation System, is an automatic two-of-four input signal with manual override.

The following main control room monitors are provided for purposes of AFS control:

- AFS trip status light.
- Discharge presssure of each AFS pump.
- Auxiliary feedwater flow to each steam generator.
- Status lights for each regulator valve.
- Alarms for AFS diesel engine temperature, oil pressure, and speed.
- Status lights for AFS power operated valves.

The instrumentation and control system is designed such that undervoltage on two of the four instrument channels results in automatic initiation of the auxiliary feedwater Trains A and B.

3.5 Supporting Systems and Sources

The active components of the AFS are dependent upon diverse sources of electrical power. Lube oil and cooling subsystems are supplied internally from the diesel engine. All valves and controls in the same train are similarly

matched to the same power source as its pump, and key devices can be manually or locally actuated as well. Four independent transmission lines supply the offsite power, and two dedicated diesel generators back up the onsite Class 1E power busses. Up to 300,000 gallons of demineralized water can be made available to the AFS from the Unit 2 condensate storage tank by a manual cross tie valve.

3.6 Technical Specification Limitations

Technical Specifications require the availability of 200,000 gallons of water in the condensate storage tank for AFS use. Tank levels are alarmed and annunciated in the main control room.

A maximum of 7 days out of service is allowed for maintenance or repair of an ESF train while the reactor is critical. If that time is exceeded, the reactor must be put in hot shutdown within the next 12 hours.

Surveillance Requirements

1) Each auxiliary feedwater pump shall be demonstrated operable:

   A. At least once per 31 days by:

      (1) Verifying that each pump develops discharge pressure of at least 90% of the manufacturer pump performance curves.

      (2) Verifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in correct position.

   B. At least once per 18 months during shutdown by:

      (1) Verifying that AFS starts automatically upon receipt of an ESF test signal.

2) The condensate storage tank shall be demonstrated operable at least once per 12 hours by verifying that the contained water volume is within its limits when the tank is the supply source for the auxiliary feedwater pumps.

3) The essential service water system shall be demonstrated to be available whenever the condensate storage tank is inoperable.

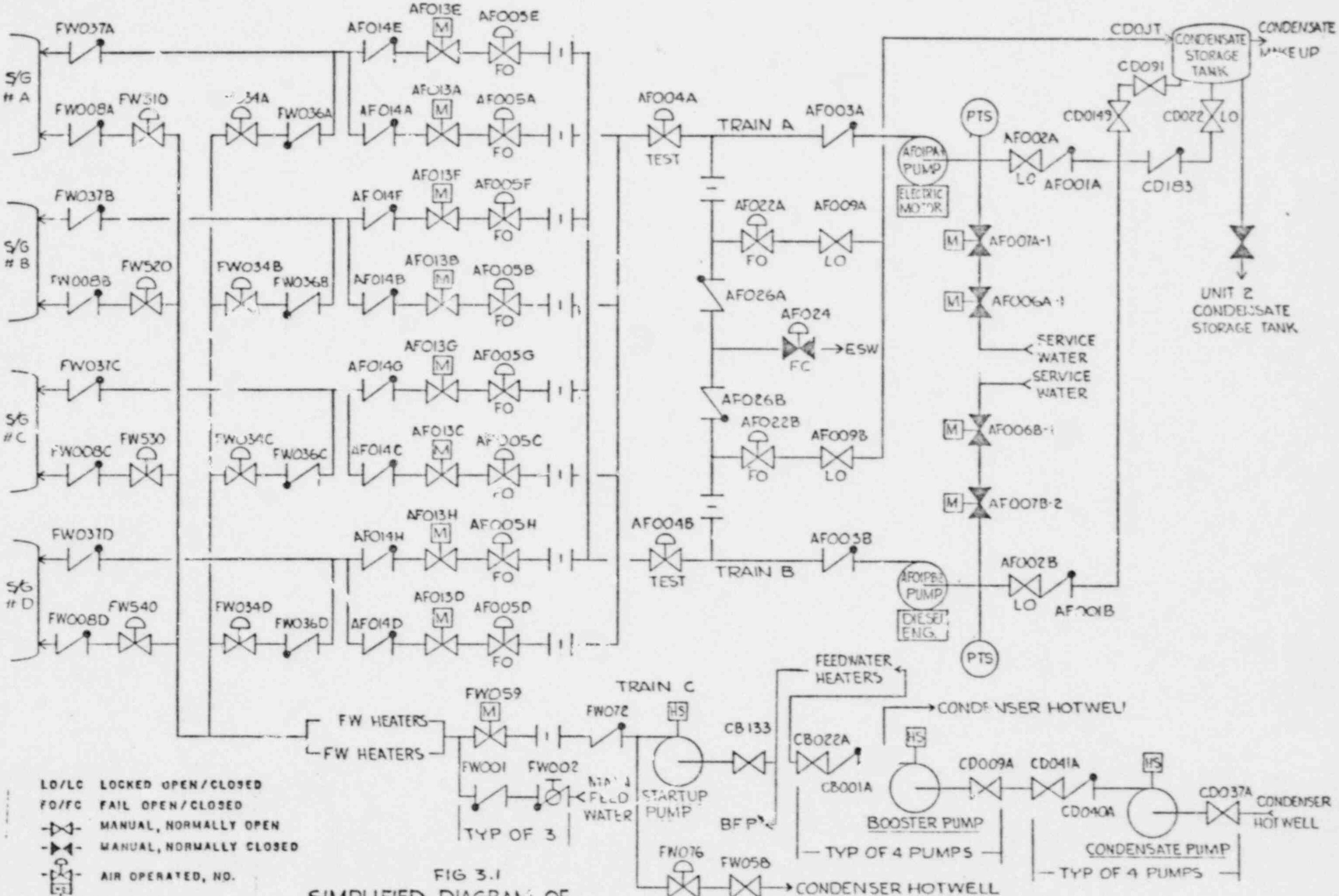

FIG 3.1
SIMPLIFIED DIAGRAM OF AFS FOR BYRON/BRAIDWOOD UNITS

3-8

4.0 Reliability Analysis

One of the most important parts of any reliability analysis is to define very carefully the boundaries of the analyzed system while retaining a perspective on the entire plant operation. In this analysis two methods were used. First, the components of the auxiliary feedwater system were defined in a reliability block diagram. This is a logic model which is based on the components needed to make the system work. The second approach is that of the event tree/fault tree analysis method. After studying the system to understand fully its operation, event trees were constructed to show the potential for various event sequences which involve the auxiliary feedwater system. The results of the event tree construction help to define the top events of fault trees. This important definition becomes the top block in a fault tree. Hence, by this process the failure conditions and component failures which lead to the top event can be defined. In the fault tree format the components can be in any system. RBDs and fault trees can be logically equivalent, if the boundary conditions are equivalent.

4.1 Event Tree Construction

Three initiating events are considered in the reliability evaluation as suggested in NUREG-0611. Accident scenarios stemming from these events are expected to dominate the plant risk for events using the AFS as shown in WASH-1400. These events are loss of the main feedwater (LMFW), loss of offsite power (LMFW/LOOP) and loss of all AC power (LMFW/LOAC). The AFS reliability has an impact on which scenario can be followed after the initiating event, but the purpose of this study is to assess only the AFS reliability under the major events listed above.

4.1.1 LMFW

Loss of main feedwater events are characterized by a reduction in steam generator water levels which results in a reactor trip, a turbine trip, and auxiliary feedwater actuation by the ESF protection system logic. Success of these actions are considered in events 1 and 2 in Figure 4.1 on the upper branches. Following reactor trip from a high initial power level, the power quickly falls to decay heat levels on the order of 6% to 3% of full power. Without auxiliary feedwater as shown in the lower branches of event 3, the steam generator water levels continue to decrease, progressively uncovering the steam generator tubes as decay heat is transferred and discharged in the form of steam preferably through the steam dump valves to the condenser or through the steam generator safety or power-operated relief valves to the atmosphere. As a result, the reactor coolant temperature increases as the residual heat in excess of that dissipated through the steam generators is absorbed. If this condition continues, the volume of reactor coolant expands and begins filling the pressurizer. Without the addition of sufficient auxiliary feedwater in about 30 minutes, further expansion will result in water being discharged through the pressurizer safety and/or relief valves into the containment and this is considered to be a failure of both the AFS and MFW. These sequences are shown with solid lines in Event 6.

If the temperature rise and the resulting volumetric expansion of the primary coolant are permitted to continue, and if the relief valves fail to reclose as shown in the lower branches of Event 7, the continuing loss of fluid from the primary coolant system may result in bulk boiling in the Reactor Coolant System and eventually in core uncovering, loss of natural circulation, and core damage. This condition can be avoided by recovery of the main feedwater or auxiliary feedwater within approximately one hour or successful closure of the relief valves. If such a situation were not recovered, the Emergency Core Cooling System could be used to supply primary coolant makeup water. After a longer time period, however, the primary coolant system pressure may exceed the shutoff head of the safety injection pumps, causing an insufficient supply of water. Hence, both success and failure of the ECCS is considered in Event 8. The timely introduction of the sufficient auxiliary feedwater is necessary to counteract the decrease in the steam generator water levels, which will reverse the rise in reactor coolant temperature, and prevent the pressurizer from filling to a solid water condition. AFS success is to establish stable hot standby conditions or prepare for restart. Subsequently, a decision may be made to proceed with plant cooldown if the problem cannot be satisfactorily corrected. The event tree of Figure 4.1 then provides the details of the potential accident sequences for later quantification. This evaluation is limited to reliability of the AFS in Event 3 and not the entire accident sequence.

4.1.2 Loss of the Offsite Power (LMFW/LOOP)

Loss of offsite power is an ESF actuation signal for the AFS. The reliability of the AFS is affected under this initiating event due to the unavailability of Train C (the non-ESF AFS train). The event tree of Figure 4.4 is a modification of Figure 4.1 which includes detailed consideration of the loss of offsite power. The difference here is in the recovery of the power supplies. The physical behavior of the plant is as described in Section 4.1.

As can be seen from Figures 4.2 and 4.3 there is a difference between the offsite power sources at the two stations. At Byron the offsite power system consists of a switchyard that is supplied from the Commonwealth Edison grid by four separate 345 kV lines. At Braidwood the system is supplied by six separate 345 kV lines. It is assumed that Braidwood Station will have a higher offsite power reliability than Byron Station. Hence conservatively the reliability data used in this study is based on the Byron system.

4.1.3 Loss of All AC Power (LMFW/LOAC)

This event is a subset of the event tree in Figure 4.4. The main affect of this event is that the diesel generator power supply fails to supply the motor driven feedwater train. If the diesel generator supplying bus 141 cannot be recovered quickly, then the AFS depends on the diesel driven pump in Train B as the only operable source of auxiliary feedwater. Hence, for this initiating event diesel-driven pump reliability is the key factor. AFS response in this case, requires that the entire Train B cooling system be independent of AC power sources. The event tree of Figure 4.4 is still applicable except that loss of all AC power sources is considered in the fault trees.

4.2 Qualitative Reliability Analysis

A qualitative reliability comparison of the Bryon/Braidwood AFS with the results of those analyzed in NUREG 0611 indicates that the B/B system is among the higher reliability plants for LMFW and LMFW/LOAC, but in the mid range for LMFW/LOOP because of only two train redundancy. However, given the reliability of the offsite power, this effect is not significant in the overall assessment.

4.2.1 NUREG 0611 Comparative Reliability Analysis

The NUREG 0611 (Ref. 1) provides to be a qualitative reliability assessment of the various operating PWR AFSs. Qualitative criteria were used to classify the system reliability in classes of relative low, medium and high reliabilities for the three initiating events (IE). The criteria were presented in a narrative form on pages III-21 through III-23. These criteria are summarized here in Table 4.1. Table 4.2 shows the NUREG 0611 criteria and the Byron/Braidwood classifications and assessments. Figure 4.5 presents the comparison of the Byron/Braidwood AFS reliability characterization with other operating plants using a Westinghouse NSSS.

4.2.1.1 Interdependencies

The potential for dependencies between trains was reviewed during the reliability analysis from the qualitative viewpoint. Separation was adequate. No single point failures were found. Even in the test procedures an automatic start signal overrides the testing thus eliminating a potential human dependency. Inadvertent closure of the pump suction valve will not incapacitate the train.

The operating experience in Appendix A indicated that the greatest potential for dependencies between the Trains was the automatic start signal. However, in every case, the manual override start was successful.

4.3 Quantitative Reliability Evaluation

The purpose of this analysis is to identify the dominant contributors to AFS unreliability. Methods employed here incorporate experience from operating plants. The basic data is taken from WASH-1400, as updated in NUREG 0611, and supplemented where needed from specific power plant operating reports. This data synthesis can help to simplify the fault trees and reliability block diagrams by grouping components into "supercomponents" which are represented by single inputs into the uncertainty evaluation code (STADIC). The methods can also be used to determine the impact of design options.

Two logic model types are utilized, each having strong points and weaknesses. First, the reliability block diagrams (RBD) logic model developed from questions such as: What is needed to make the system operate? What backups exist? What redundancies exist? This modeling technique requires knowledge of valve positions, changes of state, signal operations, etc., to properly model the system reliability characteristics. The weakness of the RBD logic modeling is that outside system interdependencies could be overlooked. For example, valve positions in other systems which impact flow in the AFS could be missed. Fault trees (FT) logic models are constructed to calculate

the probability of the "top event". The top event should be carefully defined using event trees to define the boundaries of the fault tree analysis. For example, in the event tree of Figure 4.4 there is no special provision for onsite power (i.e. Diesels 1A,1B). Therefore, these items should be included in the fault tree. A drawback to fault trees is that so many conditions can be described that the reduction of the FT to its important contributors can be very difficult. To simplify this step, a minimal cut set computer code was employed to identify each set of failure conditions within the FT. These minimal cut sets helped in assessing the common mode failure potential between required components. After the minimal cut sets were developed, FT equations were written to describe the probabilities and associated uncertainties of the AFS unavailability. A computer code called STADIC can accept this FT equation along with data inputs for each component or supercomponent. Based upon this input a probability distribution for the top event was formulated.

4.3.1 Reliability Block Diagram Analysis

4.3.1.1 Assumptions

The RBD (Appendix B) for the Bryon/Braidwood AFS was developed assuming the need for the AFS from plant near full power. The RBD delineates the various success paths for starting the system on demand. It also shows the various components that were considered in the analysis and how they are inter-related to each other. The various types of redundancies (active, automatic standby, remote manual and local manual standbys) are symbolized.

The top train on RBD pages 1, 2, 3, and 4 shows the model for Train A with the ESF electric motor drive pump and the bottom train shows the model for Train B with the ESF diesel drive pump. These trains are identical except for the pump drives. The inter-relations between these trains are the common source of water supply, the condensate storage tank. However, each train has an independent, automatic switching capability to the ESW (emergency service water) system when low pressure occurs on the pump suction coincident with a low-low SG level trip. The ESW is operable from the ESF electrical bus.

For the initiating events considered in this analysis, an automatic ESF AFS start signal was assumed. The ESF logic A signal actuates Train A and ESF logic B signal actuates Train B. Manual over-ride capabilities for each train are available in the control room and the remote shutdown room.

AFW pump recirculation is only required when the flow is throttled, or stopped, to the SGs. During initial demand of the AFS when full flow is required to the SGs, the recirculation is not required. Each train has an automatic recirculation switchover from the condensate storage tank to the ESW system. The recirculation is used during pump testing for pump warmup before reopening the pump discharge test valve to provide full flow test into the SGs. By procedure, this same test will be used after each maintenance action on the train to assure complete train functioning.

Train A and Train B can independently supply AFW to each SG through the flow limiting orifice, flow control valve and a containment isolation valve. Cross flow between these trains is prevented by check valves in the tempering flow line at the SG and on the SG blowdown line and valves.

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Train A and Train B can independently supply AFW to each SG through the flow limiting orifice, flow control valve and a containment isolation valve. Cross flow between these trains is prevented by check valves in the tempering flow line at the SG and on the SG blowdown line and valves.

Train A is available for two of the three postulated initiating events (IE); loss of main feedwater (LMFW) with offsite power, and loss of main feedwater with loss of off-site power (LMFW/LOOP) (onsite power, only).

Train B is available for all IEs, namely: LMFW, LMFW/LOOP and LMFW/LOAC (DC battery power, only).

Train C (10% full plant capacity) utilizes the normal condensate system and provides AFW through the main FW piping to the SGs. Therefore, Train C is available only for LMFW and requires remote manual startup from the control room. Train C utilizes the condensate hot well as the water source.

Assuming the plant is at power, three of the four (1/3 capacity) condensate/condensate booster pumps (one motor drives both pumps) will normally be operating. The fourth pump is on automatic standby. Two of the condensate/condensate booster pumps motors will be on SAT and two will be on UAT. With reactor/turbine trip, the UAT bus breakers will rapidly transfer (within 3 cycles) to the SAT. This will not affect the motor operation on these buses unless breaker transfer is not successful. The operating condensate/condensate booster pumps were assumed to continue to operate after a reactor/turbine trip. This requires that the condensate booster recirculation functions successfully. Otherwise, these pumps will be pumping against a shutoff head. In addition to the condensate booster pump recirculation, the two normally operating 50% capacity main FW pump recirculation valves are operable for added condensate recirculation. The standby 50% capacity main FW recirculation requires remote manual action before it can be placed into recirculation operation.

The normal condensate to the FW pump suction piping requires passing through several low pressure feedwater heaters. At least two flow paths are available, with a remote manual bypass line available around all LP FW heaters.

The startup FW pump is started from the control room. This pump motor will be on a bus fed from the SAT and will be available after a reactor/turbine trip and the LMFW initiating event.

Startup FW pump recirculation was assumed to normally operate. With no flow on its discharge line, the recirculation valve will be fully open and allow recirculation flow. Thus, with condensate/condensate booster pump operating and recirculating, this pump will also be recirculating.

The startup FW pump is discharged directly into the main FW discharge piping. The main FW pump discharge check valve or stop-check valve must close in order to prevent backflow through the main FW pumps.

The feedwater from the FW pump discharge piping to the SG inlet must pass through the high pressure heaters. Two flow paths are available with a remote manual bypass line available around these heaters.

When the automatic AFS start signals, ESF logic A and logic B, are initiated, the FW control valves in the main FW lines and in the tempering flow lines to each SG will automatically close. In addition, a check valve in each of the above lines will also close. In order to bring these lines into operation, the operator must reset trains Logic A and B which enables these

control valves to operate. The preferred path is to use the main FW lines to the SG. A redundant path using the tempering flow lines is also available to each SG. The Train C main FW line discharge to each SG is independent of Trains A and B. The Train C tempering flow line paths are the same as Trains A and B.

In addition it was assumed that the SG blowdown line valves will be closed automatically upon receipt of ESF logic signals A and B.

4.3.1.2 Reliability Block Diagram Quantitative Analysis

The RBD model shown in Appendix B for the Byron/Braidwood AFS was manually quantified using the data from Appendix A. The failure probability calculated is for the AFS fail to start and to provide AFW to ≥2 steam generators (SGs). Failure probabilities were estimated for three initiating events (IE). Two quantitative estimates were performed assuming statistical independence of the various trains and common cause on similar or identical redundant paths. Each block in the RBD was assigned a failure rate. These failure rates were used to calculate the overall failure probability, or unavailability of the AFS.

Each initiating event (IE) was divided into the following:

- Hardware/Operator Error - this assumed that the total AFS is available and is the failure probability that the AFS fail to start and provide FW to ≥ 2 SGs.

- Test - This assumes that one of the ESF trains is being tested and the IE occurs. During the first 15 to 20 minutes of testing when the discharge valve is closed to warm up the pump in the recirculation mode, a demand for the AFS would automatically open this valve. After warm up, the discharge valve is opened as part of the test. The pump then provides full auxiliary feedwater flow to the SGs. For this analysis, the "warm up" test mode was considered. The pump discharge valve is designed to open on ESF logic signal to start the AFS flow to the SG's and thus the unreliability of these components were used. Since the pump would be running and continue to run, the failure rate of the pump to start was assumed to be zero.

- Maintenance - This assumed that if the IE occurred during the time a train was out for maintenance, this train would not be available for AFS operation and thus, the failure probability of this Train is 1.

- Human Error - This assumed that the plant personnel fail to reopen or reclose valves after a testing or maintenance operation. In the Byron/Braidwood design and procedure, this type of error appears to be greatly minimized because of automatic opening to the discharge test valve and automatic switching to the ESW. In addition, after each maintenance action on the AFS train, it will use the same test procedure as the monthly periodical test which includes a full flow test into the SGs.

The test, maintenance and human error outage contribution utilized the data in Ref. 1. Fault trees for test, maintenance and human error were developed and shown in Appendix D. Applicable portions of the RBD model were used as inputs to estimate AFS failure probability as applicable to these

trees. The sum of the failure probabilities for hardware, test, maintenance, and human error became the estimate for the AFS fail to start and provide AFW to 2 SGs on demand.

4.3.1.2.1 Statistical Independent Estimate

The statistical independent estimate assumes that all redundant components are truly independent. The results are shown on Table 4.3. These results are for automatic actions only in Trains A and B and remote manual startup from the control room of Train C. The results show that the greatest contributor to AFS unreliability is in the hardware/operator error portion with the maintenance outage a relatively close second. The test and human error portions are shown to be at least 2 orders of magnitude (10^{-2}) less than the others, thus, are insignificant contributors. This is due to the automatic valve action which keeps these trains available during test and many human error problems.

Due to a concern of an inadvertent automatic initiation of the ESW, a separate analysis assuming a remote manual backup using two valves in-line was calculated. These results are also shown on Table 4.3 for comparison with the previous results. A slight increase in the failure probabilities was noted. However, the summation, or the unavailability per demand, was essentially unchanged. Thus, use of manual actuation for ESW will not impact the overall AFS quantitative reliability for the three initiating events.

4.3.1.2.2 Common Cause Estimate

The common cause estimate assumes that redundant components are not truly independent. This estimate assumes that some commonality exists between redundant components, or trains, i.e., same maintenance personnel, same procedure, same manufacturer, same environment (humidity, temperature, earthquake, etc.), same design, etc.

In order to quickly estimate the common cause effect, a generic Beta Factor of 0.03 was used for redundant components. The electric motor driven pump and diesel driven pump were assumed to be diverse. Using these assumptions, the results are shown on Table 4.4. These results are for the automatic actions only in Trains A and B and remote manual startup from the control room of Train C. The table shows that the hardware/operator error contribute to the greatest unreliability with the maintenance outage a relatively close second. Again, the test and human error portions were found to be insignificant contributors to AFS unreliability. Hence, this AFS design and procedure has done an outstanding job of reducing the test and human error contributions which have impacted many other AFSs in the past.

4.3.1.2.3 Summary of Dominant Failure Modes

The dominant failure modes for each initiating event have been assessed by review of the dominant contributors to unreliability from the RBD in Appendix B.

LMFW with Off-site Power Available

A. Automatic start with no manual backup except for Train C.

For the statistical independent estimate and the common cause estimate, the principal dominant failure mode for the hardware/operator error portion unavailability ($\overline{A}$) is the failure of the auto-start system (ESF Logic A and B), to close the SG blowdown valves. This function is common to all four SGs and all three trains. For the maintenance outage portion of the unavailability when only two trains are available for AFS cooling, the dominant failure mode is the failure to start the 2 pumps closely followed by failure of the ESF logic system when both auto-start trains (Trains A and B) are available, or failure of either Train A or B pump and auto-start and the operator errors in manually starting Train C.

The test and human error portions of the unavailability were insignificant and thus did not contribute to the dominant failure modes.

B. Manual backup to the automatic ESF logic start

Since the auto-start ESF Logic A and B system is the principal dominant failure mode, this analysis assumed that if the auto-start system failed to operate, the operator can manually over-ride the ESF logic and manually actuate Trains A and B within sufficient time for the AFS to operate satisfactorily. Tables 4.3 and 4.4 present the quantitative statistical independent and common cause estimates. With this assumption, the unavailability ($\overline{A}$) for this initiating event reduced the independent estimate from 6.6E-5 to 7.1E-6. Likewise, the common cause estimate was reduced from 3.3E-4 to 4.0E-5, both by about one order of magnitude.

For both independent and common cause estimates, the dominant failure mode becomes the failure of the pumps to start in Trains A and B and the operator error in manually starting Train C. For maintenance outage portion of the $\overline{A}$, the dominant failure mode is the failure to start the 2 pumps when both auto-start trains (Trains A and B) are available, or either Train A or B pump fail to start and the operator errors in manually starting Train C.

Again, the test and human error portions of the $\overline{A}$ were insignificant and thus did not contribute to the dominant failure mode.

LMFW with LOOP

A. Auto-start except for Train C

For the independent estimate, the dominant failure mode for the hardware/operator error contribution to unavailability ($\overline{A}$) is the diesel generator (D/G) fail to start in Train A and the diesel pump fail to start in Train B. For the maintenance outage contribution to $\overline{A}$, the dominant failure mode is the D/G fail to start when Train B is out for repair.

For the common cause estimate, the principal dominant failure mode for the hardware/operator error portion $\overline{A}$ is the D/G fail to start in Train A and the diesel pump fail to start in Train B. Failure of the ESF logic A and B contributes almost as much as the diesel pump fail to start. For the maintenance outage contribution to $\overline{A}$, the dominant failure mode is the D/G fail to start when Train B is out for repair.

B. Manual backup to the Automatic ESF logic start

For statistical independent and common cause estimates, the principal dominant failure mode for the hardware/operator error $\overline{A}$ is the D/G fail to start in Train A and the diesel pump fail to start in Train B. For the maintenance outage contribution to $\overline{A}$, the dominant failure mode is the D/G fail to start when Train B is out for repair.

With manual backup under the LOOP, the independent $\overline{A}$ was reduced from 1.2E-3 to 6.4E-4 while the common mode was reduced from 1.4E-3 to 6.7E-4, or about a factor of 2

LMFW with LOAC

A. Auto-start without manual backup except for Train C

For the independent and common cause estimates, the dominant failure mode for the hardware/operator error $\overline{A}$ is the diesel pump fail to start in Train B, the only train available for this initiating event (IE). For the maintenance outage portion of the $\overline{A}$, the dominant failure mode is when Train B is out and no train is available for this IE. This does not consider recovery of the offsite power and main feedwater within 20 minutes.

B. Automatic start manual backup to ESF logic start

The dominant failure mode is the same as that for the auto-start only.

With this assumption, the independent estimate of the $\overline{A}$ for this IE was reduced from 2.3E-2 to 1.7E-2, and from the common cause estimate 2.3E-2 to 1.6E-2.

4.3.1.3 Overall Results of RBD Analysis

The overall result, the unavailability ($\overline{A}$) per year of the AFS was estimated to be as follows:

Method of Analysis	Auto-start, Without manual backup, on ESF Logic A and B	Auto-start on ESF Logic A and B with Manual Backup
Statistical Independent	2.4E-4/yr.	4.3E-5/yr.
Common Cause	1.0E-3/yr.	1.4E-4/yr.

This shows that the positive aspects of human operators as backup to automatic signals are worth about an order of magnitude improvement in the AFS reliability.

4.3.2 Fault Tree Quantification

In addition to the RBD analysis a fault tree analysis was undertaken to quantify the uncertainty range and provide independent check on the impact of the modeling assumptions. Three initiating events were considered:

- Loss of main feedwater (LMFW)
- Loss of offsite power (LMFW/LOOP)
- Loss of AC power (LMFW/LOAC)

4.3.2.1 Methods

The fault trees were developed to consider faults within each train which could cause failure to supply auxiliary feedwater to the steam generators within 20 minutes after an initiating event. Heavy reliance was placed on the RBD modeling for grouped component inputs. For components which dominated the train failure probabilities (i.e., pumps and drives) data at the failure mode level were used. In cases where the contribution was very small, such as piping breaks, the failure probability was assumed to be a small fraction of the valve failure probability. In most cases the data were taken from the recommended values in Ref. 1. Probability distributions and uncertainty ranges were taken from Ref. 2.

Common cause failures were considered on a case-by-case basis. Three types of common cause failures were included. First, those within a train where redundant components are found with the train. Second, those which impact two or three trains from internal components within the trains. The third type of common cause failure considered is that of external faults which impact the AFS operation but are caused by systems external to the AFS. Examples of the third type are common faults which block the steam generator output steam flow and piping faults which cause the flow to bypass the steam generators. The common cause failures were considered as explained in Appendix 6 with Beta factors which represent the ratio of the common cause failures to independent and common cause failures as measured by data from redundant systems.

Each train could be out for maintenance during a seven day period while the plant operation continues. During this period, the AFS consists of two trains. After seven days, the plant must be shutdown according to the technical specifications. The impact of maintenance is included on the fault trees at the train level and are included in the STADIC code equations. A single block is included for maintenance outage. As a result, input parameters must be carefully selected. NUREG 0611 suggests that maintenance is performed on a train about once every 4.5 months for an average of 19 hours. The range on the maintenance time could be from one-half hour to 170 hours.

The point estimate is:

$$\frac{1}{4.5\ \text{Month}} \times 12\ \text{months/year} \times \frac{\text{yr}}{8760\ \text{hr.}} \times 19\ \text{hr.} = 0.0057$$

The log normal range between the 50th and the 95th is estimated to be 3. These assumptions are made for each train even though it is unlikely that maintenance on Train C could be performed while the plant is in operation.

The impact of maintenance outage is greatest on the independent estimate, but it only contributes 10% to 20% when common cause failures are included.

The uncertainty for this unavailability is estimated to be 3 for a log normal distribution. This accounts for the more frequent short outages for minor repair and for the less frequent major repairs. The impact of maintenance outage on the AFS reliability contributes about 20% to the system unreliability.

If maintenance is not properly performed and if inspection and testing did not uncover the fault, AFS unreliability would appear as a failure to start in the data base. If a maintenance action could negate 2 or 3 trains, then the fault would be counted as a common cause fault. Such conditions are included in the fault tree as common cause faults which are the dominant contributors to failures in redundant systems.

Table 4.1 Criteria for NUREG 0611 Reliability Comparison - Ref. 1

LMFW and LOOP	AC Blackout
Low Reliability	**Low Reliability**
. Manual actuation*	. AC dependencies-turbine lube pumps*
. Minimum redundancy - 2 pump	
. Single point failure	
. No time limit on train outage	
Medium Reliability	**Low to Medium Reliability**
. Auto start with manual backup	. AC dependencies - valves with local manual control*
. Single point failure*	
. No time limit on train outage	. No time limit on train outage
. Water-hammer concerns	
. System interactions (safety and non-safety)	
. Human interactions	
. tests not staggered	
. test by same personnel and same shift	
. Testing incapacities more than one train	
High Reliability	**High Reliability**
. High redundancy	. No identifiable AC power dependencies
. Auto start with manual backup	. Auto start with manual backup
. No observed single point failure	
. Human interactions	
. Tests staggered and different shifts	
. Testing incapacitates only one train	
. Time limit on train outage	

*Dominant contributor.

Table 4.2 NUREG 0611 Qualitative Reliability Analysis

IE	NUREG ASSESS. CRITERIA	NUREG CRITERIA	BYRON/BRAIDWOOD REMARK AND ASSESSMENT
LMFW	High	. High redundancy	. Three trains
		. Auto start with manual backup.	. Two trains-auto start with manual backup; third train - manual start.
		. No observed single point failure.	. Comply
		. Human interactions . Test staggered and different shifts. . Maintenance valve inadvertently left closed.	. Exceeds criteria. Testing does not incapacitate ESF train. Maintenance suction valve inadvertently left closed does not incapacitate ESF train.
		. Testing incapacitates only one train.	. Exceed criteria. Testing does not incapacitate ESF train.
		. Time limit on train outage.	. Tech Spec requirement on train outage. Qualitative Assessment: Med/High range
LOSP	Low	. Minimum redundancy (2 pump)	. 2 Pump system
	High	. Same as LMFW except minimum redundancy.	. Same as LMFW except minimum redundancy. Qualitative Assessment:High/Med range.
LOAC	High	. No identifiable AC power dependencies.	. Complies
		. Auto start with manual backup.	. Complies Qualitative Assessment: Med/High range.

Table 4.3 Statistical Independent Estimate

Freq. Per Year	Trans. Event	Byron/Braidwood AFS Reliability Estimates					Unavail Per Demand	Unavail Per Year
		Method	Hardware	Test	Maintenance	Human Error		
With Automatic Service Water Backup to Condensate Storage Tank								
3	LMFW		5.7E-5	2.1E-7	8.6E-6	2.9E-9	6.6E-5	2.0E-4
.02	LOSP		8.0E-4	2.8E-6	3.5E-4	3.2E-8	1.2E-3	2.4E-5
$5x10^{-4}$	LOAC		1.7E-2	5.3E-5	6.0E-3	3.6E-7	2.3E-2	1.2E-5
								$\sum$2.4E-4
Manual Water Backup to Condensate Storage Tank								
3	LMFW		5.8E-5	2.1E-7	8.7E-6	1.1E-8	6.7E-5	2.0E-4
.02	LOSP		8.1E-4	2.9E-6	3.5E-4	4.3E-7	1.2E-3	2.4E-5
$5x10^{-4}$	LOAC		1.7E-2	5.3E-5	6.0E-3	7.2E-6	2.3E-2	1.2E-5
								$\sum$2.4E-4
Automatic Startup With Manual Backup								
3	LMFW		3.3E-6	1.6E-8	3.8E-6	6.2E-10	7.1E-6	2.1E-5
.02	LOSP		3.7E-4	1.4E-6	2.7E-4	1.8E-8	6.4E-4	1.3E-5
$5x10^{-4}$	LOAC		1.0E-2	4.0E-5	5.9E-3	4.3E-7	1.7E-2	8.5E-6
								$\sum$4.3E-5

Table 4.4 Common Cause Estimate $\beta = .03$

Freq. Per Year	Trans. Event	Byron/Braidwood AFS Reliability Estimates					Unavail Per Demand*	Unavail Per Year
		Method	Hardware	Test	Maintenance	Human Error		
Automatic Start Without Manual Backup								
3	LMFW		3.1E-4	1.1E-6	1.5E-5	5.1E-5	3.3E-4	9.9E-4
.02	LOOP		1.0E-3	3.7E-5	3.5E-4	3.8E-8	1.4E-3	2.8E-5
$5x10^{-4}$	LOAC		1.7E-2	6.6E-5	6.0E-3	5.8E-7	2.3E-2	1.2E-5
								$\sum$ 1.03E-3
Automatic Start With Manual Backup								
3	LMFW		3.4E-5	1.3E-7	5.5E-6	1.3E-9	4.0E-5	1.2E-4
.02	LOOP		4.0E-4	1.5E-6	2.7E-4	[illegible].3E-8	6.7E-4	1.3E-5
$5x10^{-4}$	LOAC		1.0E-2	4.0E-5	5.9E-3	4.3E-7	1.6E-2	8.0E-6
								$\sum$ 1.4E-4

Table 4.5 Auxiliary Feedwater System Unavailability+

Initiating Event	Frequency Per Yr[t]	Unavailability Per Demand Mean (Median)		Unavailability Per Year Mean (Median)	
		Independent	Independent* +CC	Independent	Independent + CC.*
Loss of Main Feedwater	3.0	1.8×10^{-5} (1×10^{-5})	7.2×10^{-4} (7×10^{-4})	6×10^{-5} (3×10^{-5})	2.4×10^{-3} (2.1×10^{-3})
Loss of Offsite Power	.02	1×10^{-3} (7×10^{-4})	2×10^{-3} (1.6×10^{-3})	3×10^{-5} (1.2×10^{-5})	5×10^{-5} (2.7×10^{-5})
Loss of all AC	5×10^{-4}	6.9×10^{-2} (6.3×10^{-2})	7.2×10^{-2} (6.7×10^{-2})	3.5×10^{-5} (4.2×10^{-5})	3.6×10^{-5} (4.3×10^{-5})

t = Median value
* cc = Common cause failure
\+ = Includes hardware failures, testing and maintenance errors, human errors, and maintenance outage.

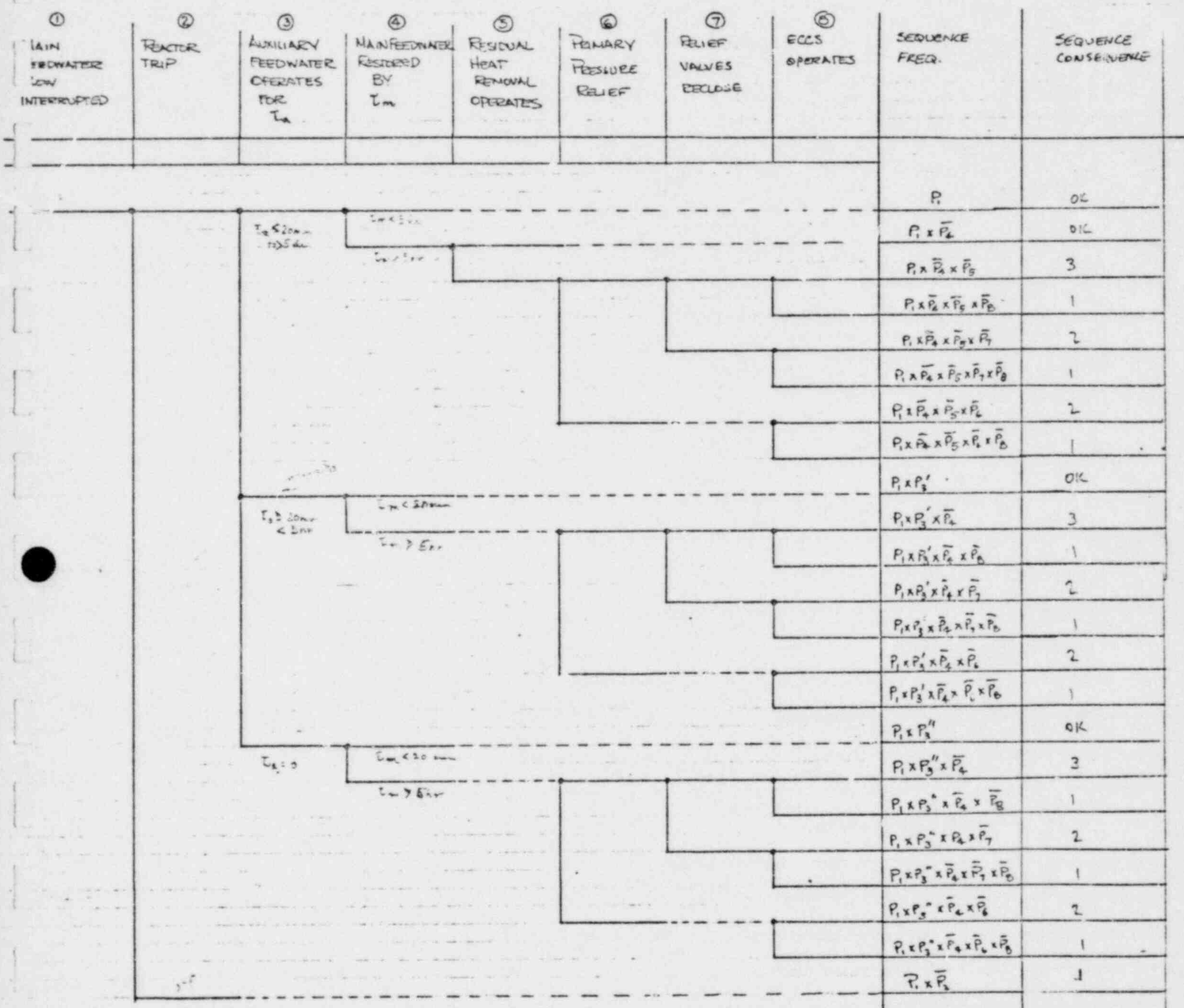

Figure 4.1 Event Tree for LMFW Events with Electric Power

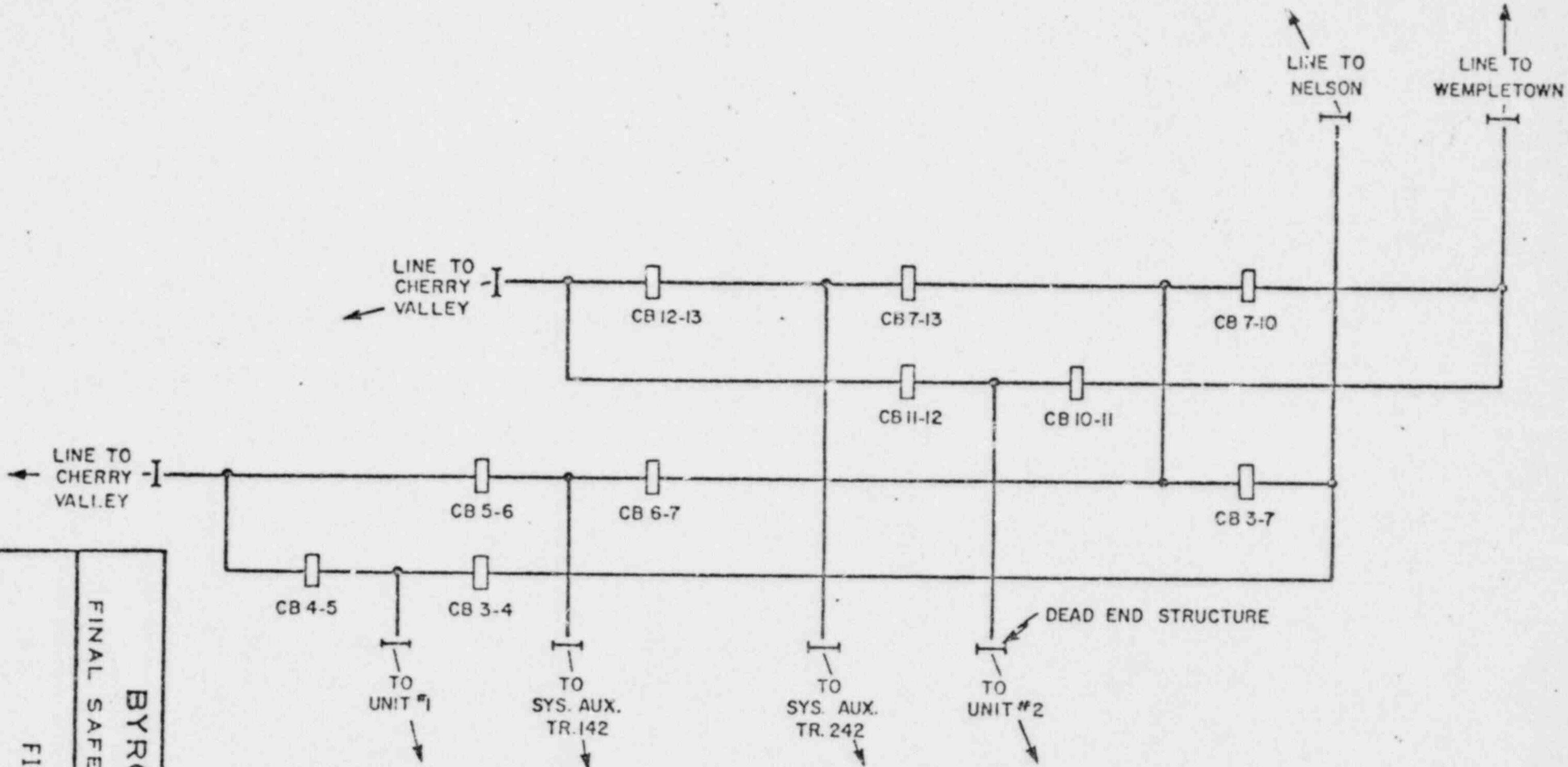

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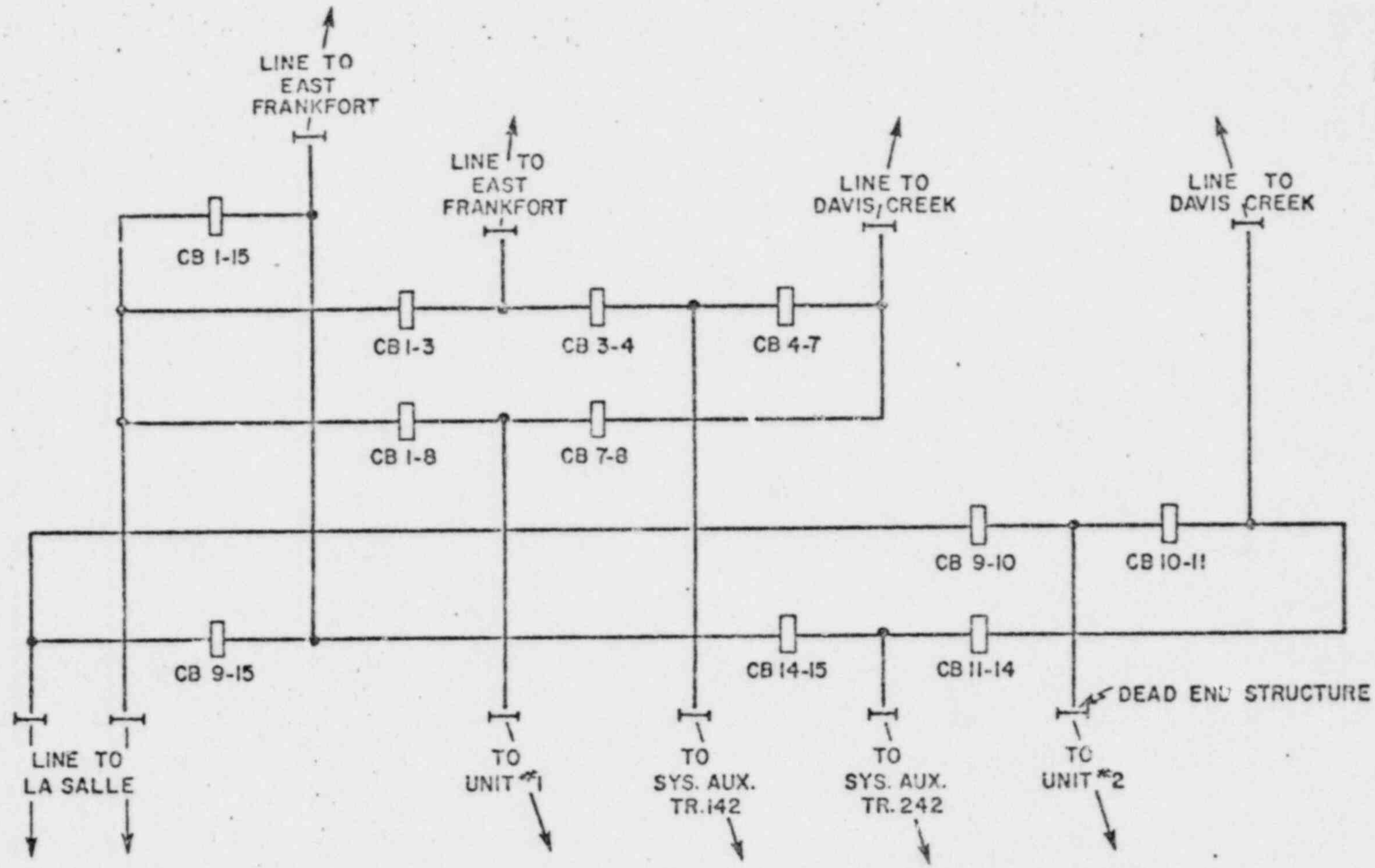
FIGURE 4.2

345-kV SWITCHYARD BUS ARRANGEMENT



BRAIDWOOD STATION
FINAL SAFETY ANALYSIS REPORT

FIGURE 4.3

345-kV SWITCHYARD BUS ARRANGEMENT

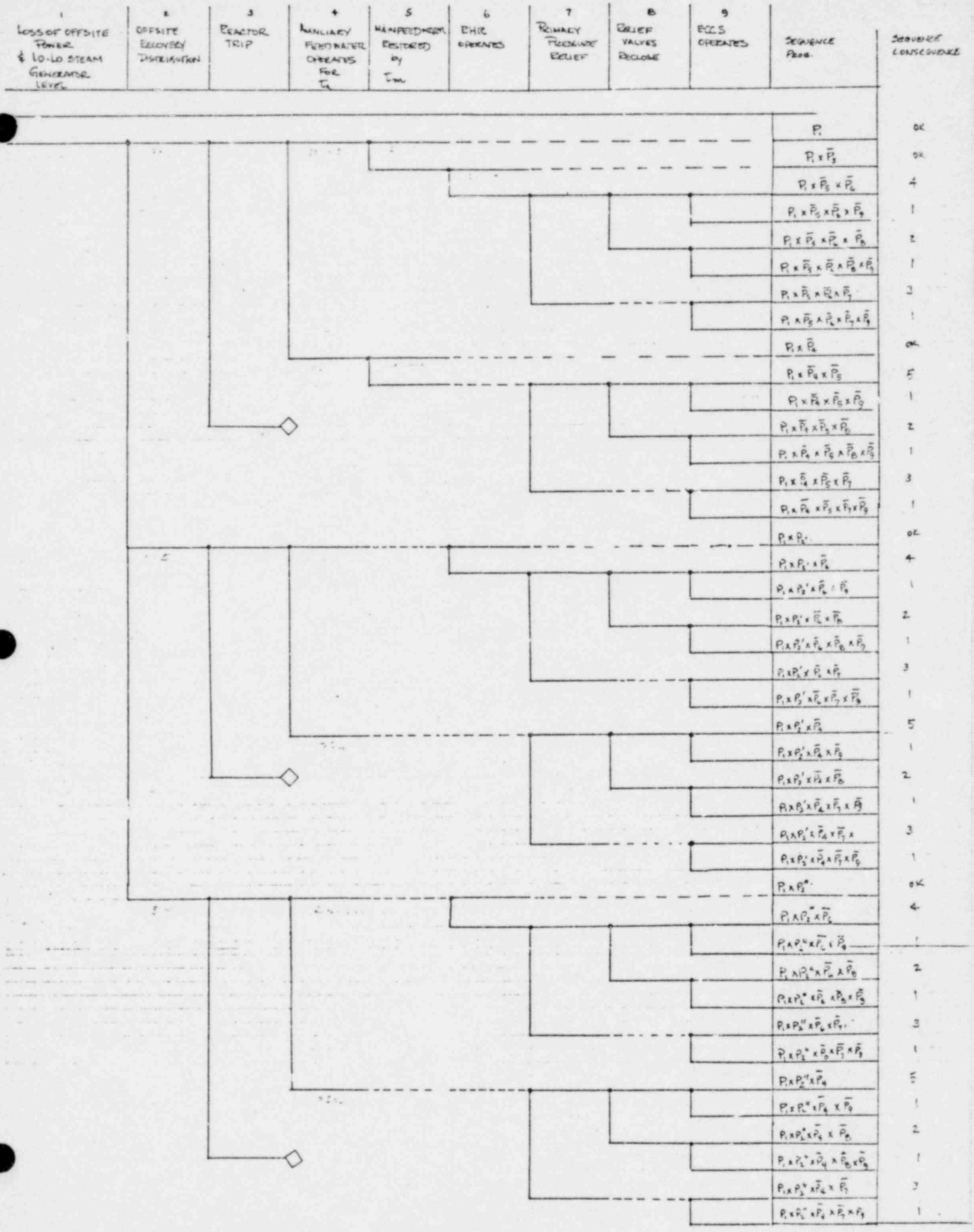
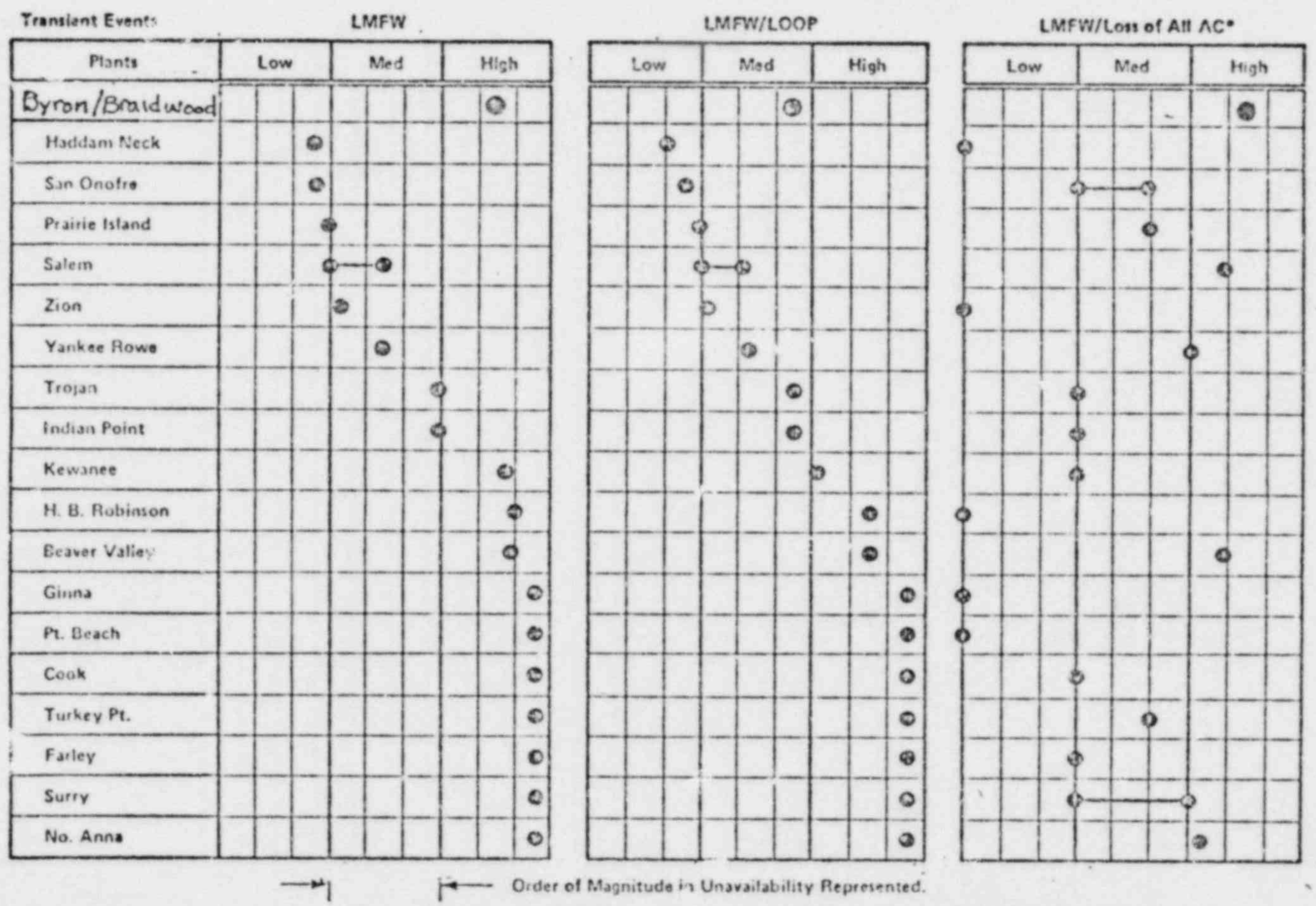


Figure 4.4 Event Tree for LMFW/LOOP and LMFW/LOAC Events



*Note: The scale for this event is not the same as that for the LMFW and LMFW/LOOP.

Figure 4.5 Reliability Characterizations for AFS Designs in Plants Using the Westin house NSSS

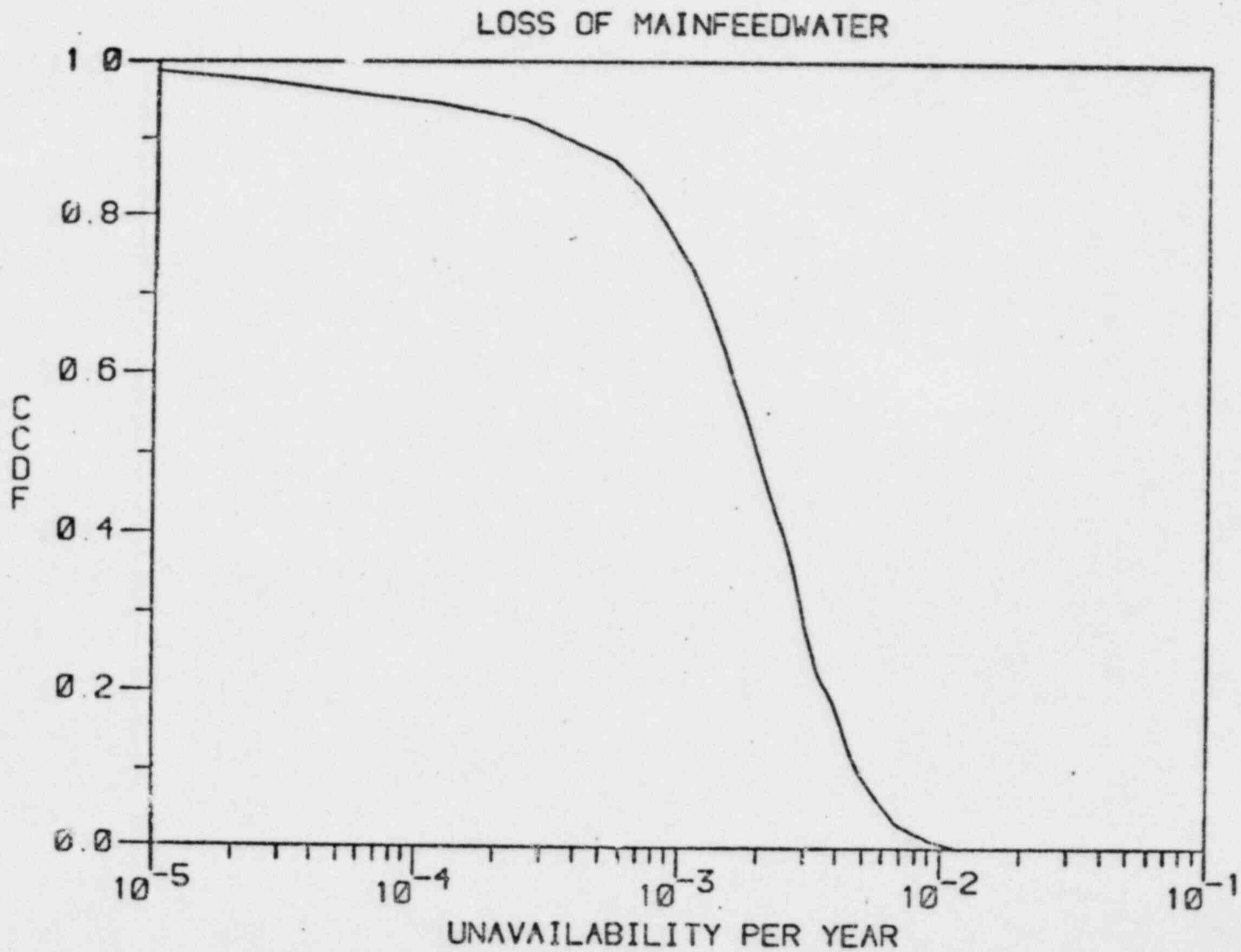


Figure 4.6 Probability Distribution for Auxiliary Feedwater System Unavailability Given a Loss of Feedwater Event

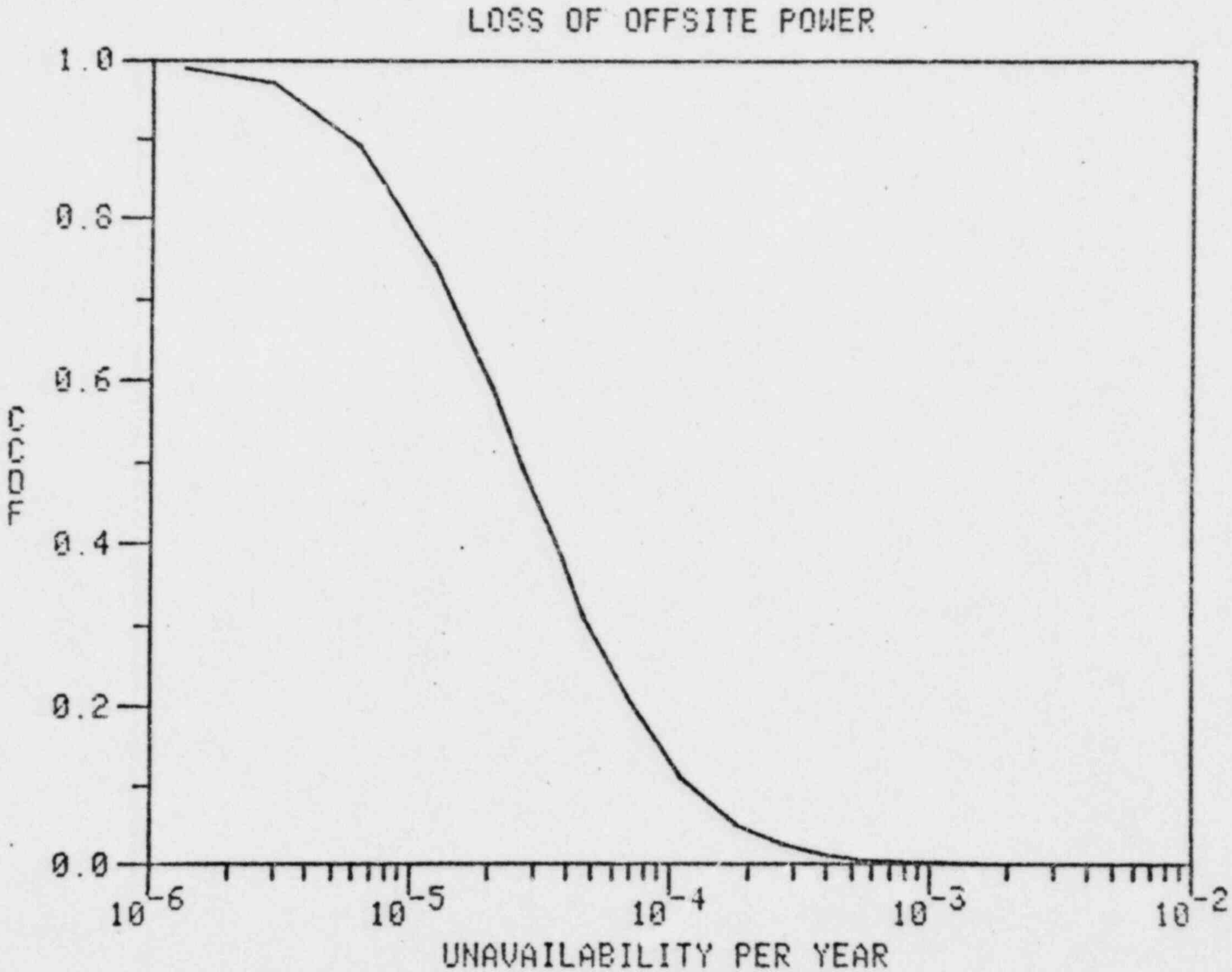


Figure 4.7 Probability Distribution for Auxiliary Feedwater System Unavailability Given a Loss of Offsite Power

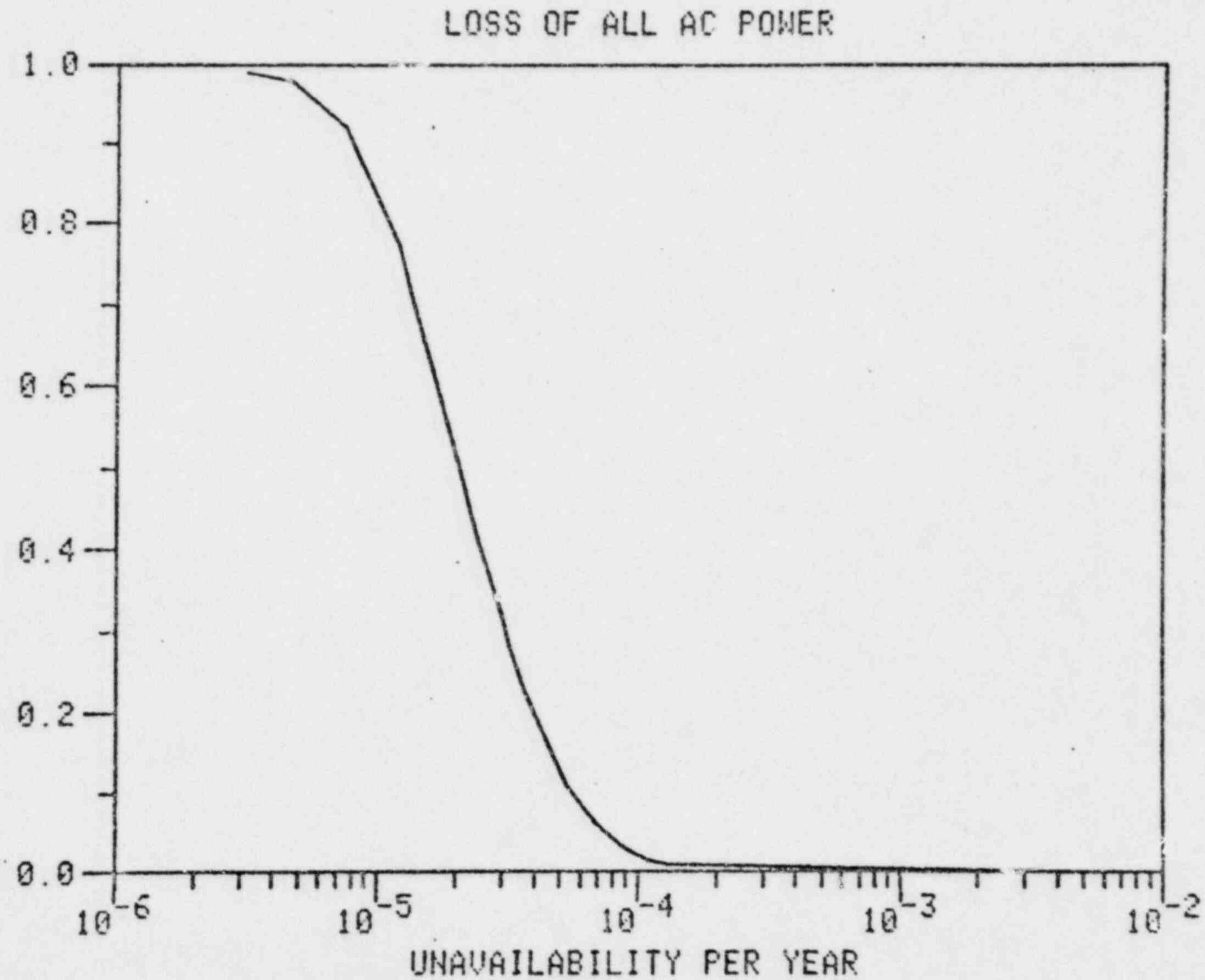


Figure 4.8 Probability Distribution for Auxiliary Feedwater System Unavailability Given a Loss of all AC Power

5.0 References

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2. WASH-1400 (NUREG-75/014) Reactor Safety Study an Assessment of Accident Risks in U.S. Commercial Nuclear Plants, 1975.

3. Atwood, C. L., "Common Cause and Individual Failure and Fault Rates for Licensee Event Reports of Pumps at U.S. Commercial Nuclear Power Plants", EGG-EA-5289, (Draft), November, 1980.

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5. "Reliability of Off-Site Power Supply to Zion Station", November 5, 1979.

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APPENDIX A

DATA BASE

APPENDIX A

DATA BASE

A I. Basic Reliability Data - Used for the Byron/Braidwood Analyses

The basic reliability data (failure rates) utilized for the quantitative analysis were taken from Ref. 1 (NUREG 0611) and are repeated here in Table A.I. When the required data were not given in Ref. 1, then Ref. 2 (WASH-1400) was utilized and these data are repeated here in Table A.I.2. Ref. 1 was also utilized for the test and maintenance outage contributions, human acts and error failure data and the frequency of occurrence or the various initiating events per reactor year. In two cases, however, special data searches were undertaken.

In the case of the diesel driven pump, no data were given in References 1 and 2. Therefore, a special investigation was undertaken to identify the best data sources. A first cut sample was reviewed from Ref. 3 and this was updated from Trojan. The results are that (1) the diesel driven pump, after correction of early problems has achieved a reliability equivalent to turbine driven pumps, and (2) the diesel pump is more reliable on start than the diesel generators by about a factor of three.

In the case of loss of offsite power frequency, the operational characteristics of the particular area were felt to be more important than the industry-wide average. Direct data from the B/B power lines indicates that the outage frequency in those areas is at least a factor of 10 less likely than the NUREG-0611 estimate which includes data from Florida where the connections are more prone to outages than the midwestern high voltage grids. The data input to the fault tree evaluation is presented in Table AIV.1.

A II. AFW Pump Data:

A II.1 AFWS Data From Ref. 3

A review of data sources showed that the generic base presented in Ref. 1 for pumps does not differentiate between the type of drives. Most AFW pumps are either electric-motor- or steam-turbine-driven. Ref. 3, which is a study of LER (Licensee Event Report) pump failures, listed a table of PWR plant data with the number of the various AFW pump type, critical hours, calendar months, etc.. Ref. 3 also had a listing of the PWR AFW pump problems from LERs. The AFW pump problems were divided into the following categories as required by the model, or various components and classifications:

- Fail to start or Fail to run after start.
    - Motor drive
    - Turbine drive
    - Diesel drive

- Auto/Manual start circuit
- Human/Operator error

To estimate a failure rate from that data the system demand cycles were first determined by assuming one test per pump per month plus three AFS start requirements per reactor year. This first-cut comparison is summarized in Table A II.1.1.

In this first cut analysis the diesel drive AFW pump appears to have a very high failure rate, both demand and run. Therefore, additional investigation was undertaken to more accurately understand the nature of the failures and the frequency estimates. This indepth investigation is discussed in Section A II.2.

A II.2 Trojan Nuclear Power AFS Experience

The Reference 3 listing of AFS drives showed that the only diesel driven auxiliary feedwater pump with any experience was at the Trojan Nuclear Power Plant. Trojan utilizes their AFS for normal startup and shutdown operations.[2]

Thus, the numbers of AFS starts were underestimated in Section A II.1, causing the failure rate of the AFS diesel pump to be high. The Graybooks (through March 1981) and the annual operation reports for the years 1975, 1976, and 1977 were used to estimate the numbers of starts and runs that the Trojan AFS had experienced.

From initial criticality (12/15/75) through 2/29/76, Trojan was experiencing many problems with both the turbine and diesel drive pumps. Docket 50344-250 summarizes the major equipment changes and modifications to improve the reliability of the AFS. These changes and modifications occur between 2/29/76 through 3/21/76. The data were analyzed in two parts, before corrective actions (CA) and after CA. The AFS components and its failure modes, estimated numbers of starts and runs, and estimated time to restore the system are presented in Tables A II.2.2.

Tables A II.2.1 summarizes the data and provides estimates of the failure rates for different modes of failure, including both before corrective action and after corrective action. After CA, the diesel pump and its controls was estimated to be 5.2×10^{-3}/demand. Including a human factor "fail to start", the diesel pump would be estimated to be 1.0×10^{-2}/demand. Reference 2 (WASH-1400) diesel/generator fail to start is assessed at 3×10^{-2} with an error factor of 3. Thus, the Trojan diesel AFW pump failure rate per demand appears to be less than that of Ref. 2 diesel/generator. Including the before and after data for the pump and its controls without operator error, the failure rate per demand would be

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2. Conversations with the plant operating personnel.

assessed at 7/254 = 0.029 or with operator error at 8/254 = 0.031. In both cases, these assessments are practically the same as that for diesel/generators in Ref. 2.

The Trojan experience on both the diesel pump and its controls (without human factor) fail to start on demand of 0.0052 and the turbine drive and its controls of 0.0063 are practically the same. The turbine drive and its controls from Section A II was assessed at 0.0068. Thus, from this data, it can be concluded that diesel pump fail to start failure rate is at least as good as for the turbine pumps, and about a factor of 3 better than the diesel generator start failure rate.

A.III Reliability of Commonwealth Edison Off-Site Power Supplies (5)

The reliability of high voltage supplies in the midwest are significantly better than the U.S. average figures of 0.2 to 0.3/yr quoted in NUREG 0611. For the B/B units the outage frequency is expected to be 0.08/year in which 50% are restored within 1 hour. The dominant contributors are double line outages at about 0.017/station year and loss of an auxiliary transformer at 0.064/transformer year. Hence, the B/B stations have a loss of offsite power frequency which is a factor of 3 lower than the average figures presented in NUREG 0611. The following description from reference 5 describes in more detail the reasons behind these figures.

A.III.1 Description of Off-Site Power Supplies

The reliability of the offsite supplies for the auxiliary power systems at B/B stations is a function of the reliability of the two auxiliary power transformers connected to the 345 kV bus, the bus itself with the associated circuit breakers, and four or six 345 kV lines emanating from the bus Byron, Braidwood respectively. These lines, in turn, are connected to an extensive generation and transmission network.

Sources of power of the offsite supply are Edison's extensive interconnected 345 kV network and the B/B units themselves. About 9500 MW of Edison's generation is connected to the 765 and 345 kV systems at four and six stations, respectively. In addition, 6300 MW of generation is connected to the 138 and 69 kV systems, which in turn are tied to the 345 kV system at numerous locations through large autotransformers. Edison also has 29 high voltage interconnections with adjacent companies.

These high voltage ties with other companies are supported by the bulk power facilities of the interconnected systems east of the Rocky Mountains. The entire bulk power electric system from parts of Canada to the Gulf of Mexico and from Nebraska to the east coast (except Texas) is a single interconnected network of generation and high voltage transmission. Edison and the systems directly tied to it are the members of one or the other of three regional reliability councils -- Mid-America Interpool Network (MAIN), Mid-Continent Area Reliability Coordination Agreement (MARCA), and the East Central Area Reliability Council (ECAR). These three regional reliability councils are among nine such councils which form the National Electric Reliability Council.

Periodically, the MAIN Engineering Committee through the cooperation of its members and adjacent councils, conducts Extreme Disturbance Studies. These studies examine the strength of the interconnected network to withstand very severe but highly improbable contingencies. Events such as the loss of an entire power plant, the severing of a complete EHV right-of-way, and total destructions of a major EHV substation are examples of the types of contingencies studied. These studies have shown the reliability of MAIN and its surrounding regions to be very reliable and resistive to power interruptions at the stations.

Electrical diagrams of the connections at the Byron/Braidwood 345 kV bus are shown in Figures 4.2 and 4.3.

A.III.2 Interruption of Off-Site Supply

Given certain events, it is possible that the offsite supply to the auxiliary power system could be interrupted temporarily as discussed in the following paragraphs:

A. Outage Frequency of a Reserve Auxiliary Power Transformer

The outage of either Auxiliary Transformer No. 142 or No. 242 would result in loss of supply to the associated auxiliary system. However, operating experience with this type of transformer (system aux.) has been very good with only one failure on the Edison system to date. There are seventeen reserve auxiliary power transformers on the system including fossil and nuclear units with an aggregate of about 113 transformer years of operation. The outage frequency of these transformers is 0.064 per year. Switching facilities are available to supply the essential service bus of one unit from the other unit system reserve auxiliary power transformer. Such switching can be accomplished manually within thirt minutes. Approximately 50 percent of the transformer outages are caused by troubles which have been corrected within one hour. The other outages involved failures which require major repairs.

B. 345 kV Bus Outage

The 345 kV bus at both Byron and Braidwood Stations is a double ring configuration with each line and major piece of equipment connected to a separate bus section. A fault will be cleared in the same manner whether it occurs on the bus section or on the line connected to it; thus, single bus faults do not lead to the outage of more than one line. A common outage would require conditions which have not yet occurred on the Commonwealth Edison System.

The simultaneous outage of certain pairs of 345 kV transmission lines result in the loss of power supply to a reserve auxiliary power transformer. In particular, from Figures 4.2 and 4.3 the simultaneous outage lines on both sides of CB3-4 and CB4-7 for Braidwood and CB5-6 and CB6-7 for Byron will leave auxiliary transformer 142 de-energized. A simultaneous outage of lines CB-12-15 and CB11-14 for Braidwood and CB12-13, CB7-13 for Byron will similarly affect transformer 242. The lines in each pair have separate towers and do not have a common circuit

breaker. The frequency of these double line outages is expected to be 0.017 per year with an average outage duration of approximately seven hours.

The two-line outages cited above must be either simultaneous or occur within thirty minutes to affect the supply to the auxiliary transformers for the following reasons. In the event of a fault on one unit the breakers on either side of the line position would open, clearing the fault and leaving the rest of the lines in service. If it were determined that the line could not be re-energized immediately, then disconnects on the line would be opened, and the breakers closed to restore the ring bus. Similarly, when a line is taken out for maintenance, it is isolated from the bus and the bus is restored as a ring. Thus, if a line were out of repair, and a fault occurred on another line, the reliability of off-site supply to the reserve transformers would not be affected significantly.

Other multiple line outage events which affect the supply to the auxiliary transformers include simultaneous outages of three, four, five or six lines. The probabilities of these more severe outages occurring are not significant compared to the probability of the two conditions described above. Also, the relay protective system at B/B incorporates a high speed bus sectionalizing scheme to enhance the stability of the B/B units. The probability of these conditions occurring is relatively insignificant because three-phase line faults are involved with less than one percent of Commonwealth Edison's 345 Kv line tripout experiences.

Certain catastrophic events, such as a tornado or an airplane accident, could cause such outages if substantial damage were done to the B/B 345 kV bus or lines. These catastrophic events ocur at a frequency which is not significant when compared to the transformer and line outages described above.

A.III.3 <u>Quantitative Evaluation of Reliability of the Off-Site Power Supply</u>

In evaluating the reliability of the typical plant off-site power supply, failure is considered to result whenever the engineered safety feature bus is without a power source for longer than ten seconds, neglecting the unit itself and the diesel generators as power sources. The probability of such a condition existing is defined as P_F where:

$$P_F = \frac{\text{Downtime (hrs/year)}}{\text{Uptime + Downtime (hrs/year)}}$$

$$= \frac{\text{Downtime (hrs/year)}}{\text{8760 (hrs/year)}}$$

$$= \frac{.064x1 + .017x7}{8760} = 2 \times 10^{-5}$$

where:

0.064 = outage frequency per year of the reserve auxiliary transformer

1 hr = median outage duration

0.017 = double line outage per year of the 345 kV transmission lines

7 hr = average outage duration

The probability of failure of the B/B off-site power supplies for Unit 1 is estimated about 2×10^{-5}.

Failure would result during the specific conditions of certain line and/or reserve transformer outages described earlier. The most probable of these conditions are the outage of transformer 142 or the simultaneous outage of supply lines for Unit 1. For Unit 2, the critical conditions are the outage of transformer 242 or the simultaneous outage of its supply lines.

As a basis of a quantitative prediction of the reliability of the B/B off-site power supply, outage data for Commonwealth Edison's 345 kV transmission lines has been compiled for the period from 1965 through 1975. Corresponding outage data for the auxiliary power transformers was based on Commonwealth Edison Company experience supplemented with data obtained from the Edison Electric Institute Apparatus Trouble Report, prepared by the Electrical System and Equipment Committee.

In summary, the loss of off-site power supplies to the essential service buses of the B/B units due to all causes described above are expected to occur once in approximately twelve years (Freq. = 0.081/year) with an outage duration of one half hour which is required for switching to the alternate line or transformer supply.

A.III.4 Estimate of LOOP and LOAC Frequency Per Year

The loss of offsite power supplies to the B/B units essential service buses due to all causes described above are expected to occur with a frequency of 0.081/year. The major contributor to this frequency is the auxiliary transformer failure at 0.064/year. This failure could easily be corrected by switching to good lines, hence is not a contributor after 20 minutes. The other contributor is the double line outages at 0.017/year with an average outage duration of approximately 7 hours. Thus, the LOOP frequency per year for the B/B units is estimated to be 0.017. This is a factor of ten improvement over the average values in NUREG 0611.

The LOAC frequency per year is directly related to the LOOP frequency per year. The diesel/generator fail to start is assessed at 3×10^{-2}. Thus the LOAC frequency per year is $0.017 \times 3 \times 10^{-2}$ = 5.1E-4 or 5E-4.

Table A.I.1 Recommended Data from NUREG 0611

BASIC DATA USED FOR PURPOSES OF CONDUCTING A COMPARATIVE ASSESSMENT OF EXISTING AFWS DESIGNS & THEIR POTENTIAL RELIABILITIES

Component (Hardware) Failure Data	Point Value Estimate of Probability of* Failure on Demand
a. Valves:	
Manual Valves (plugged)	$\sim 1 \times 10^{-4}$
Check Valves	$\sim 1 \times 10^{-4}$
Motor Operated Valves	
. Mechanical Components	$\sim 1 \times 10^{-3}$
. Plugging Contribution	$\sim 1 \times 10^{-4}$
. Control Circuit (local to Valve)	
w/quarterly tests	$\sim 6 \times 10^{-3}$
w/monthly tests	$\sim 2 \times 10^{-3}$
Piston Actuated Valves	
. MOV-Mechanical Components	$\sim 3 \times 10^{-4}$
. SOV-Mechanical Components	$\sim 1 \times 10^{-3}$
. Control Circuit (Note: Use MOV Failure Rate if Valve is not Fail Safe)	ε**
b. Pumps: (1 Pump)	
. Mechanical Components	$\sim 1 \times 10^{-3}$
. Control Circuit (Local to Pump - applies to Electrical Pumps)	
w/Quarterly tests	$\sim 7 \times 10^{-3}$
w/Monthly tests	$\sim 4 \times 10^{-3}$
c. Actuation Logic (Assumes at least 1 of 2 logic)	$\sim 7 \times 10^{-3}$/train

*Error factors of 3-10 (up and down) about such valves are not unexpected for basic data uncertainties.

**ε represents a number so small in magnitude that it may be neglected for basis of this study.

Table A.I.2

Data Tables from WASH 1400 (Ref. 2)

FAILURE MODES		ASSESSMENT MEDIAN	LOWER & UPPER BOUND
Pumps	Failure To Start	1×10^{-3}/D	3×10^{-4} 3×10^{-3}
	Failure To Run – Normal	3×10^{-5}/HR	3×10^{-6} 3×10^{-4}
	Failure To Run – Extreme ENV	1×10^{-3}/HR	1×10^{-4} 1×10^{-2}
Valves MOV	Fails To Operate	1×10^{-3}/D	3×10^{-4} 3×10^{-3}
	Failure To Remain Open (Plug)	1×10^{-4}/D	3×10^{-5} 3×10^{-4}
	External Leak Or Rupture	1×10^{-8}/HR	1×10^{-9} 1×10^{-7}
Valves (SOV)	Fails To Operate	1×10^{-3}/D	1×10^{-4} 3×10^{-3}
Valves (AOV)	Fails To Operate	3×10^{-4}/D	1×10^{-4} 1×10^{-3}
	Failure To Remain Open (Plug)	1×10^{-4}/D	3×10^{-5} [illegible]
	External Leak – Rupture	1×10^{-8}/HR	1×10^{-9} 1×10^{-7}
Valves (Check)	Failure To Open	1×10^{-4}/D	3×10^{-5} 3×10^{-4}
	Internal Leak	3×10^{-7}/HR	1×10^{-7} 1×10^{-6}
	External Leak – Rupture	1×10^{-8}/HR	1×10^{-9} 1×10^{-7}
Valves (Vacuum)	Failure To Operate	3×10^{-5}/D	1×10^{-5} 1×10^{-4}
	Rupture	1×10^{-8}/HR	1×10^{-9} 1×10^{-7}
Valves, Orifices, Flow Meters, Tees	Rupture	1×10^{-8}/HR	1×10^{-9} 1×10^{-7}
Valves (Manual)	Failure To Remain Open (Plug)	1×10^{-4}/D	3×10^{-5} 3×10^{-4}
Valves (Relief)	Fail To Open/D	1×10^{-5}/D	3×10^{-6} 3×10^{-5}
	Premature Open/HR	1×10^{-5}/HR	1×10^{-6} 3×10^{-5}
Pipes ≤ 2" Hi Quality	Rupture (Gross)	1×10^{-10}/HR	3×10^{-12} 3×10^{-9}
Pipes > 3"	Rupture	1×10^{-9}/HR	3×10^{-11} 3×10^{-8}
Gaskets	Leak	3×10^{-6}/HR	1×10^{-7} 1×10^{-4}
Flanges, Closures, Elbows	Leak/Rupture	3×10^{-7}/HR	1×10^{-8} 1×10^{-5}
Welds	Leak	3×10^{-9}/HR	1×10^{-10} 1×10^{-7}
Diesel (Complete Plant)	Failure to Start (d)	3×10^{-2}/D	1×10^{-2} 1×10^{-1}
(Emergency Loads)	Failure to Run (e)	3×10^{-3}/HR	3×10^{-4} 3×10^{-2}
Diesel (Engine Only)	Failure to Run (e)	3×10^{-4}/HR	3×10^{-5} 3×10^{-3}
Batteries, Power Supplies	NO Output (e)	3×10^{-6}/HR	1×10^{-6} 1×10^{-5}
Instrumentation	Failure to Operate (e)	1×10^{-6}/HR	1×10^{-7} 1×10^{-5}
(Amplifiers, Annunciators, Transducers, Combinations)	Shift Calibration (e)	3×10^{-5}/HR	3×10^{-6} 3×10^{-4}

FAILURE MODES		ASSESSMENT MEDIAN	LOWER & UPPER BOUND
CLUTCH ELEC	Failure to Operate	3×10^{-4}/D	1×10^{-4} - 1×10^{-3}
	Premature Open	1×10^{-6}/HR	1×10^{-7} - 1×10^{-5}
CLUTCH MECH	Failure to Open	3×10^{-7}/HR	3×10^{-8} - 3×10^{-6}
	Failure to Operate	3×10^{-4}/D	1×10^{-4} - 1×10^{-3}
SCRAM RODS	Failure to Insert (Single Rod)	1×10^{-4}/D	3×10^{-5} - 3×10^{-4}
ELECTRIC MOTORS	Failure to Start	3×10^{-4}/D	1×10^{-4} - 1×10^{-3}
	Failure to Run	1×10^{-5}/HR	3×10^{-6} - 3×10^{-5}
	Failure to Run (Extreme ENVIR)	1×10^{-3}/HR	1×10^{-4} - 1×10^{-2}
RELAYS	Failure to Energize	1×10^{-4}/D	3×10^{-5} - 3×10^{-4}
	Failure NO Contact to Close	3×10^{-7}/HR	1×10^{-7} - 1×10^{-6}
	Short Across NO/NC Contact	1×10^{-8}/HR	1×10^{-9} - 1×10^{-7}
	Open NC Contact	1×10^{-7}/HR	3×10^{-8} - 3×10^{-7}
SWITCHES	Limit: Failure to Operate	3×10^{-4}/D	1×10^{-4} - 1×10^{-3}
	Torque: Fail to OPER	1×10^{-4}/D	3×10^{-5} - 3×10^{-4}
	Pressure Fail to OPER	1×10^{-4}/D	[illegible]
	Manual, Fail to TRANS	1×10^{-5}/D	3×10^{-6} - 3×10^{-5}
	Contacts Short	1×10^{-8}/HR	1×10^{-9} - 1×10^{-7}
CIRCUIT BREAKERS	Failure to Operate	1×10^{-3}/D	3×10^{-4} - 3×10^{-3}
	Premature Transfer	1×10^{-6}/HR	3×10^{-7} - 3×10^{-6}
FUSES	Premature, Open	1×10^{-6}/HR	3×10^{-7} - 3×10^{-5}
	Failure to Open	1×10^{-5}/D	3×10^{-6} - 3×10^{-5}
WIRES	Open	3×10^{-6}/HR	1×10^{-6} - 1×10^{-5}
	Short to GND	3×10^{-7}/HR	3×10^{-8} - 3×10^{-6}
	Short to PWR	1×10^{-8}/HR	1×10^{-9} - 1×10^{-7}
TRANSFORMERS	Open CKT	1×10^{-6}/HR	3×10^{-7} - 3×10^{-6}
	Short	1×10^{-6}/HR	3×10^{-7} - 3×10^{-6}
SOLID STATE DEVICES Hi PWR Applications	Fail to Function	3×10^{-6}/HR	3×10^{-7} - 3×10^{-5}
	Shorts	1×10^{-6}/HR	1×10^{-7} - 1×10^{-5}
Low PWR Applications	Fails To Function	1×10^{-6}/HR	1×10^{-7} - 1×10^{-5}
	Shorts	1×10^{-7}/HR	1×10^{-8} - 1×10^{-6}

TABLE A II.1.1

FIRST CUT AFS Component Estimated Failure Rates from Ref. 3 Data

Pump Type	Est. No. of Starts (1)	Fail To Start (Cause)	Start Failure Rate Per Demand	Est. No. of Runs (Cause	Fail to Run	Run Failure Fraction	Run Failure Rate Per Hour
Pump and its Controls							
Turbine	2803	19(2)	.0068	2745	13	.0047	5×10^{-4}
Motor	3132	2	.00064	3067	5	.0016	2×10^{-4}
Diesel	35	4	.11	25	5	.2	2×10^{-2}
Auto Start Circuit							
Turbine	560	9cc*(3)	.016				
Motor	626	3(4)	.0048				
Diesel	7	1	.14				

NOTES:

(1) Assumed one test per month per train plus 3 AFS requirements per reactor year.
(2) 3 Pumps fail to start during test. 1 pump had control problems and 2 pumps had excessive tight packing.
(3) 3 pumps fail to autostart. Manual started 3 pumps O.K. Fuses not installed. 7 days after initial criticality and before on-line.
(4) 2 electric drive pump fail to autostart due to defective switches. Turbine drive pump started O.K.

*cc - Common cause failure.

TABLE A II.1.1 (cont'd)

FIRST CUT AFS Component Estimated Failure Rates from Ref. 3 Data

Pump Type	(1) Est. No. of Starts	Fail To Start (Cause)	Start Failure Rate Per Demand	Est. No. of Runs (Cause	Fail to Run	Run Failure Fraction	Run Failure Rate Per Hour
Miscellaneous							
Turbine	2803	2 (e) (e)	.00071	2745	4 (b)(h) (2)(b) (h)(g)	.0015	
Motor	3132	3 (c)(d) (e)	.00096	3067	7 (f)(h) (2)(f) (h)(2) (f)(h) (3)(h)(3)	.0023	
Diesel	35	1(a)	.029	25	0	>.04	

Outage Causes

(a) Inadequate procedure, (b) No. 3 SG FW reg bypass valve recirculating (Haddem Neck unique).,(c) pump motor breaker, (d) 480V bus breaker, (e) unknown, (f) air in suction header, (g) recirculator line and orifice redesign, (h) plugged strainer during plant startup.

Notes

(1) Assumed 1 test per month per train plus 3 AFS requirements per reactor year.
(2) During startup, AFW pumps flow reduced due to startup-strainers plugged.
(3) During test, 2 motor-drive pumps lost suction due to air in the section header.

TABLE A II.1.1 (cont.)

FIRST CUT AFS Component Estimated Failure Rates from Ref. 3 Data

Pump Type	(1) Est. No. of Starts	Fail To Start (Cause)	Start Failure Rate Per Demand	Est. No. of Runs (Cause	Fail to Run	Run Failure Fraction	Run Failure Rate Per Hour
			OPERATOR ERROR				
Turbine and its control	2803	8	.0029	2745	0		
Motor & its control	3132	4 (3)	.0013	3067	0		
Diesel & its control	35	2	.057	25	0		
Auto start turbine	560	1 cc*(2)	.0018				
Auto start motor	626	0	<.0016				
Auto start diesel	7	1 cc*(2)	.14				

Notes

(1) Assumed 1 test per month per train plus 3 AFS requirements per reactor year.

(2) All pumps fail to auto start, manual start one pump O.K.. Mislogged leads in auto-start circuitry.

(3) During monthly test, two electric drive pumps failed to start until third manual attempt. Auxiliary oil pump not energized.

*Common cause failure.

Table A II.2.1

TROJAN NUCLEAR POWER PLANT AFS EXPERIENCE FROM OPERATIONAL REPORTS

Pump Type	Est. No. Starts	Fail To Start	Start Failure Rate/ Demand	Est. No. of Runs	Fail to Run	Run Failure Rate/ Demand
			Pump and Its Controls Before CA (Corrective Actions)			
Diesel	51	6	.12	42	2	.048
Turbine	25	2	.08	25	2	.08
			After CA			
Diesel	194	1	.0052	190	5	.026
Turbine	159	1	.0063	157	1	.0064
			Auto Start Circuit Before CA			
Logic B	9	0	<.11			
Logic A	9	0	<.11			
			After CA			
Logic B	28	1	.036			
Logic A	28	0	<.036			

*cc = common cause

Table A II.2.1 (Cont'd)

TROJAN NUCLEAR POWER PLANT AFS EXPERIENCE FROM OPERATIONAL REPORTS

Pump Type	(1) Est. No. Starts	Fail To Start	Start Failure Rate/ Demand	Est. No. of Runs	Fail to Run	Run Failure Rate/ Demand
			Operator Error Before CA			
Diesel Auto Start	9	1 cc*(1)	.11			
Turbine Auto Start	9	1 cc*(1)	.11			
			After CA			
Diesel Auto Start	28	1 cc(2)	.036			
Turbine Auto Start	28	1 cc(2)	.036			
			After CA			
Diesel & Its Controls	194	1	.0052	190	0	.0052
			Miscellaneous before CA			
Diesel	51	1(3)	.020			
Turbine	51	0	< .020			
			After CA			
Diesel	194	0	< .0052	190	0	< .0052
Turbine	159	0	< .0063	157	0	< .0064

(1) Mislogged lifted leads (manual start O.K.)
(2) Wiring error (Manual start O.K.)
(3) Procedure inadequate.

Table A II.2.2 EVENT SUMMARY

Component Type: Diesel Driven AFW Pump and its Controls

Date	Time to Restore	Cause
		Failure Mode: Fail to run after starting.
12/24/75	>20m, <5h	Fail to set current limiter on speed controller.
1/4/76	1.1h	Speed signal lead vibrated loose.
2/14/76	>20m, <5h	Jacket cooling water setpoint changed.

12/75-2/76 Operational runs: 39 plant startups and shutdowns.
POT-5-1: 3 at 1 per month

2/29/76 - 3/21/76 Corrective Actions and Verification Tests (Docket 50344-250)

Date	Time to Restore	Cause
3/28/76	<20m	Insufficient margin on overspeed setpoint.
9/2/76	>5h, ~7days	Loose adjustment spring in jacket temp. sensing device.
9/9/76	Same	Same
3/24/77	145h	Broken crank shaft.
2/7/80	>20m,<5h	Broken cooling water hose.

3/76-2/81 Operational runs: 83 plant startups and shutdowns.

Verification Tests: 48 POT-5-1: 59 at 1 per month

Table A II.2.2 EVENT SUMMARY (Cont'd)

Component Type: Diesel Driven AFW Pump and its Controls

Date	Time to Restore	Cause
		Failure Mode: Fail to start on demand.
12/19/75	>20m, <5h	Special sequence required after each engine shutdown to reset fuel racks and control circuitry.
12/24/75	<20m	Fail to reset fuel rocks.
1/9/76	5h	Misaligned governor.
1/23/76	<20m	Low lube oil press., cold start, 2nd start O.K.
1/24/76	<20m	Same
1/24/76	<20m	Same
2/29/76	2m	Cold start, overspeed trip, local manual start. (Turbine drive pump declared inoperable 2 days before.)

12/75 - 2/76 Operational starts: 48 plant startup and shutdowns.
POT-5-1: 3 at 1 per month

2/29/76 - 3/21/76 Corrective Actions and Verification Tests (Docket 50344-250)

12/17/77	>20m, <5h	Speed microswitch, out of adjustment.
9/3/80	>20m, <5h	No details-fail to start during test.

3/76 - 2/81 Operational starts: 86 plant startups and shutdowns.

Verification Tests: 48 POT-5-1: 60 at 1 per month

Table A II.2.2 EVENT SUMMARY (Cont'd)

Component Type: Diesel Driven AFW Pump and its Controls

Date	Time to Restore	Cause
	Component Type: AFW Pumps **Failure Mode : Declared Inoperable**	
1/13/76	24.4 (16.0 repair)	Turbine-steam leak
2/27/76	~25 days*	Turbine-governor oil problem.
3/29/76 - 3/2/76	Corrective Actions and Verification Tests (Docket 50344-250)	
12/17/77	>20m, <5h	Turbine - could not be reset for auto start, limit switch failure.
12/28/78	>20m, <5h	Diesel - fuel oil leak
4/15/79	>20m, <5h	Diesel - fuel line crack
	Component Type: AFW Isolation Valve **Failure Mode : Stuck partially open.**	
11/17/77	>5 h	AFWIV between diesel pump and "B" SG stuck partially open (90%). Damaged sealing gasket allowed water to leak into main operator.

*2 days later on 2/29/76, diesel pump pump fail to auto start. Remote manual start failed. Local manual start OK after resetting overspeed trip. Started in about 2 minutes after initial failure.

Table A II.2.2 EVENT SUMMARY (Cont'd)

Component Type: Diesel Driven AFW Pump and its Controls

Date	Time to Restore	Cause
	Component Type: AFS auto start system. **Failure Mode: Fail to initiate auto start signal.**	
1/16/76	<20m (>5h repair)	Mislogged lifted leads in auto start circuitry. Both diesel and turbine pumps fail to start. Manually started one of the pumps (cc)*.
	12/77 - 2/76 Auto starts: 9	
2/29/76 - 3/25/76		No change indicated in auto start system (Docket 50344-250)
9/3/76	<20m (<5h repair)	Blown fuse in diesel pump auto start circuitry. Manual start O.K.
10/3/80	<20m (>5h repair)	Wiring error, both diesel and turbine pumps fail to auto start. Assumed manual start O.K. (cc)*.
10/11/80 circuitry	<20m (5h repair)	Blown fuse in diesel pump auto start Assumed manual start O.K.
	3/76 - 2/81 Auto starts: 28	

*Common cause.

Table A.IV.1 Fault Tree Data Input

Components	Symbol	Beta (Error Factor)	Data Median Q (Demand)	Error Factor*	Dist.	Ref.	Comments
1. Bypass to condenser blocked	QP_1	-	$5x10^{-8}$	10	ln	2	
2. Relief Valves insufficient	QP_2	$.001^{(3)}$	$1x10^{-5}$	10	ln	2	Common to A,B, & C Note a.
3. Steam Generator manual override fails	QP_3	NR	10^{-2}	.02**	Normal		15 min. NR is nonredundant
4. Spurious signal for SG isolation	QP_4	$.01^{(2)}$	$1x10^{-4}$	3	ln	2	Common to A, B, C Note a.
5. Condensate tank rupture	QP_{64}	NR	$1x10^{-6}$	3	ln	2	
6. SG check valve FW037 fails to open	QP_6	$.01^{(2)}$	$1x10^{-4}$	3	ln	2	Common to A and B Note a.
7. CST valve AF001A fails to open	QP_8	$.03^{(3)}$	$1x10^{-5}$	3	ln	1,2	Since valve is already open 10% of rate in Ref. 1. Common with QP_{29} Note a.
8. CST valve 1CD183 fails to open	QP_9	$.03^{(3)}$	$1x10^{-5}$	3	ln		Common with QP_{61} Note a.

$*\frac{Q95\%}{Q50\%}$

$**(Q_{95}-Q_{50})$

Table A.IV.1 Fault Tree Data Input

Components	Symbol	Beta (Error Factor)	Data Median Q (Demand)	Error Factor*	Dist.	Ref.	Comments
9. ESW valve AF017A fails to open	QP_{10}	.1(2)	$1.3x10^{-3}$	2.3	ln	1	Common with QP_{31}. Note a.
10. ESW valve AF006A fails to open	QP_{11}	.1(2)	$1.3x10^{-3}$	2.3	ln	1	Common with QP_{32} Note a.
11. Mainfeed water check valves fail to close (Train A)	QP_7	.01(2)	10^{-4}	3	ln	2	Common with QP_{30},QP_{51} Note a.
12. Train A valves AF003 fail to open	QP_{12}	.03(3)	$1x10^{-5}$	3	ln	1,2	(1)Common to Train B QP_{34} Note a.
13. Train A SG valves AF014A fail to open	QP_{13}	.03(3)	$1x10^{-5}$	3	ln		Common to Train B Note a.
14. Loss of aux. transformer supply	QP_{14}	NR	$2x10^{-5}$	10	ln		Appendix A fails during cool down.
15. Diesel generator 1A fails to operate	QP_{15}	NR	$3x10^{-2}$	3	ln	1,2	
16. Electrical manual override fails	QP_{16}	NR	$1x10^{-2}$	$1x10^{-1}$**	Normal	2	15 min.

$*\frac{Q95\%}{Q50\%}$

$**(Q_{95}-Q_{50})$

Table A.IV.1 Fault Tree Data Input

Components	Symbol	Beta (Error Factor)	Data Median Q (Demand)	Error Factor*	Dist.	Ref.	Comments
17. Electrical control automatic startup failure	QP_{17}	$.1^{(2)}$	$7x10^{-3}$	.003**	Normal .003	1	Common to Train B Note a.
18. Pump lubrication fault	QP_{18}	NR	$6x10^{-3}$	3	ln		
19. Pump cooling insufficent	OP19	(included in 18 above)					
20. Pump fault	QP_{20}	(included in 18 above)					
21. Pump motor	QP_{21}	(included in 18 above)					
22. Loss of lines to 345 KV switchyard	QP_{22}	Incl.	$1x10^{-5}$	3	ln		Appendix A
23. ACB 1414 (Breaker) fails to close	QP_{23}	$.1^{(2)}$	$1x10^{-3}$	3	ln	2	Common to QP_{24} Note a.
24. ACB 1412 (Breaker) fails to open	QP_{24}	$.1^{(2)}$	$1x10^{-3}$	3	ln	2	Note a.
25. Diesel Generator 1A fails to start	QP_{25}	NR	$3x10^{-2}$	3	ln	2	
26. ACB 1413 closes	QP_{26}	$.1^{(2)}$	$1x10^{-3}$	3	ln	2	Common to QP_{27} Note a.
27. ACB 1412 Opens	QP_{27}	$.1^{(2)}$	$1x10^{-3}$	3	ln	2	

$*\frac{Q95\%}{Q50\%}$

$**(Q_{95}-Q_{50})$

Table A.IV.1 Fault Tree Data Input

Components	Symbol	Beta (Error Factor)	Data Median Q (Demand)	Error Factor*	Dist.	Ref.	Comments
28. ACB 1414 remains open	QP_{28}	-	$1X10^{-6}$	3	ln	2	
29. Condensate storage tank check valve AF001B fails to open	QP_{29}	.02(3)	$1x10^{-4}$	3	ln	1	Note a.
30. Main feedwater check valves in Train B fail to close	QP_{30}	.01(3)	$1x10^{-4}$	3	ln	1	Note a.
31. ESW valve to Tain B (AF006B) fails to open	QP_{31}	.1(2)	$1x10^{-4}$	3	ln	1	Note a.
32. ESW valve to Train B (AF017B) fails to open	QP_{32}	.1(2)	$1x10^{-4}$	3	ln	1	Note a.
33. Train B check valves to SG's fail to open	QP_{33}	.01(2)	$1x10^{-4}$	3	ln	1	Note a.
34. Train B valve AF003B fails to open	QP_{34}	.03(2)	$1x10^{-5}$	3	ln	1,2	Common to QP_{12}

*$\frac{Q95\%}{Q50\%}$

**$(Q_{95}-Q_{50})$

Table A.IV.1 Fault Tree Data Input

Components	Symbol	Beta (Error Factor)	Data Median Q (Demand)	Error Factor*	Dist.	Ref.	Comments
35. DC start batteries fail	QP_{35}	NR	$1x10^{-6}/H$ $3x10^{-4}/D$	3	ln	IEEE	1/2 PTI
36. Fuel supply failure (diesel drive pump) pump)	QP_{36}	NR	$3x10^{-3}$	3	ln	NPRDS	3 fuel fail. 26 tot. fail x $3x10^{-2}$
37. Automatic signal failures	QP_{37}	.1(2)	$7x10^{-3}$	2	ln	1	Common to QP_5 Note a.
38. Manual override failure (diesel pump)	QP_{38}	NR	$1x10^{-2}$	3			
39. Internal engine failure (diesel) in cooling system	QP_{39}	.001(2)	$1x10^{-3}$	2	ln		Common to QP_{40} Note a.
40. External pump failure (diesel) in cooling system	QP_{40}	.001(2)	$1x10^{-3}$	3	ln	2	Common to AC QP_{39} Note a.
41. Internal engine failure (lubrication system)	QP_{41}	.001(2)	$2x10^{-3}$	2	ln	NPRDS 2	$3/52x3x10^{-2}$ Note a.
42. External AC pump (lubrication system)	QP_{42}	.001(2)	$1x10^{-3}$	3	ln	2	Note a.

*C95%
Q50%

**(Q_{95}-Q_{50})

Table A.IV.1 Fault Tree Data Input

Components	Symbol	Beta (Error Factor)	Data Median Q (Demand)	Erro Factor*	Dist.	Ref.	Comments
43. Gear box failure	QP43	.001(2)	(2×10^{-4}/D) 5×10^{-5}/hr	3			NPRDSQ = $5\times5\times10^{-4}$ NPRDS assumes 5 hr operation Note a.
44. Diesel engine air intake blocked	QP_{44}	NR	2×10^{-3}	3	ln	2	5 hr run
45. Diesel engine air intake blocked	QP_{45}	NR	10^{-3}	2	ln	NPRDS	2/50 x 3×10^{-2}
46. Diesel engine exhaust blocked	QP_{46}	NR	10^{-3}	2	ln		1/50 x 3×10^{-2}
47. Diesel engine breakdown	QP47	NR	10^{-3}	2	ln		
48. Rupture of condenser hotwell (Train C)	QP_{49}	NR	10^{-6}	3	ln		est.
49. Trains A and B SG check valves fail to open	QP_{50}	.01(3)	10^{-4}	3	ln	1	Common Trains A,B,C Note a.
50. Train C check valves (A) fail to open	QP_{51}	.01(3)	10^{-4}	3	ln	1	Note a.

$*\frac{Q95\%}{Q50\%}$

$**(Q_{95}-Q_{50})$

Table A.IV.1 Fault Tree Data Input

Components	Symbol	Beta (Error Factor)	Data Median Q (Demand)	Error Factor*	Dist.	Ref.	Comments
51. Train C check valves (B) fail to open	QP_{52}	$.01^{(3)}$	10^{-4}	3	ln	1	Note a.
52. BUS 157 fails to recover (loss of offsite power)	QP_{53}	NR	.5	3	ln	App.A	Power failure for Train C .5 in 1 hr.
53. Breaker switching failure	QP_{54}	.1(2)	$1x10^{-3}$	3	ln	2	Common QP_{26}
54. Pump (Train C) mech. failure	QP_{55}	.01(3)	$1x10^{-4}$	3	ln	1	Common to QP_{57} to QP_{60} Note a
55. Pump fails to initiate on (Train C)	QP_{56}	NR	$4x10^{-3}$	3	ln	1	
56. Condensate/ booster pump drive fails	QP_{57}	.03(3)	$6.8x10^{-4}$	3	ln	RBD	Common to QP_{57} to QP_{60} Note a.
57. Condensate/ booster pump	QP_{58}	(same as above)					
58. Condensate/ booster pump valve alignment changed	QP59	(same as above)					
59. Condensate/ booster pump strainer fails	QP_{60}	(same as above)					

$*\frac{Q95\%}{Q50\%}$

**(Q95-Q50)

Table A.IV.1 Fault Tree Data Input

Components		Symbol	Beta (Error Factor)	Data Median Q (Demand)	Error Factor*	Dist.	Ref.	Comments
60.	Condensate tank valve 1CD091 plugged	QP_{61}	$.03^{(3)}$	$1x10^{-5}$	3	ln	1,2	Normally open valves. Common to QP_{61}, QP_{62}, QP_{63}, QP_9 Note a.
61.	Valve 1CD149 plugged.	QP_{62}	$.03^{(3)}$	$1x10^{-5}$	3	ln	1,2	Note a.
62.	Valve 1CD111	QP_{63}	$.03^{(3)}$	$1x10^{-5}$	3	ln	1,2	Note a.
63.	Train A, B or C out for maintenance.	XLM	-	$5.7x10^{-3}$	3	ln	1,2	Only one loop out at a time. (.22/month x 19 hour repair)

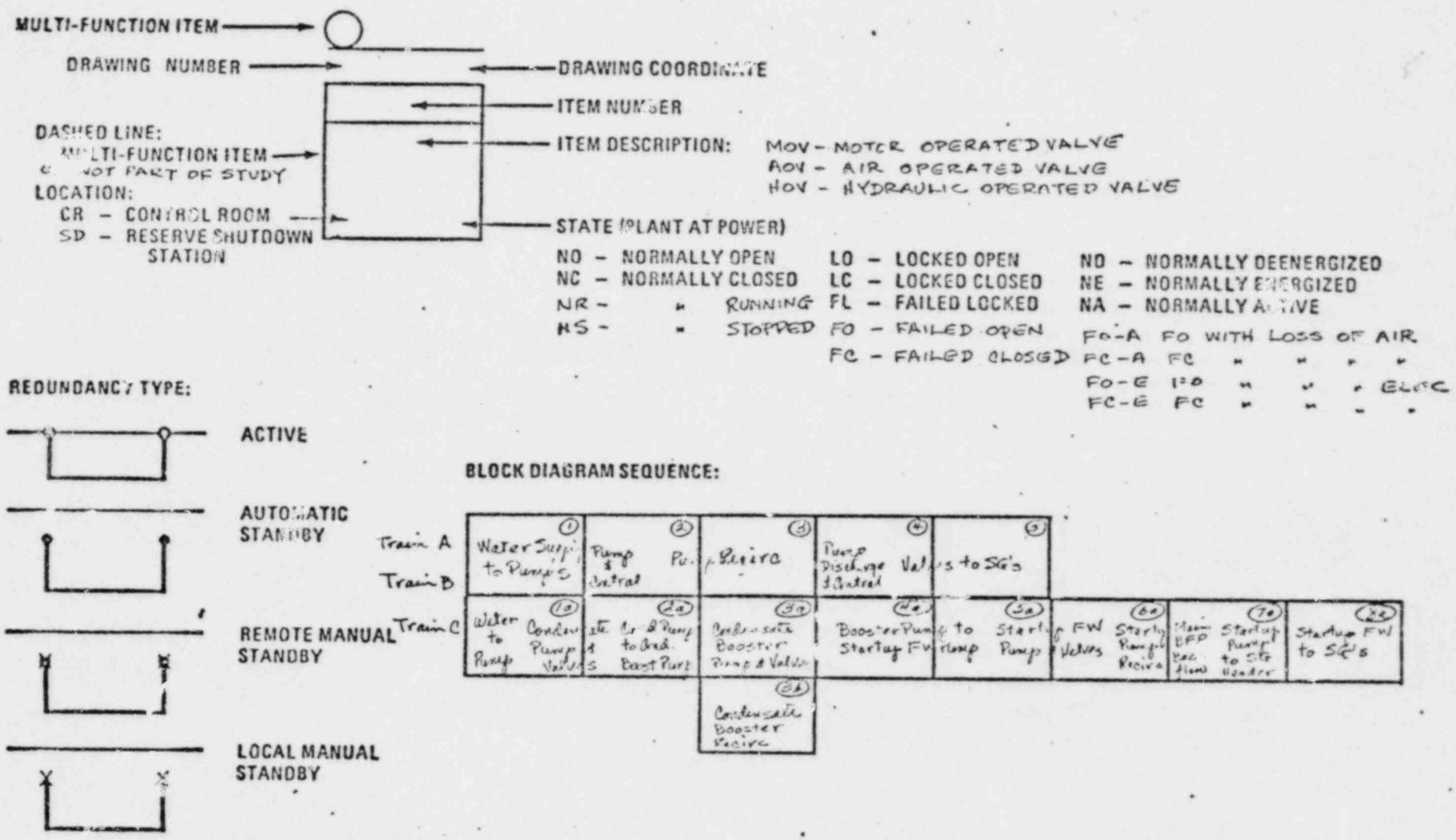
NOTE a. The Beta factors for each redundant pair of components was not adjusted for the time to restore the fault after first discovered. A 20 minute adjustment could amount to reduction of 50% in some Beta's. All Beta's also characterized by normal distribution.

$*\frac{Q95\%}{Q50\%}$

$**(Q_{95}-Q_{50})$

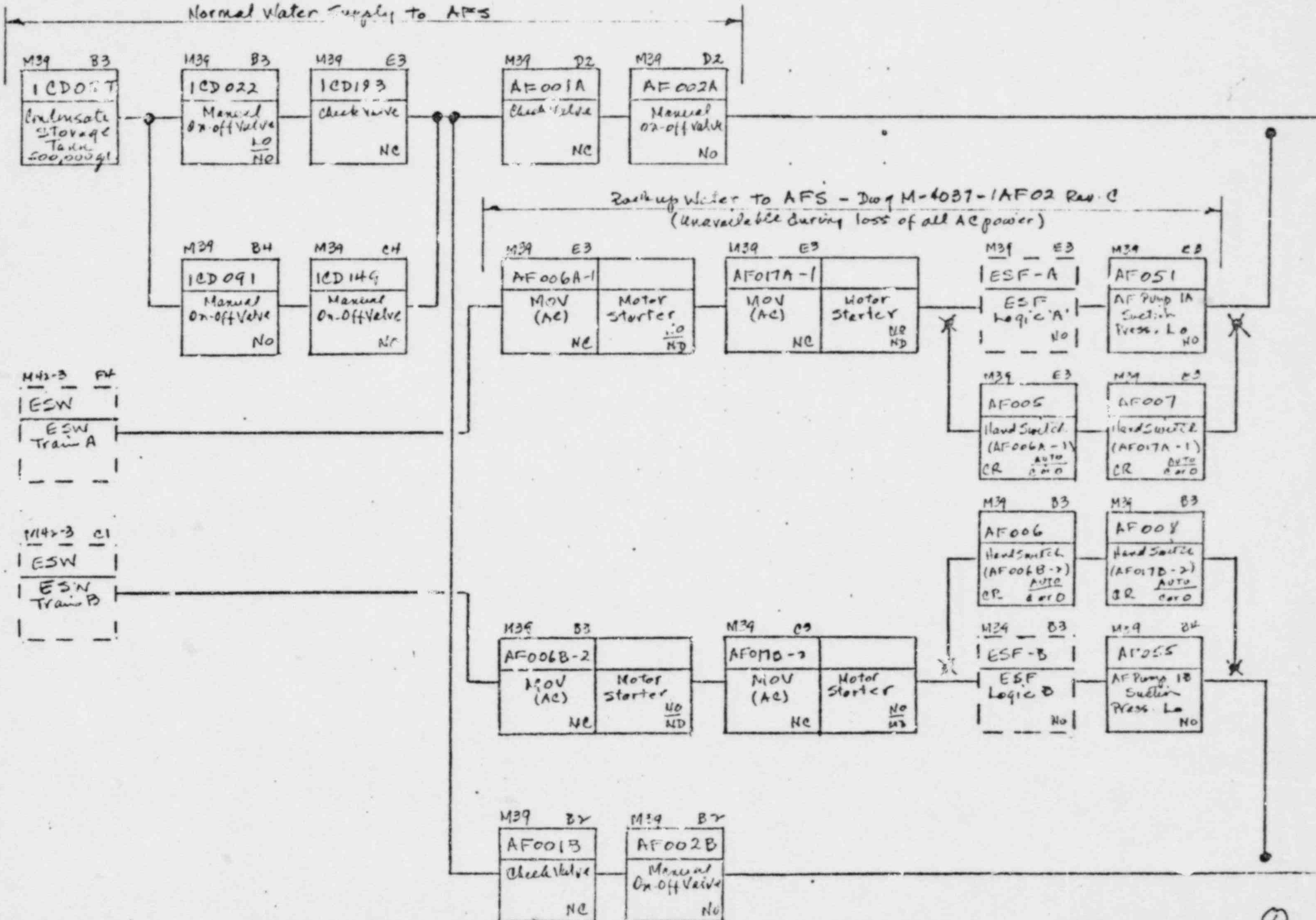
APPENDIX B

AFS RELIABILITY BLOCK DIAGRAM

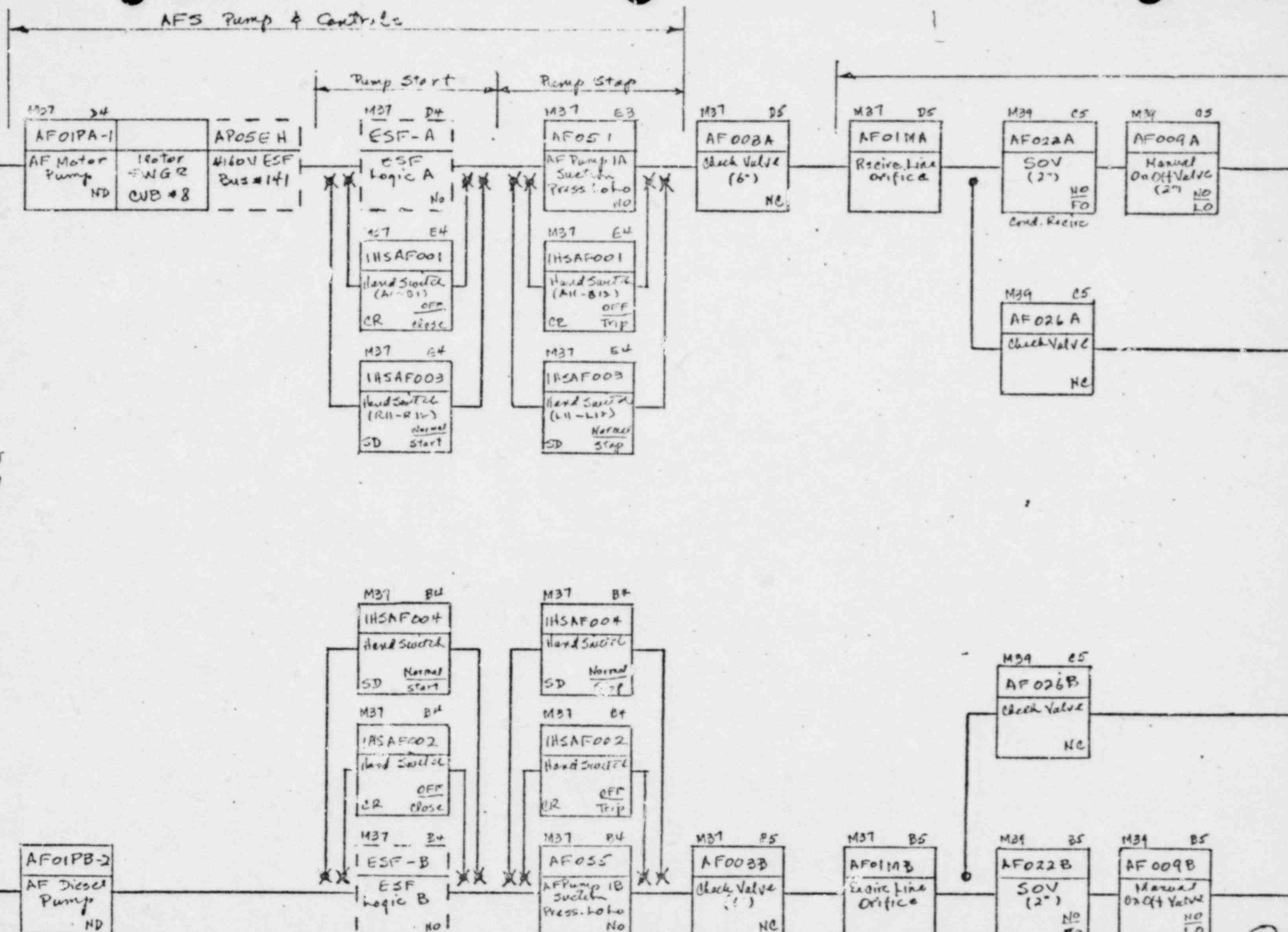


AFS RELIABILITY BLOCK DIAGRAM

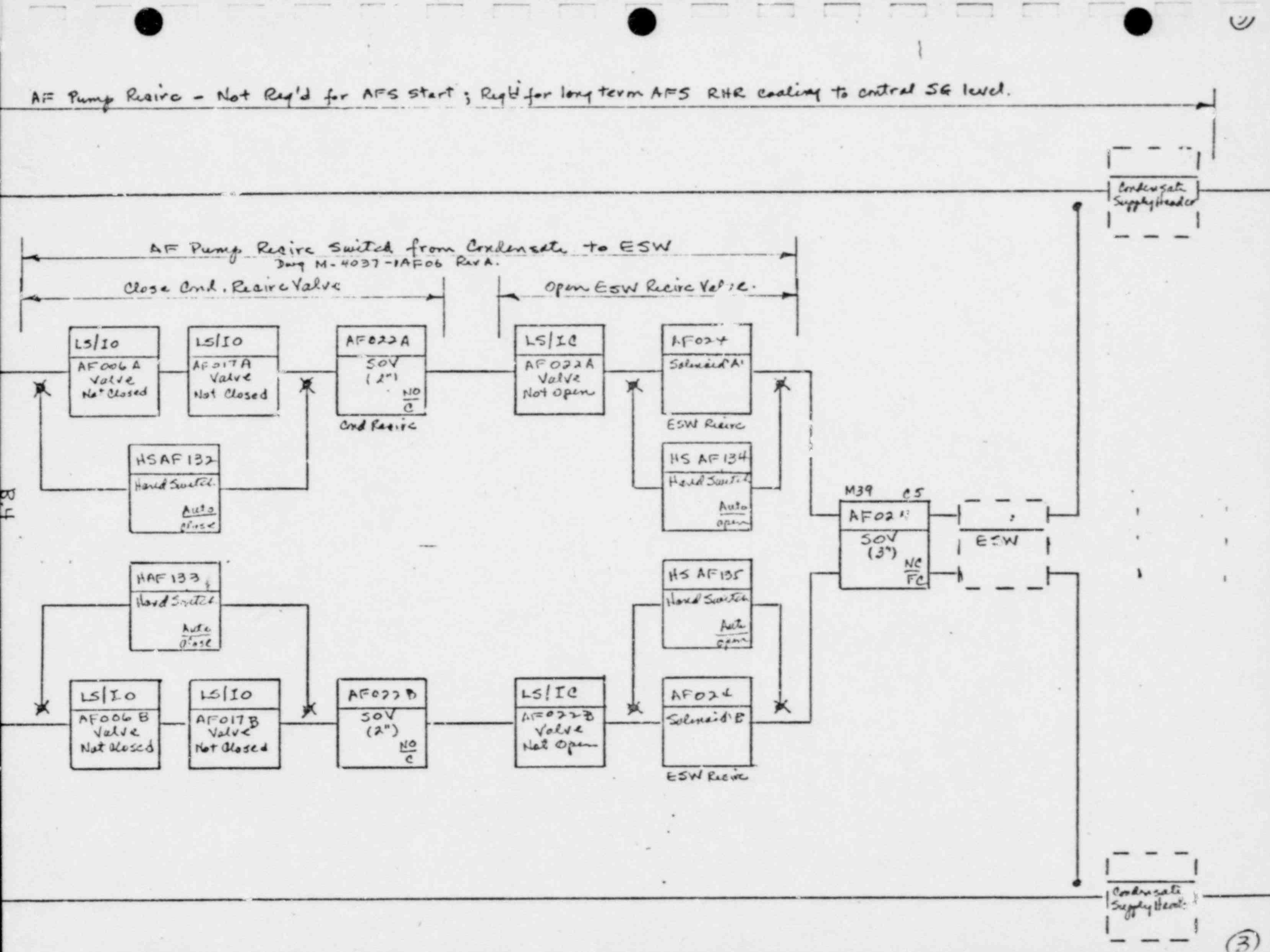
B.1

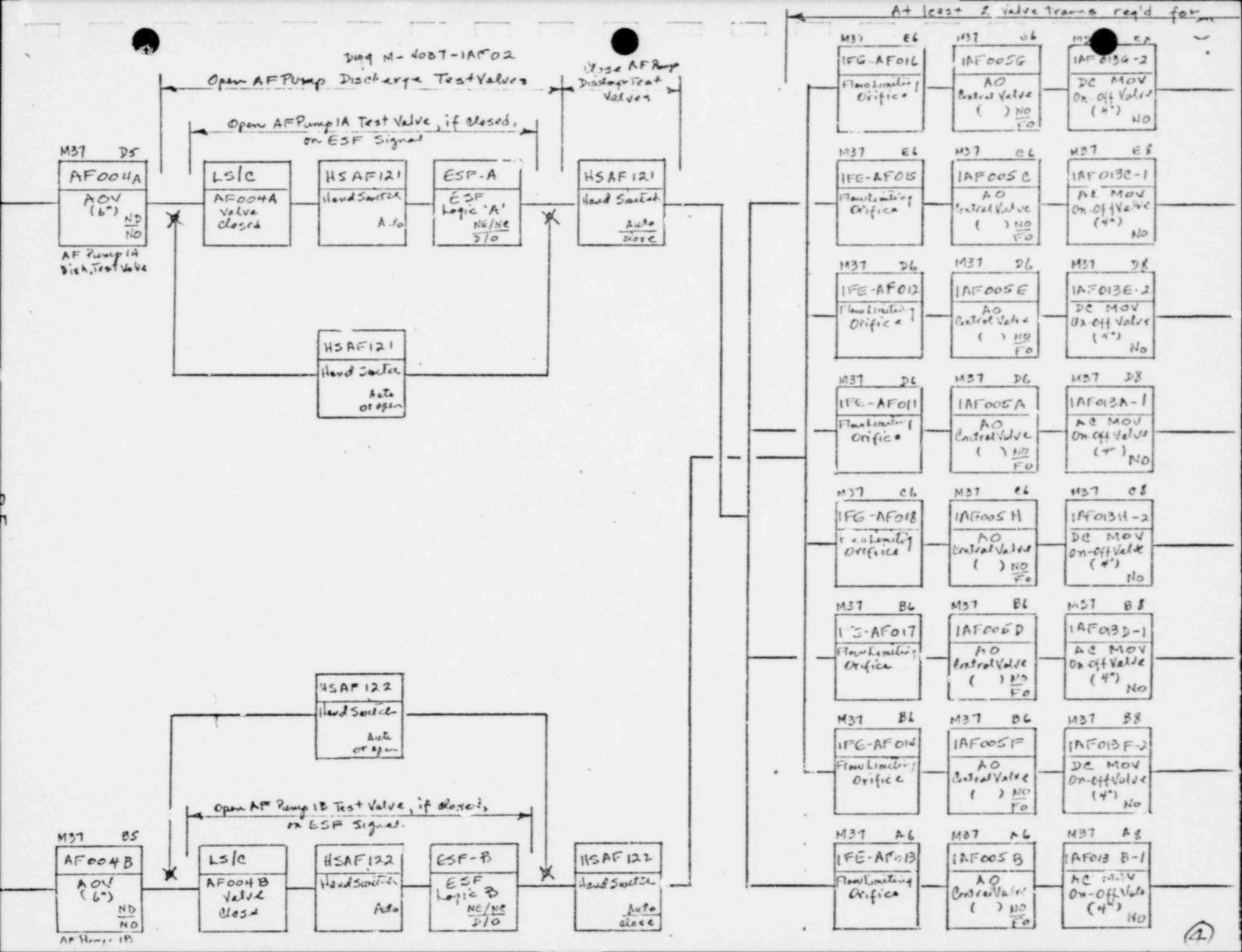

Normal Water Supply to AFS
M39 B3
1CD01T
Condensate Storage Tank 500,000 gl
M39 B3
1CD022
Manual On-off Valve
LO
NO
M39 E3
1CD193
Check Valve
NC
M39 D2
AF001A
Check Valve
NC
M39 D2
AF002A
Manual On-off Valve
NO
M39 B4
1CD091
Manual On-off Valve
NO
M39 C4
1CD149
Manual On-off Valve
NC
Backup Water to AFS - Dwg M-4037-1AF02 Rev. C
(Unavailable during loss of all AC power)
M39 E3
AF006A-1
MOV (AC)
NC
Motor Starter
M39 E3
AF017A-1
MOV (AC)
NC
Motor Starter
M39 E3
ESF-A
ESF Logic 'A'
NO
M39 E3
AF051
AF Pump 1A Suction Press. Lo
NO
M39 E3
AF005
Hand Switch (AF006A-1)
CR
AUTO
M39 E3
AF007
Hand Switch (AF017A-1)
CR
AUTO
M42-3 F4
ESW
ESW Train A
M42-3 C1
ESW
ESW Train B
M39 B3
AF006
Hand Switch (AF006B-2)
CR
AUTO
M39 B3
AF008
Hand Switch (AF017B-2)
CR
AUTO
M39 B3
AF006B-2
MOV (AC)
NC
Motor Starter
NO
ND
M39 C3
AF017B-2
MOV (AC)
NC
Motor Starter
NO
M39 B3
ESF-B
ESF Logic B
NO
M39 B4
AF055
AF Pump 1B Suction Press. Lo
NO
M39 B2
AF001B
Check Valve
NC
M39 B2
AF002B
Manual On-off Valve
NO

1

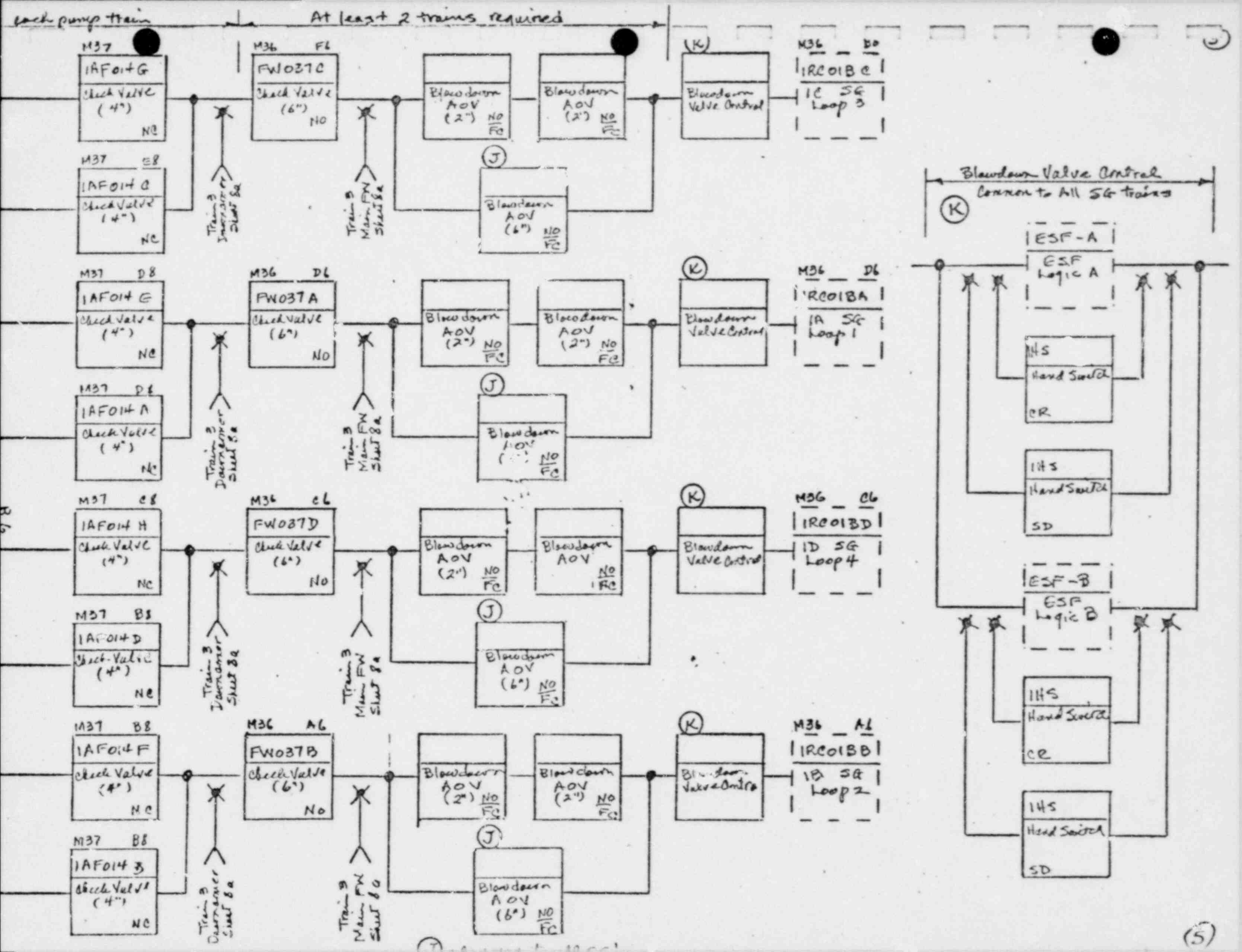

AFS Pump & Controls
Pump Start
Pump Stop
M37 D4 AF01PA-1 AF Motor Pump ND
Motor SWGR CUB #8
AP05E H 4160V ESF Bus #141
M37 D4 ESF-A ESF Logic A NO
M37 E4 1HSAF001 Hand Switch (A11-B11) CR OFF Close
M37 E4 1HSAF003 Hand Switch (R11-R12) SD Normal Start
M37 E3 AF051 AF Pump 1A Suction Press. Lo Lo NO
M37 E4 1HSAF001 Hand Switch (A11-B12) CR OFF Trip
M37 E4 1HSAF003 Hand Switch (L11-L12) SD Normal Stop
M37 D5 AF008A Check Valve (6") NC
M37 D5 AF014A Recirc Line Orifice
M39 C5 AF022A SOV (2") NO FO
Cond. Recirc
M39 D5 AF009A Manual On Off Valve (2") NO LO
M39 C5 AF026A Check Valve NC
AF01PB-2 AF Diesel Pump ND
M37 B4 1HSAF004 Hand Switch SD Normal Start
M37 B4 1HSAF002 Hand Switch CR OFF Close
M37 B4 ESF-B ESF Logic B NO
M37 B4 1HSAF004 Hand Switch SD Normal Stop
M37 B4 1HSAF002 Hand Switch CR OFF Trip
M37 B4 AF055 AF Pump 1B Suction Press. Lo Lo NO
M37 B5 AF003B Check Valve NC
M37 B5 AF014B Recirc Line Orifice
M39 C5 AF026B Check Valve NC
M39 B5 AF022B SOV (2") NO FO
M39 B5 AF009B Manual On Off Valve NO LO

②


AF Pump Recirc - Not Req'd for AFS start ; Req'd for long term AFS RHR cooling to control SG level.
AF Pump Recirc Switch from Condensate to ESW
Dwg M-4037-1AF06 Rev A.
Close Cnd. Recirc Valve
Open ESW Recirc Valve.
LS/IO
AF006 A Valve Not Closed
LS/IO
AF017A Valve Not Closed
AF022A
SOV (2")
NO/C
Cnd Recirc
LS/IC
AF022A Valve Not Open
AF024
Solenoid A'
ESW Recirc
HSAF132
Hand Switch
Auto/Close
HS AF134
Hand Switch
Auto/open
M39
C5
AF02
SOV (3")
NC/FC
ESW
Condensate Supply Header
HAF133
Hand Switch
Auto/Close
HS AF135
Hand Switch
Auto/open
LS/IO
AF006 B Valve Not Closed
LS/IO
AF017B Valve Not Closed
AF022B
SOV (2")
NO/C
LS/IC
AF022B Valve Not Open
AF024
Solenoid B
ESW Recirc
Condensate Supply Header
B.4
(3)

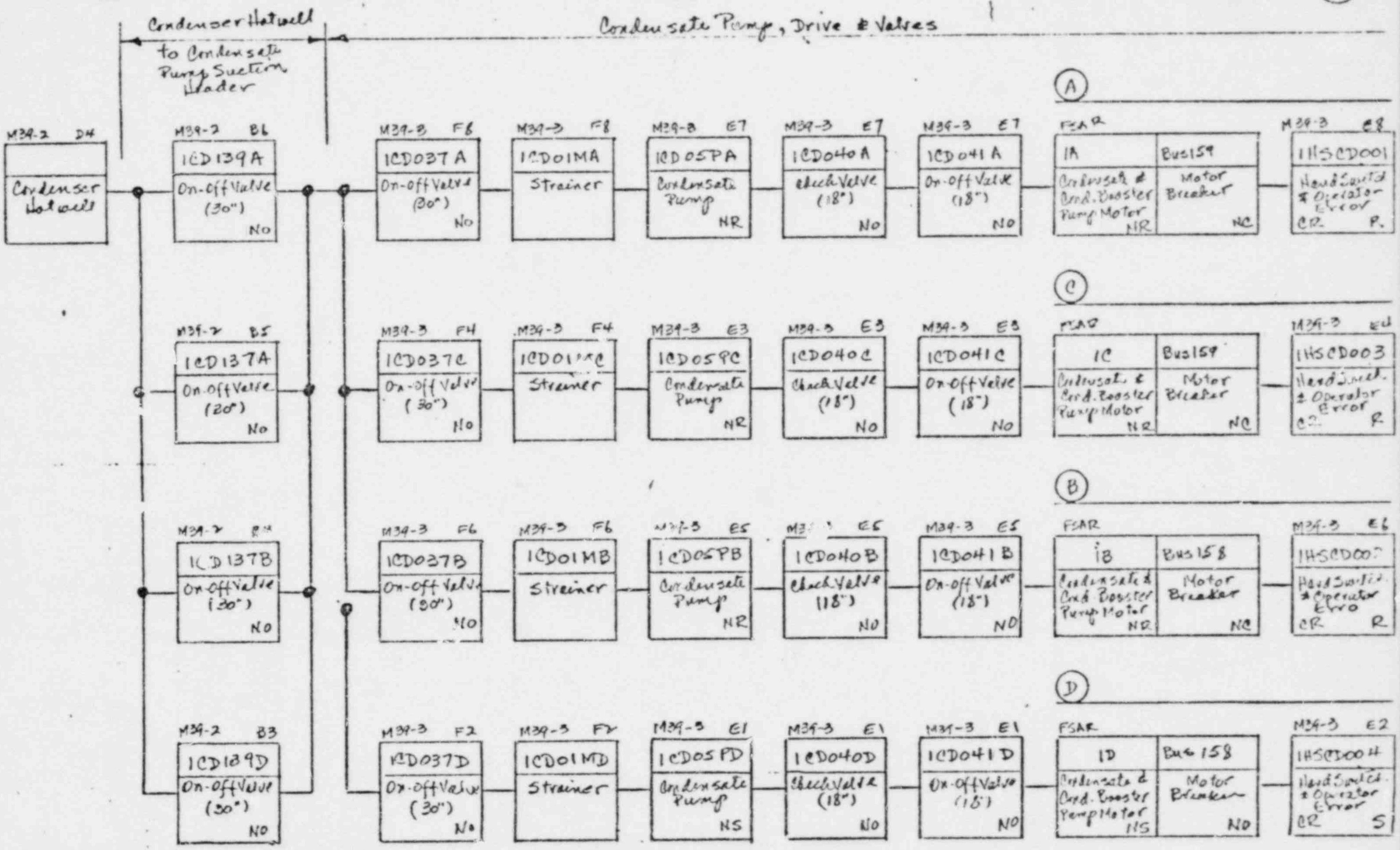


Dwg M-4037-1AF02
Open AF Pump Discharge Test Valves
Close AF Pump Discharge Test Valves
Open AF Pump 1A Test Valve, if closed, on ESF Signal
M37 D5
AF004A
AOV (6") ND/NO
AF Pump 1A Disch. Test Valve
LS/C
AF004A Valve Closed
HSAF121
Hand Switch
Auto
ESF-A
ESF Logic 'A' NE/NE D/O
HSAF121
Hand Switch
Auto close
HSAF121
Hand Switch
Auto or open
At least 2 valve trains req'd for
M37 E6
1FE-AF016
Flow Limiting Orifice
M37 E6
1AF005G
AO Control Valve ( ) NO/FO
M37 E8
1AF013G-2
DC MOV On-Off Valve (4") NO
M37 E6
1FE-AF015
Flow Limiting Orifice
M37 E6
1AF005C
AO Control Valve ( ) NO/FO
M37 E8
1AF013C-1
AC MOV On-Off Valve (4") NO
M37 D6
1FE-AF012
Flow Limiting Orifice
M37 D6
1AF005E
AO Control Valve ( ) NO/FO
M37 D8
1AF013E-2
DC MOV On-Off Valve (4") NO
M37 D6
1FE-AF011
Flow Limiting Orifice
M37 D6
1AF005A
AO Control Valve ( ) NO/FO
M37 D8
1AF013A-1
AC MOV On-Off Valve (4") NO
M37 C6
1FG-AF018
Flow Limiting Orifice
M37 C6
1AF005H
AO Control Valve ( ) NO/FO
M37 C8
1AF013H-2
DC MOV On-Off Valve (4") NO
M37 B6
AF017
Flow Limiting Orifice
M37 B6
1AF005D
AO Control Valve ( ) NO/FO
M37 B8
1AF013D-1
AC MOV On-Off Valve (4") NO
M37 B6
1FE-AF014
Flow Limiting Orifice
M37 B6
1AF005F
AO Control Valve ( ) NO/FO
M37 B8
1AF013F-2
DC MOV On-Off Valve (4") NO
M37 A6
1FE-AF013
Flow Limiting Orifice
M37 A6
1AF005B
AO Control Valve ( ) NO/FO
M37 A8
1AF013B-1
AC MOV On-Off Valve (4") NO
HSAF122
Hand Switch
Auto or open
Open AF Pump 1B Test Valve, if closed, on ESF Signal.
M37 B5
AF004B
AOV (6") ND/NO
AF Pump 1B
LS/C
AF004B Valve Close
HSAF122
Hand Switch
Auto
ESF-B
ESF Logic B NE/NE D/O
HSAF122
Hand Switch
Auto close
4

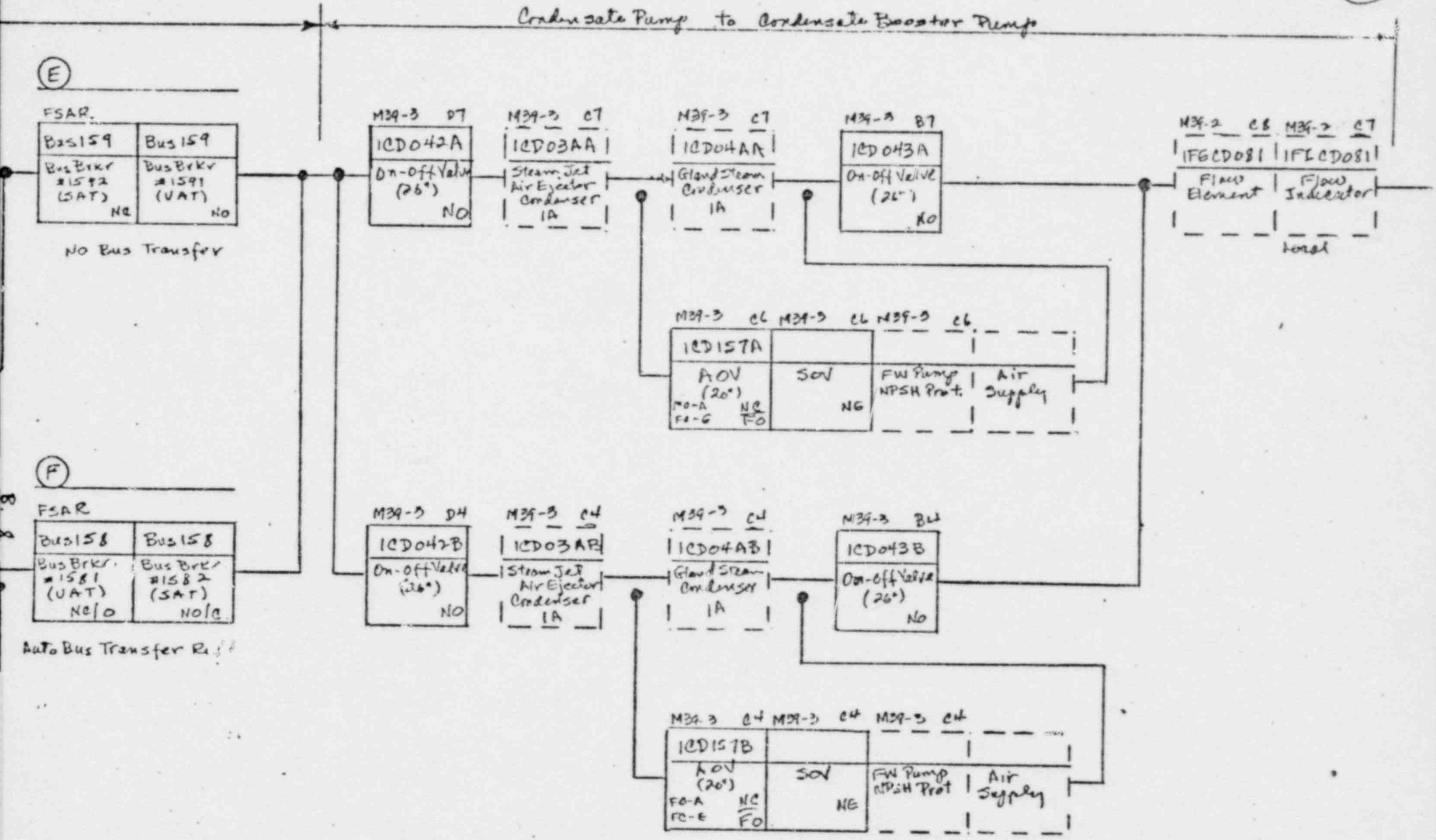


each pump train
At least 2 trains required
M37
1AF014G
Check Valve (4")
NC
M36 F6
FW037C
Check Valve (6")
NO
Blowdown AOV (2") NO FC
Blowdown AOV (2") NO FC
Blowdown Valve Control
M36 B6
1RC01BC
1C SG Loop 3
M37 E8
1AF014C
Check Valve (4")
NC
Train 3 Interseror Sheet 8a
Train 3 Main FW Sheet 8a
J
Blowdown AOV (6") NO FC
K
M37 D8
1AF014E
Check Valve (4")
NC
M36 D6
FW037A
Check Valve (6")
NO
Blowdown AOV (2") NO FC
Blowdown AOV (2") NO FC
Blowdown Valve Control
M36 D6
1RC01BA
1A SG Loop 1
M37 D8
1AF014A
Check Valve (4")
NC
Train 3 Daerator Sheet 8a
Train 3 Main FW Sheet 8a
Blowdown AOV NO FC
M37 C8
1AF014H
Check Valve (4")
NC
M36 C6
FW037D
Check Valve (6")
NO
Blowdown AOV (2") NO FC
Blowdown AOV NO FC
Blowdown Valve Control
M36 C6
1RC01BD
1D SG Loop 4
M37 B8
1AF014D
Check Valve (4")
NC
Blowdown AOV (6") NO FC
M37 B8
1AF014F
Check Valve (4")
NC
M36 A6
FW037B
Check Valve (6")
NO
Blowdown AOV (2") NO FC
Blowdown AOV (2") NO FC
Blowdown Valve Control
M36 A6
1RC01BB
1B SG Loop 2
M37 B8
1AF014B
Check Valve (4")
NC
Blowdown AOV (6") NO FC
Blowdown Valve Control
Common to All SG trains
ESF-A
ESF Logic A
1HS
Hand Switch
CR
1HS
Hand Switch
SD
ESF-B
ESF Logic B
1HS
Hand Switch
CR
1HS
Hand Switch
SD
(5)

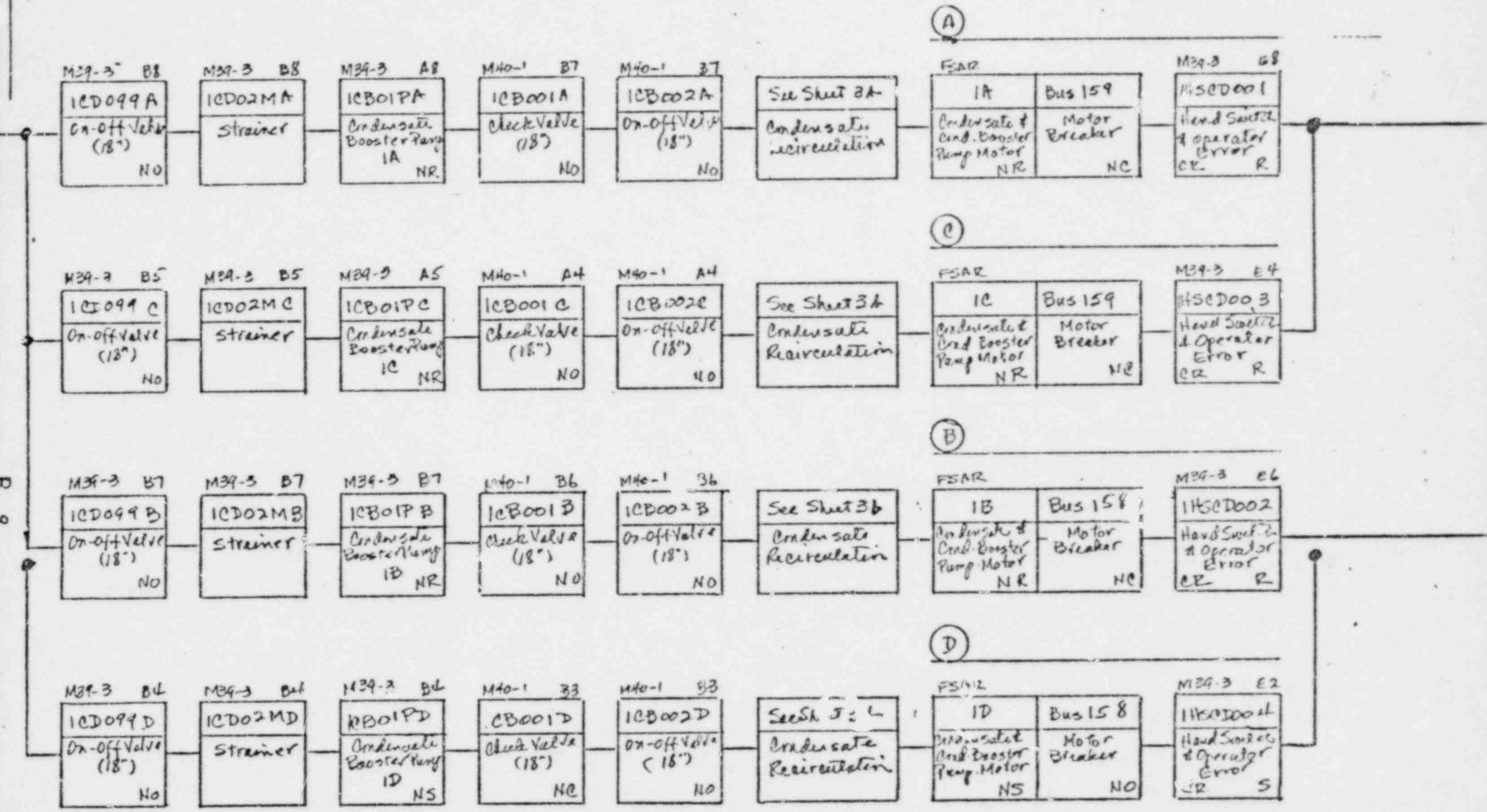
(1a)


Condenser Hotwell to Condensate Pump Suction Header
Condensate Pump, Drive & Valves
M39-2 D4
Condenser Hotwell
M39-2 B6
1CD139A
On-Off Valve (30")
NO
M39-2 B5
1CD137A
On-Off Valve (20")
NO
1CD137B
On-Off Valve (30")
NO
M39-2 B3
1CD139D
On-Off Valve (30")
NO
A
M39-3 F8
1CD037A
On-Off Valve (30")
NO
M39-3 F8
1CD01MA
Strainer
M39-3 E7
1CD05PA
Condensate Pump
NR
M39-3 E7
1CD040A
Check Valve (18")
NO
M39-3 E7
1CD041A
On-Off Valve (18")
NO
FSAR
1A
Condensate & Cond. Booster Pump Motor
NR
Bus 159
Motor Breaker
NC
M39-3 E8
1HSCD001
Hand Switch & Operator Error
CR
R
C
M39-3 F4
1CD037C
On-Off Valve (30")
NO
M39-3 F4
Strainer
M39-3 E3
1CD05PC
Condensate Pump
NR
M39-3 E3
1CD040C
Check Valve (18")
NO
M39-3 E3
1CD041C
On-Off Valve (18")
NO
FSAR
1C
Condensate & Cond. Booster Pump Motor
NR
Bus 159
Motor Breaker
NC
1HSCD003
Hand Switch & Operator Error
R
B
M39-3 F6
1CD037B
On-Off Valve (30")
NO
M39-3 F6
1CD01MB
Strainer
E5
1CD05PB
Condensate Pump
NR
E5
1CD040B
Check Valve (18")
NO
M39-3 E5
1CD041B
On-Off Valve (18")
NO
FSAR
1B
Condensate & Cond. Booster Pump Motor
NR
Bus 158
Motor Breaker
NC
M39-3 E6
Hand Switch & Operator Error
CR
R
D
M39-3 F2
1CD037D
On-Off Valve (30")
NO
M39-3 F2
1CD01MD
Strainer
M39-3 E1
1CD05PD
Condensate Pump
NS
M39-3 E1
1CD040D
Check Valve (18")
NO
M39-3 E1
1CD041D
On-Off Valve (18")
NO
FSAR
1D
Condensate & Cond. Booster Pump Motor
NS
Bus 158
Motor Breaker
NO
M39-3 E2
1HSCD004
Hand Switch & Operator Error
CR
S

(1a)


Condensate Pump to Condensate Booster Pump
E
FSAR
Bus159
Bus Brkr #1592 (SAT) NC
Bus 159
Bus Brkr #1591 (UAT) NO
No Bus Transfer
M39-3 D7
1CD042A
On-Off Valve (26")
NO
M39-3 C7
1CD03AA
Steam Jet Air Ejector Condenser 1A
M39-3 C7
1CD04AA
Gland Steam Condenser 1A
M39-3 B7
1CD043A
On-Off Valve (26")
NO
M39-2 C8
1FECD081
Flow Element
M39-2 C7
1FICD081
Flow Indicator
Local
M39-3 C6
1CD157A
AOV (20")
FO-A
FC-G
NC
FO
M39-3 C6
SOV
NC
M39-3 C6
FW Pump NPSH Prot.
Air Supply
F
FSAR
Bus158
Bus Brkr. #1581 (UAT) NC/O
Bus158
Bus Brkr #1582 (SAT) NO/C
Auto Bus Transfer
M39-3 D4
1CD042B
On-Off Valve (26")
NO
M39-3 C4
1CD03AB
Steam Jet Air Ejector Condenser 1A
M39-3 C4
1CD04AB
Gland Steam Condenser 1A
M39-3 B4
1CD043B
On-Off Valve (26")
NO
M39-3 C4
1CD157B
AOV (20")
FO-A
FC-E
NC
FO
M39-3 C4
SOV
NC
M39-3 C4
FW Pump NPSH Prot
Air Supply

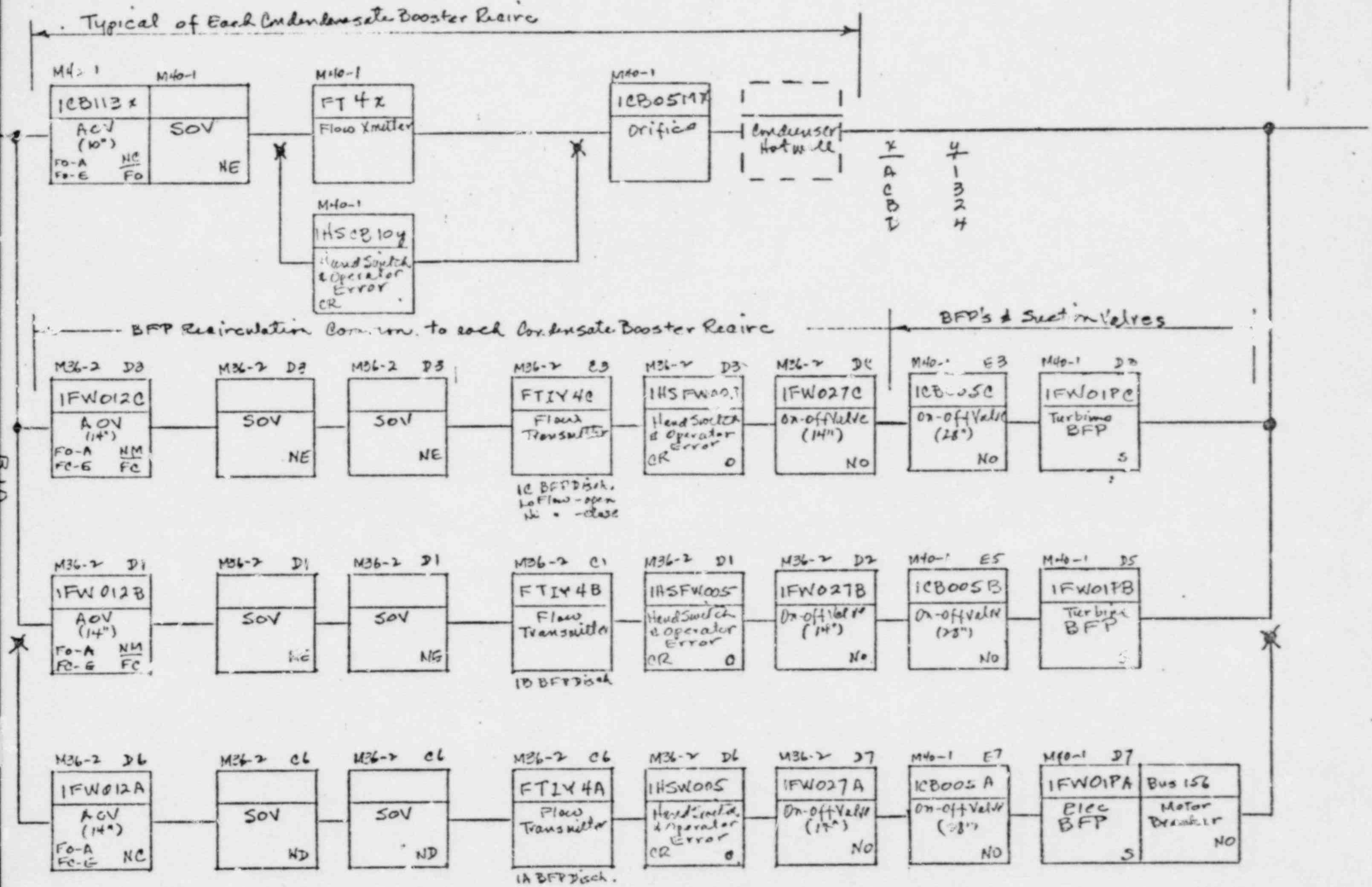
Condensate Booster Pumps, Drive & Valves



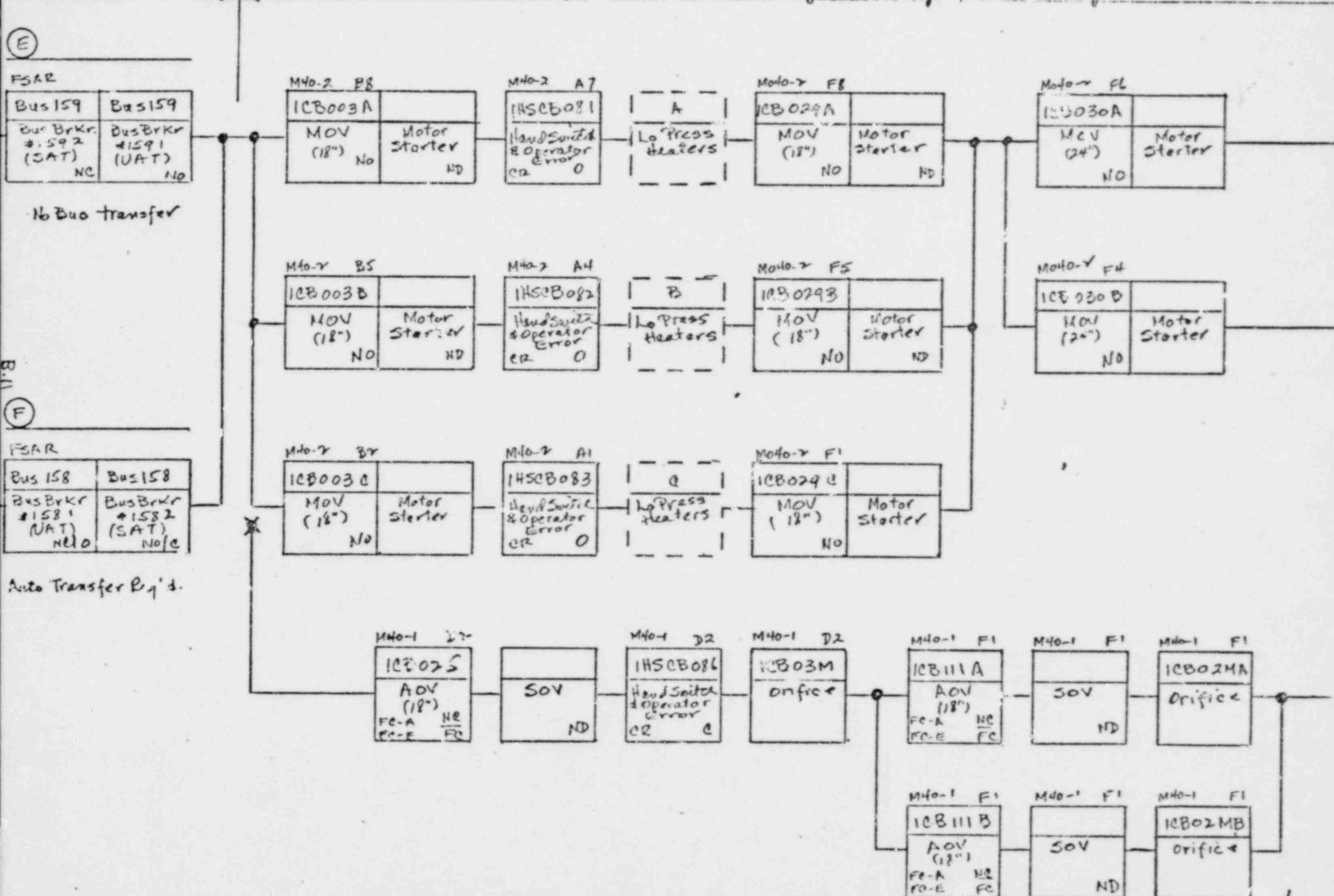
B. 9

(3a)

Condensate Recirculation

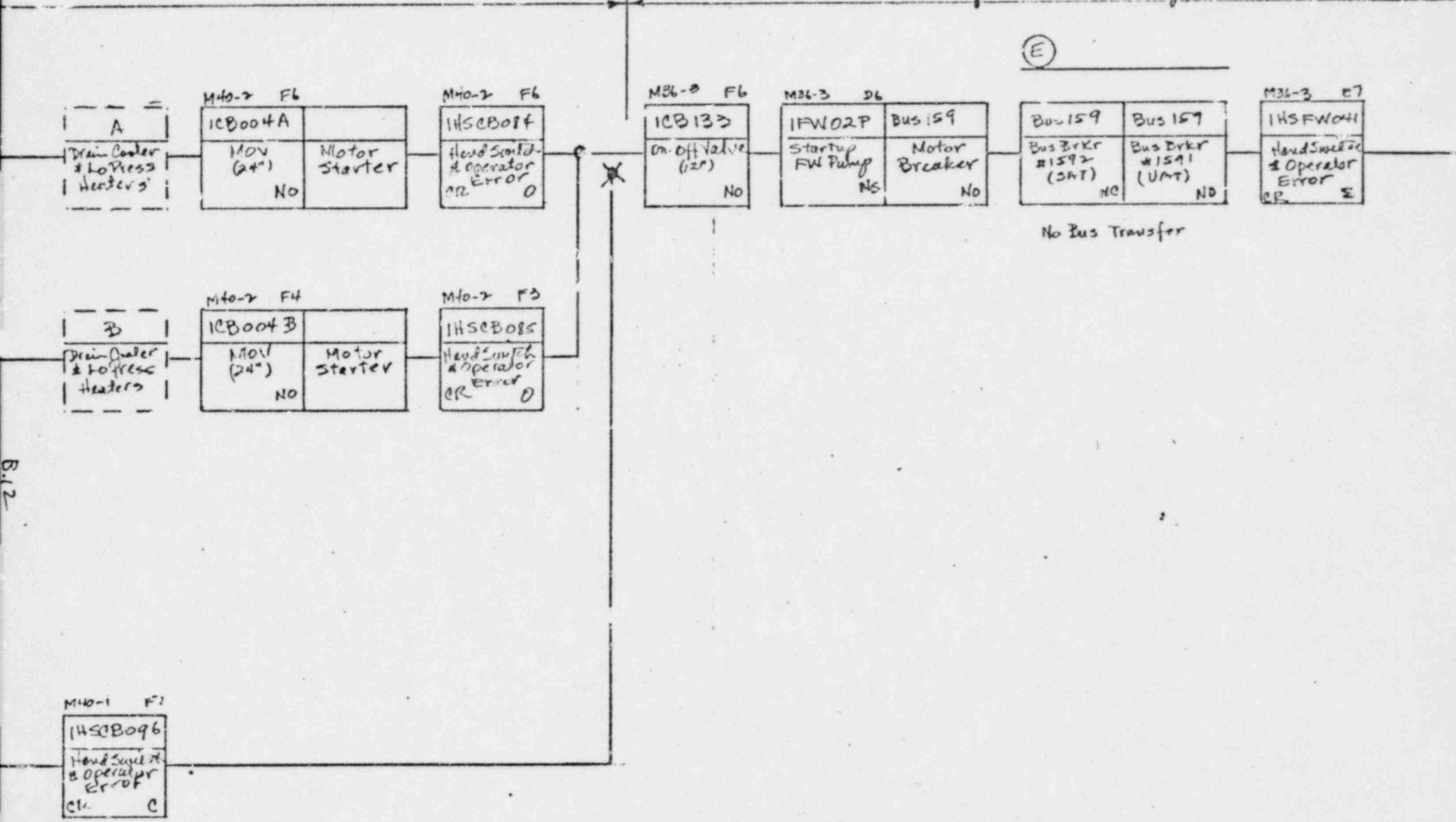


Condensate Booster Pump to Startup Feedwater Pump



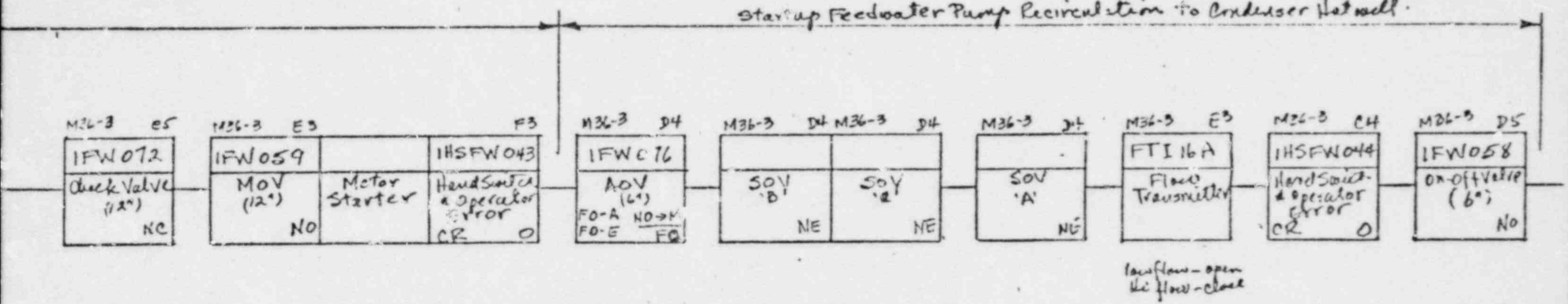
B.11

Startup Feedwater Pump, Drive & Valves

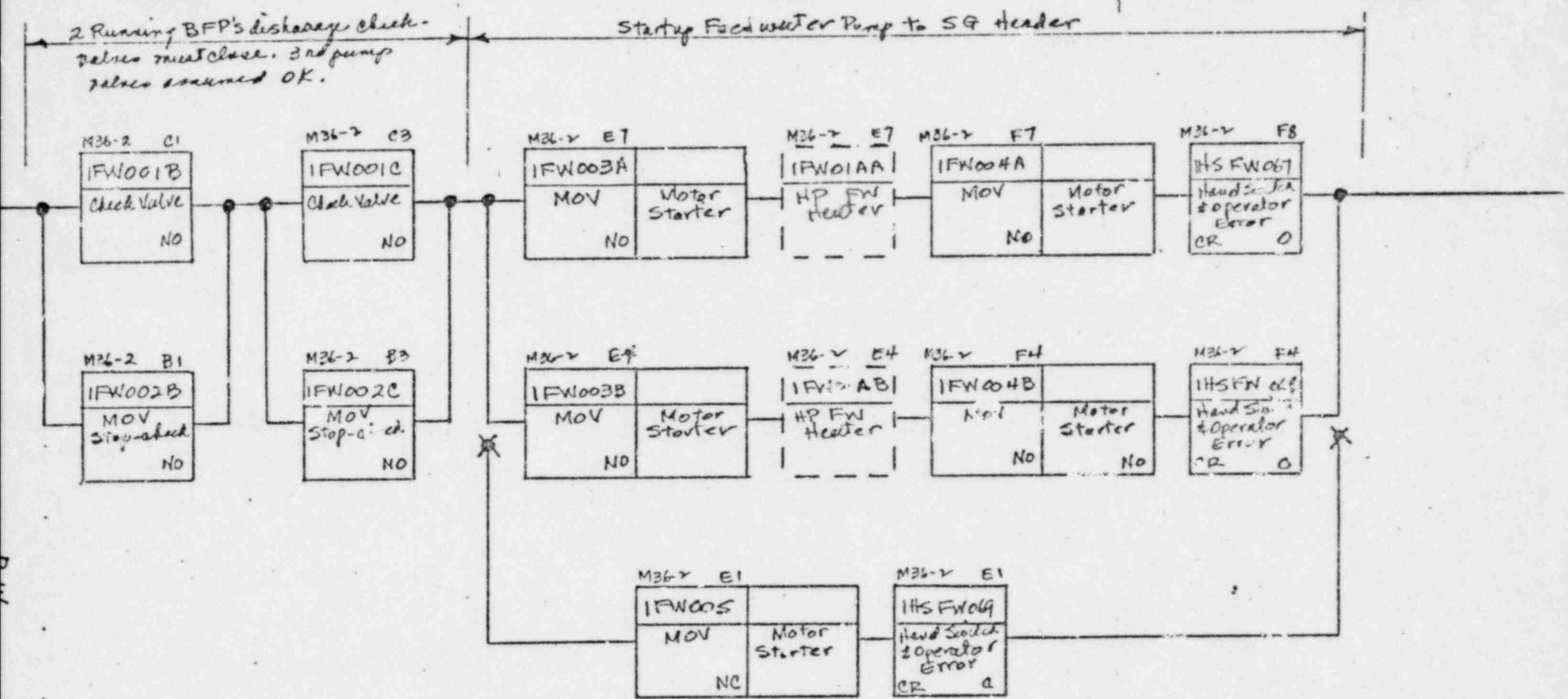


B.12

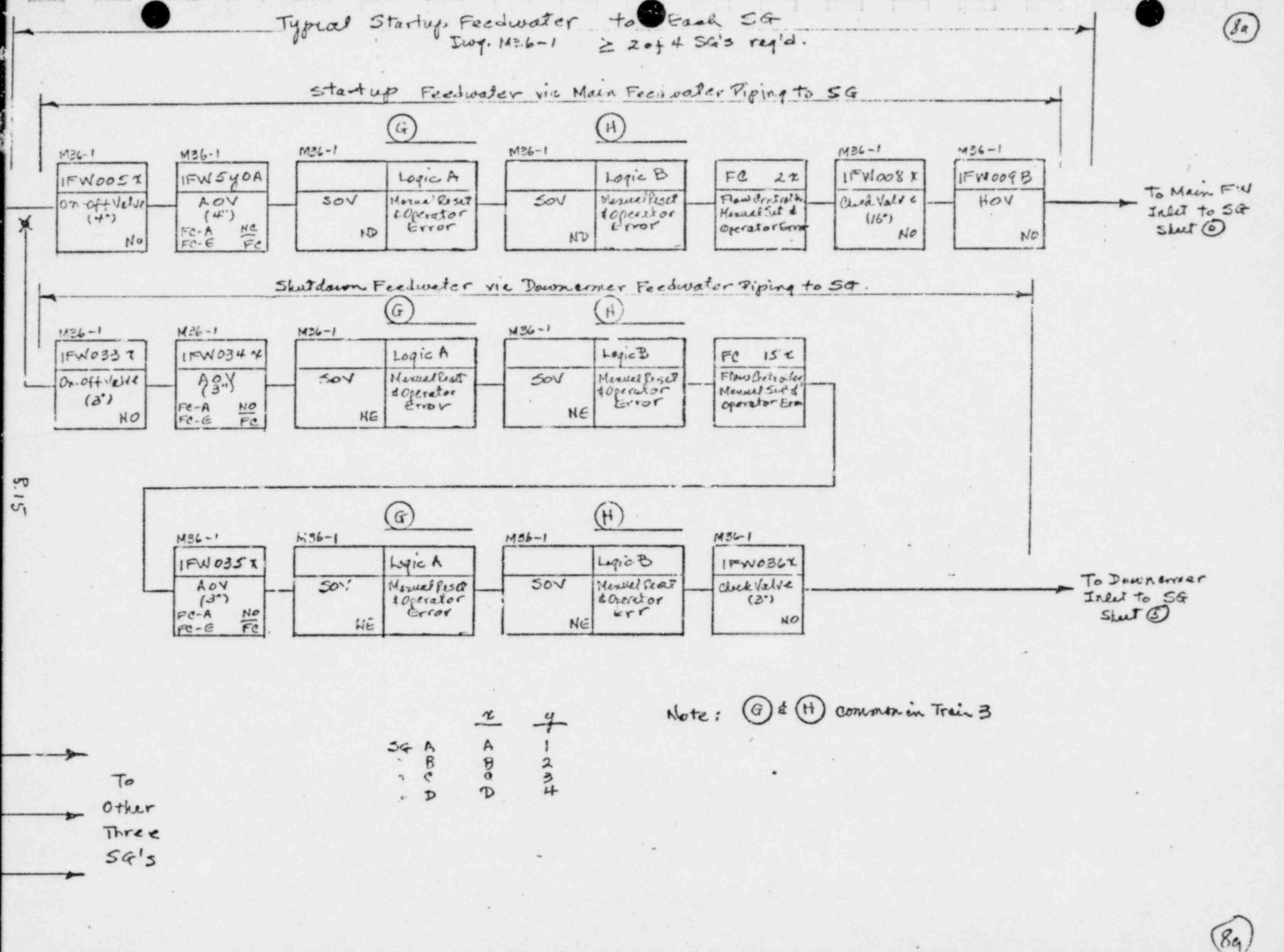
(6a)


Startup Feedwater Pump Recirculation to Condenser Hotwell.
M36-3 E5
1FW072
Check Valve (12")
NC
M36-3 E3
1FW059
MOV (12")
NO
Motor Starter
F3
1HSFW043
Hand Switch & Operator Error
CR O
M36-3 D4
1FWC76
AOV (6")
FO-A NO→K
FO-E FQ
M36-3 D4
SOV 'B'
NE
M36-3 D4
SOV '2'
NE
M36-3 D4
SOV 'A'
NE
M36-3 E3
FT116A
Flow Transmitter
low flow - open
hi flow - close
M36-3 C4
1HSFW044
Hand Switch & Operator Error
CR O
M36-3 D5
1FW058
On-Off Valve (6")
NO

(6a)



2 Running BFP's discharge check-valves must close. 3rd pump valves assumed OK.
Startup Feedwater Pump to SG Header
M36-2 C1
1FW001B
Check Valve
NO
M36-2 C3
1FW001C
Check Valve
NO
M36-2 B1
1FW002B
MOV
Stop-check
NO
M36-2 B3
1FW002C
MOV
NO
M36-2 E7
1FW003A
MOV
Motor Starter
NO
M36-2 E7
1FW01AA
HP FW Heater
M36-2 F7
1FW004A
MOV
Motor Starter
NO
M36-2 F8
1HS FW067
Hand Switch & Operator Error
CR
O
M36-2 E4
1FW003B
MOV
Motor Starter
NO
M36-2 E4
HP FW Heater
M36-2 F4
1FW004B
Motor Starter
NO
NO
M36-2 F4
Hand Switch & Operator Error
O
M36-2 E1
1FW005
MOV
Motor Starter
NC
M36-2 E1
1HS FW069
Hand Switch & Operator Error
CR



		x	y
SG	A	A	1
	B	B	2
	C	C	3
	D	D	4

Note: G & H common in Train 3

8.15

APPENDIX C

MASTER FAULT TREE

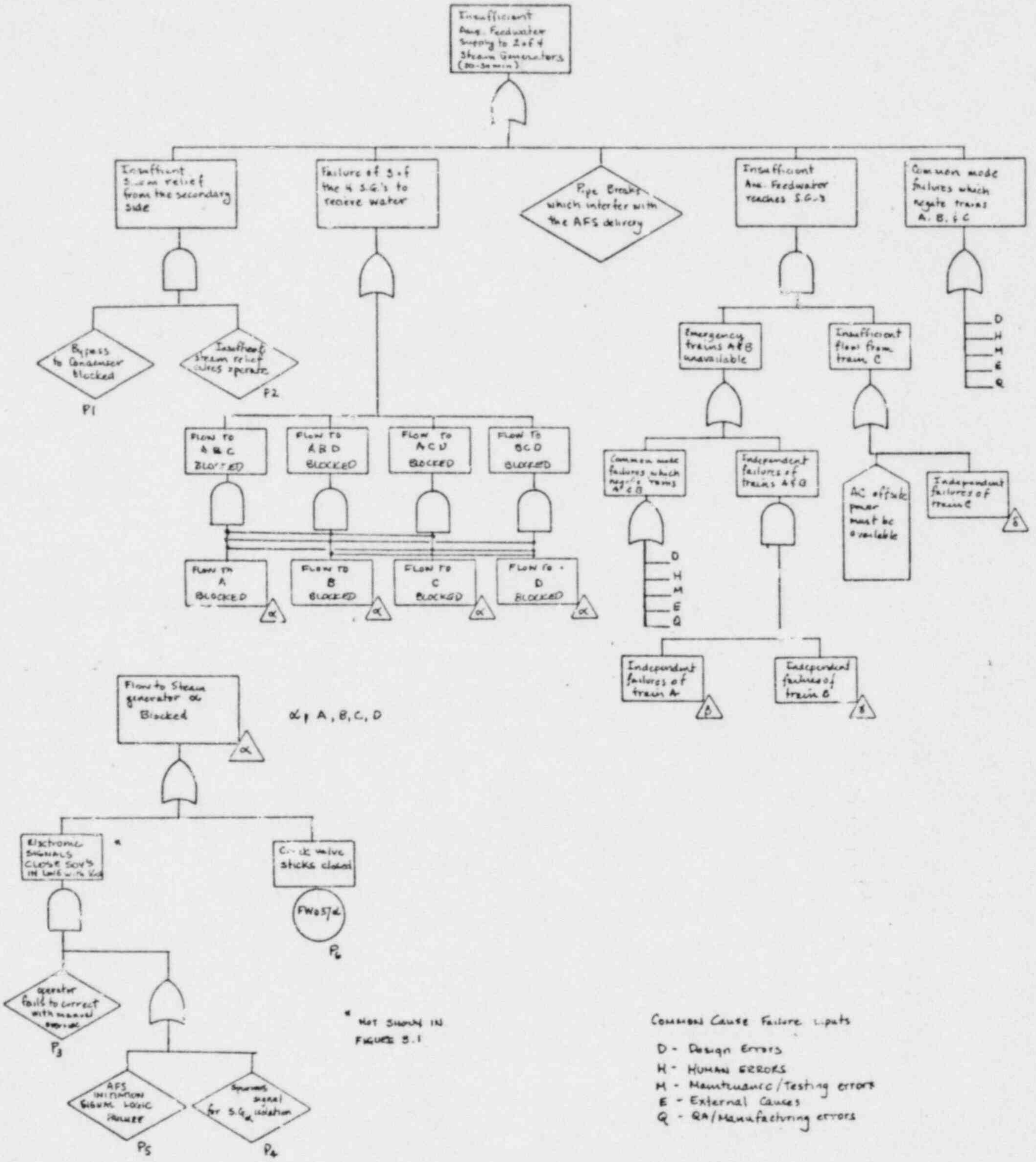


Figure 9-1 MASTER FAULT TREE PAGE 1 OF 5

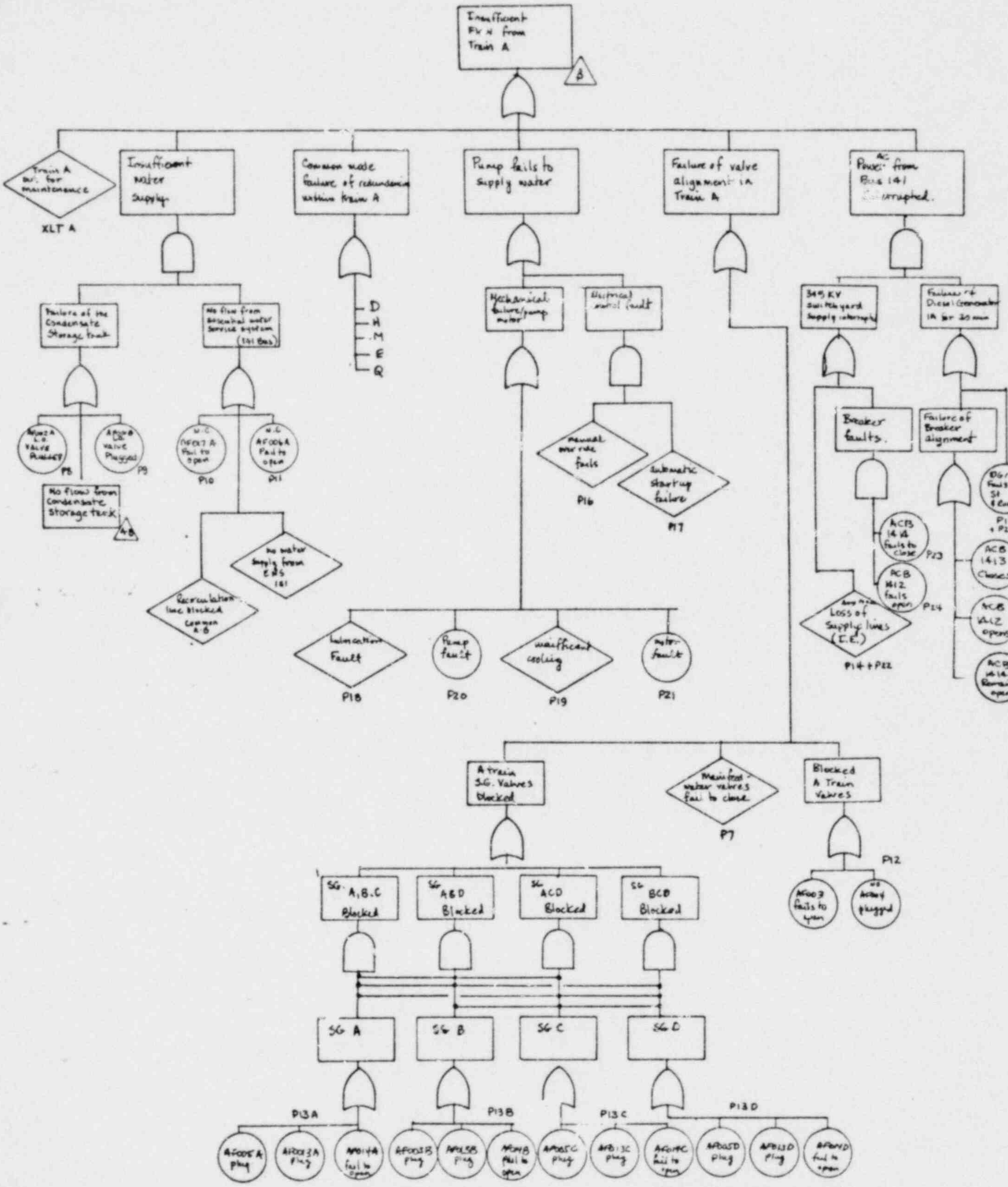


FIGURE C-1 MASTER FAULT TREE PAGE 2 OF 5

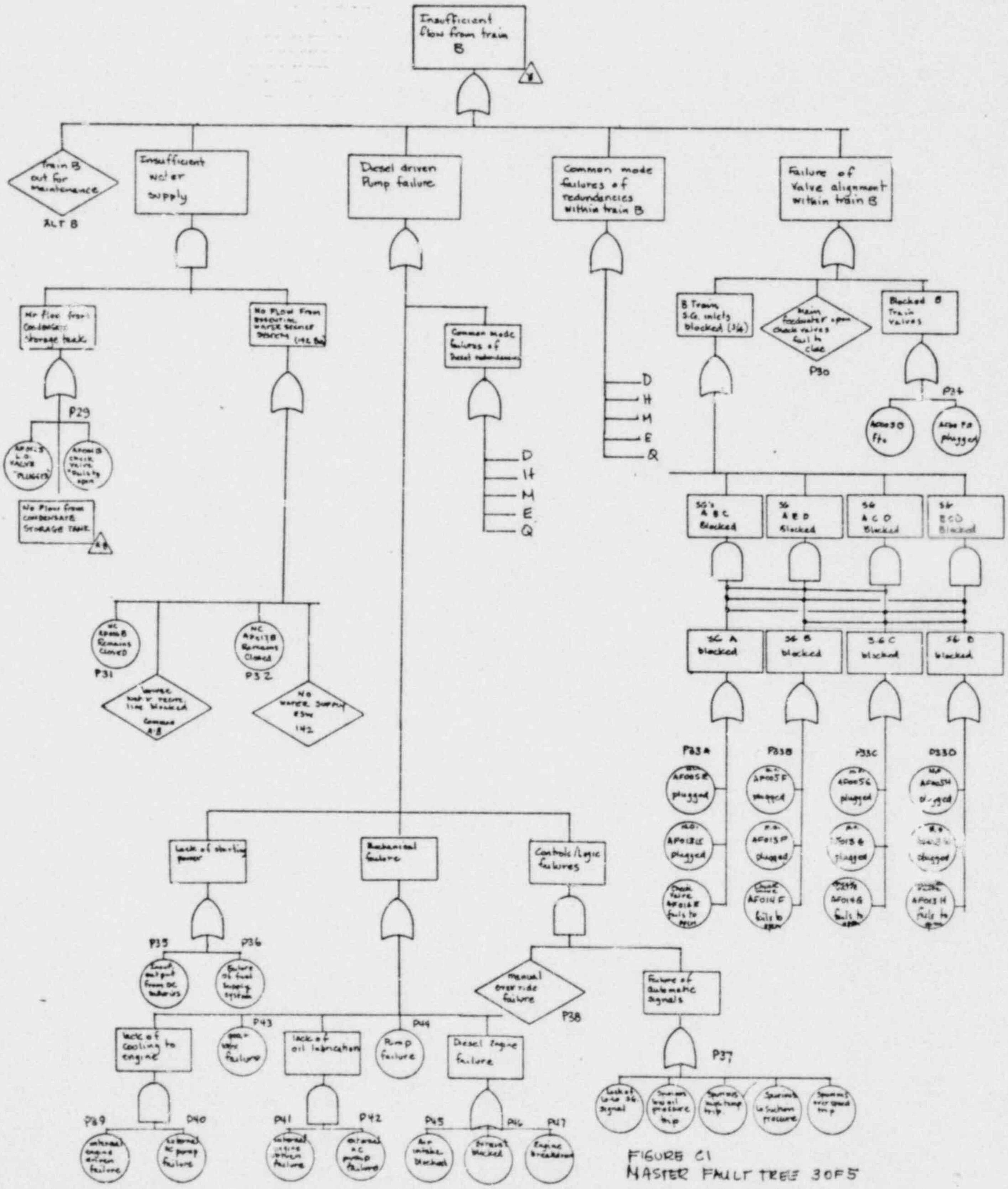


FIGURE C1
MASTER FAULT TREE 3OF5

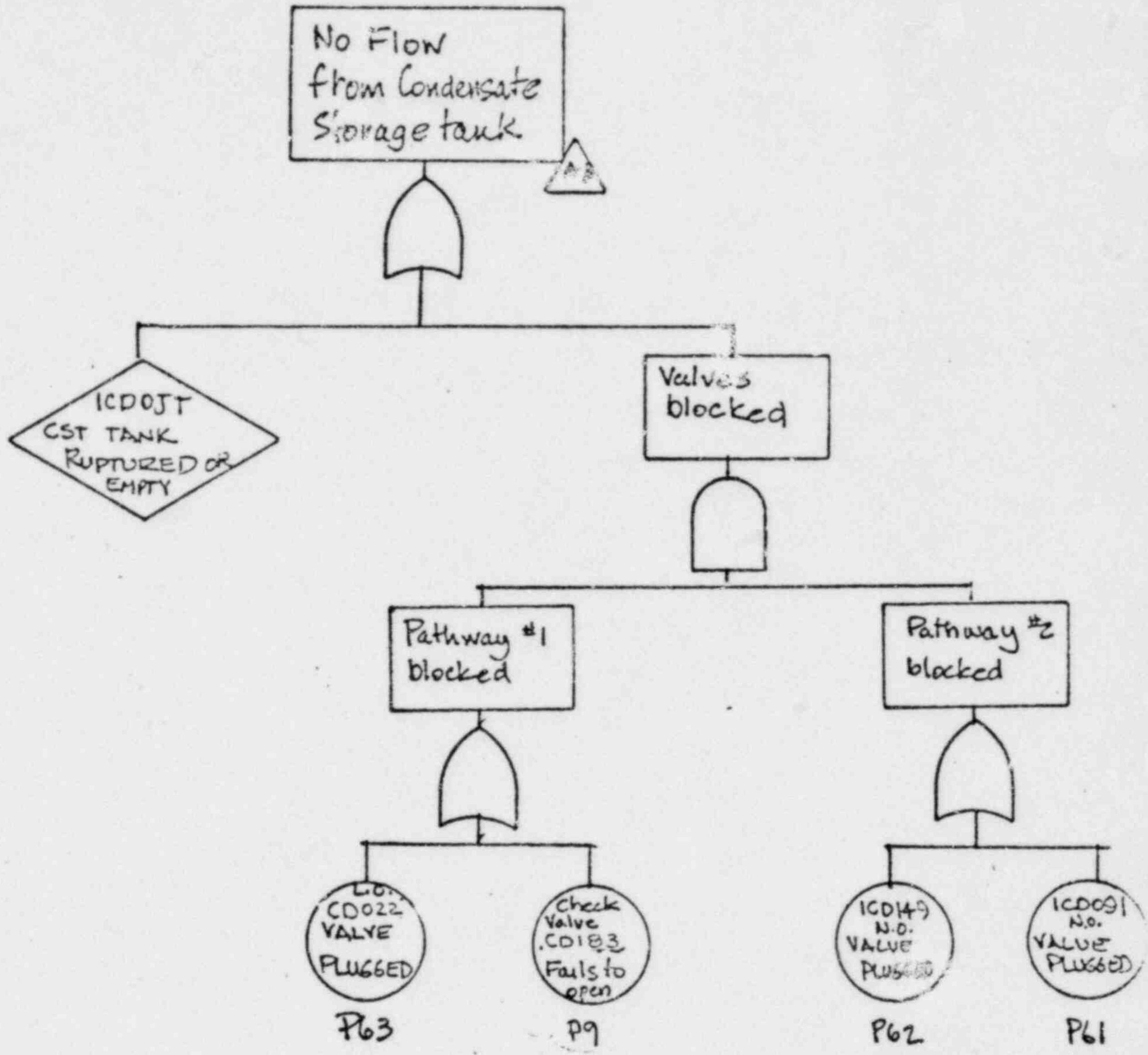


FIGURE C-1 MASTER FAULT TREE PAGE 4 OF 5

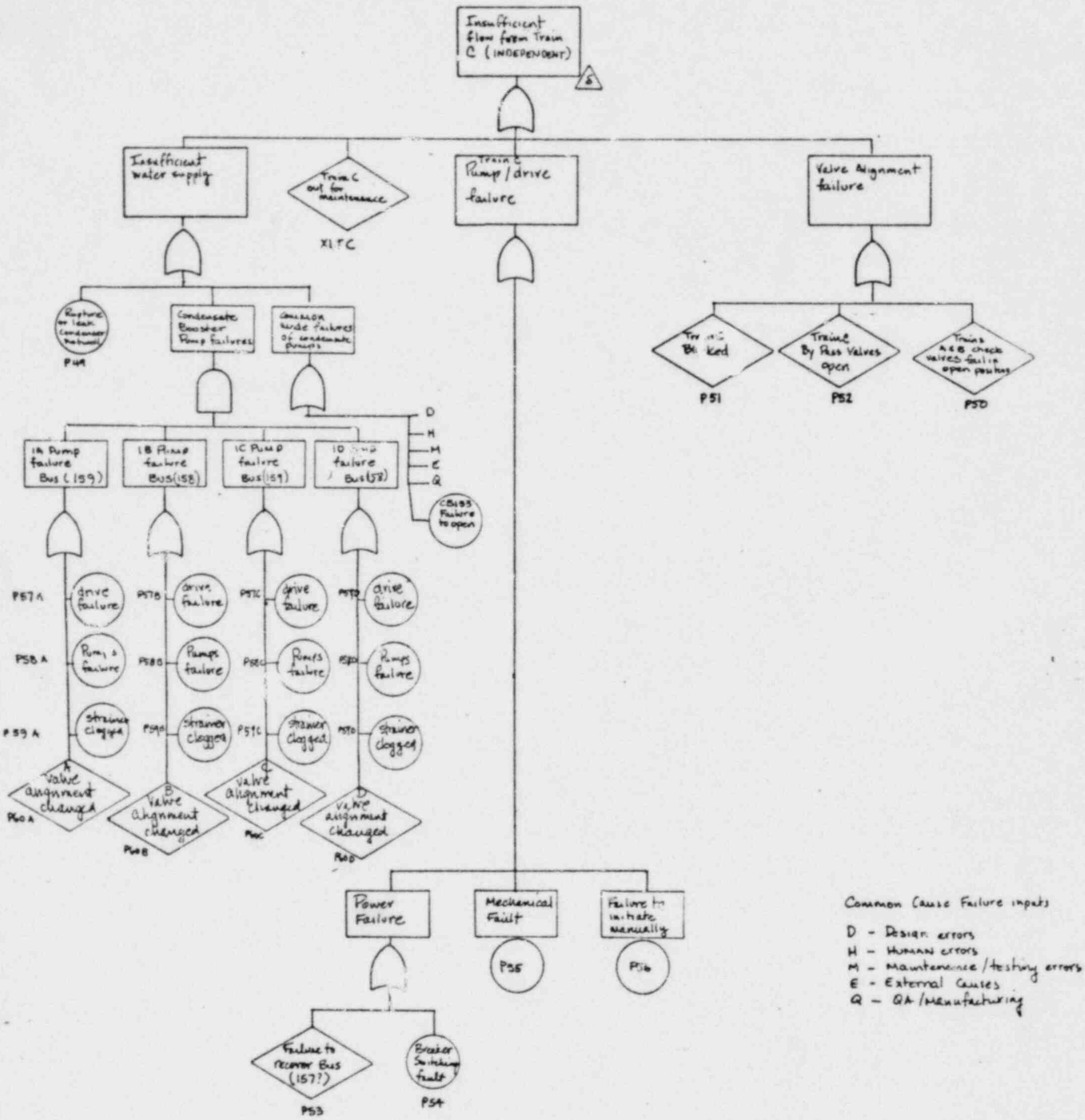
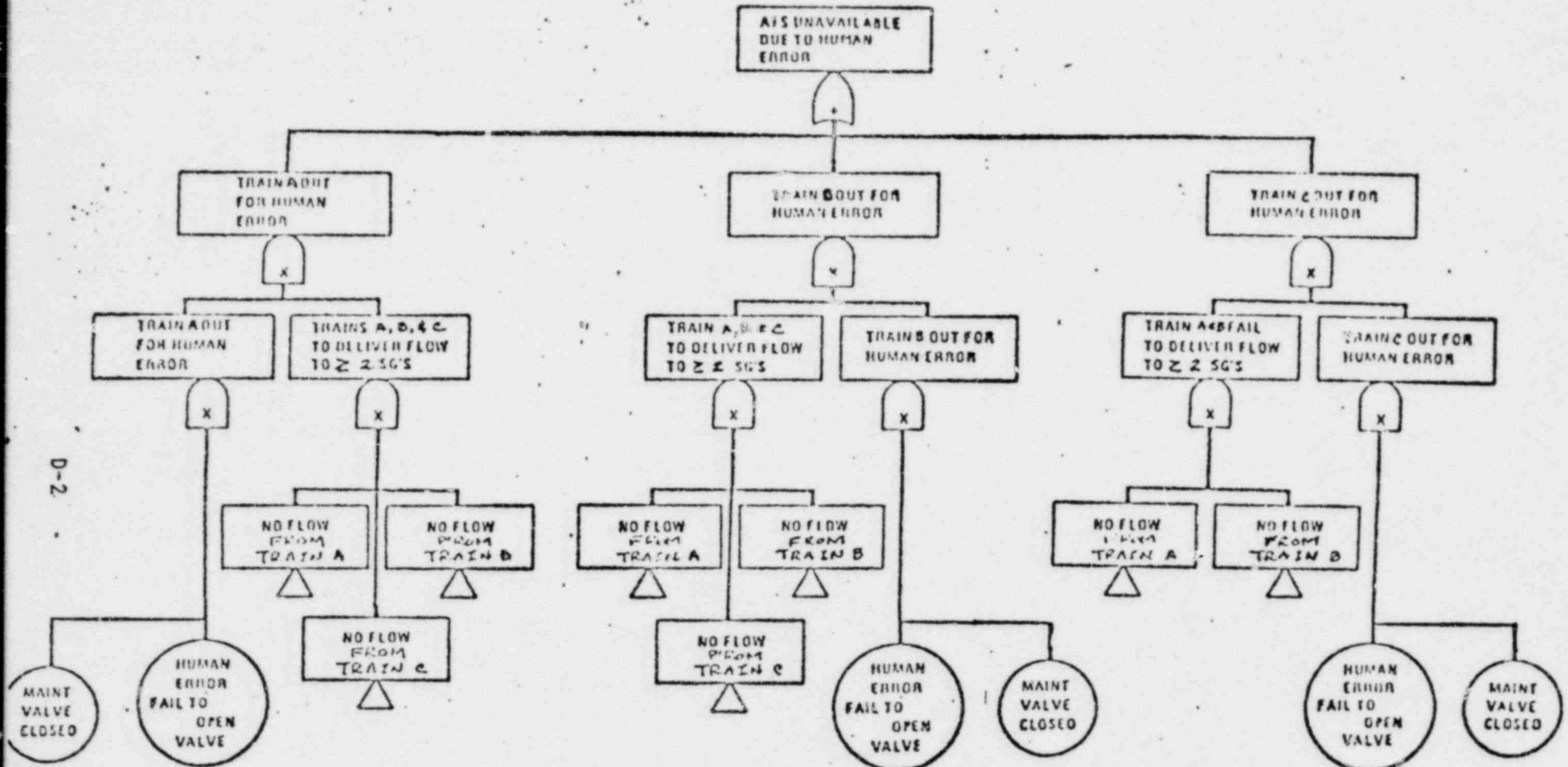


FIGURE C-1 MASTER FAULT TREE PAGE 5 OF 5

APPENDIX D

HUMAN ERROR FAULT TREE



* Pump (mechanical) fail to start

Fig. D-1 HUMAN ERROR FAULT TREE

APPENDIX E

TESTING AND MAINTENANCE FAULT TREES

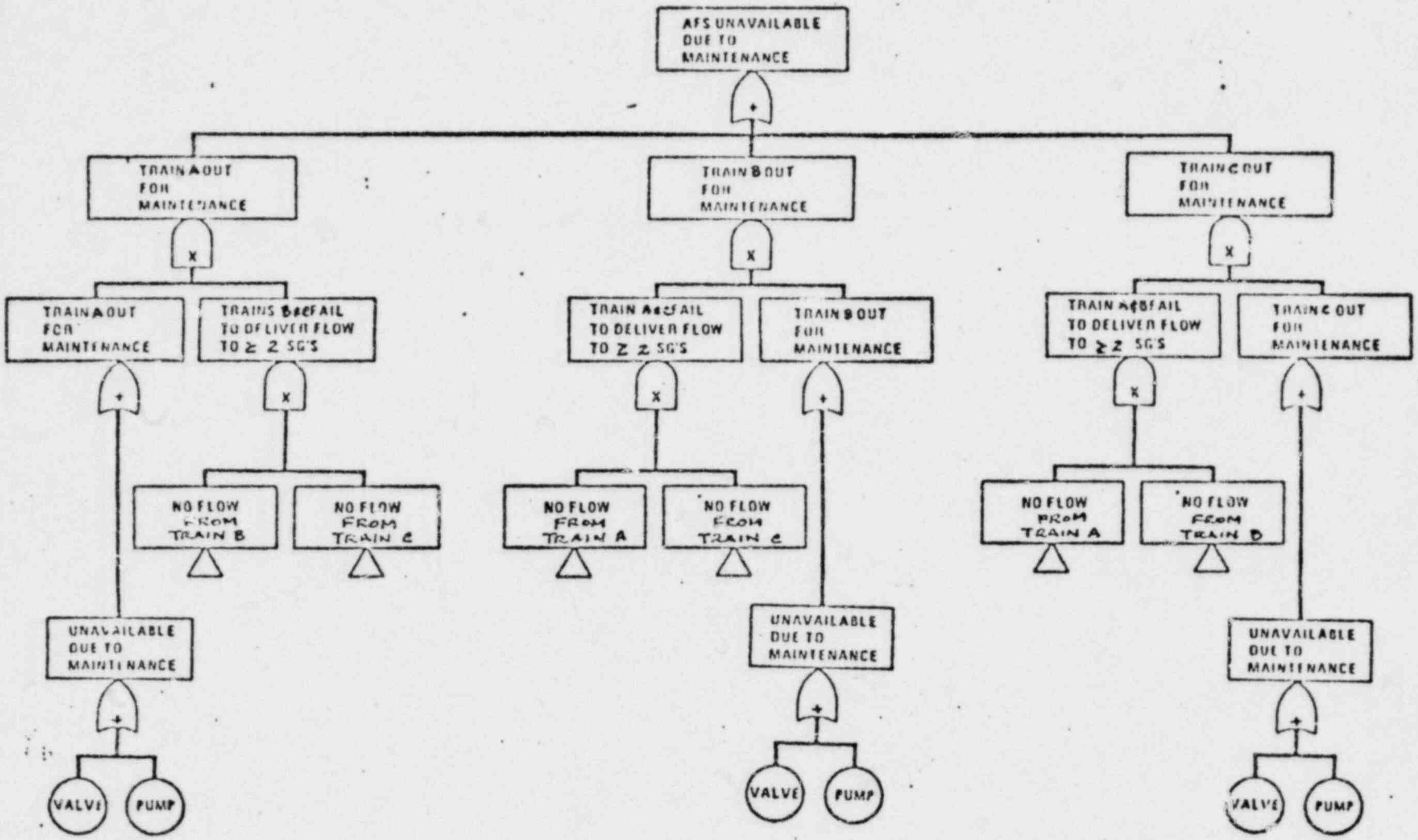
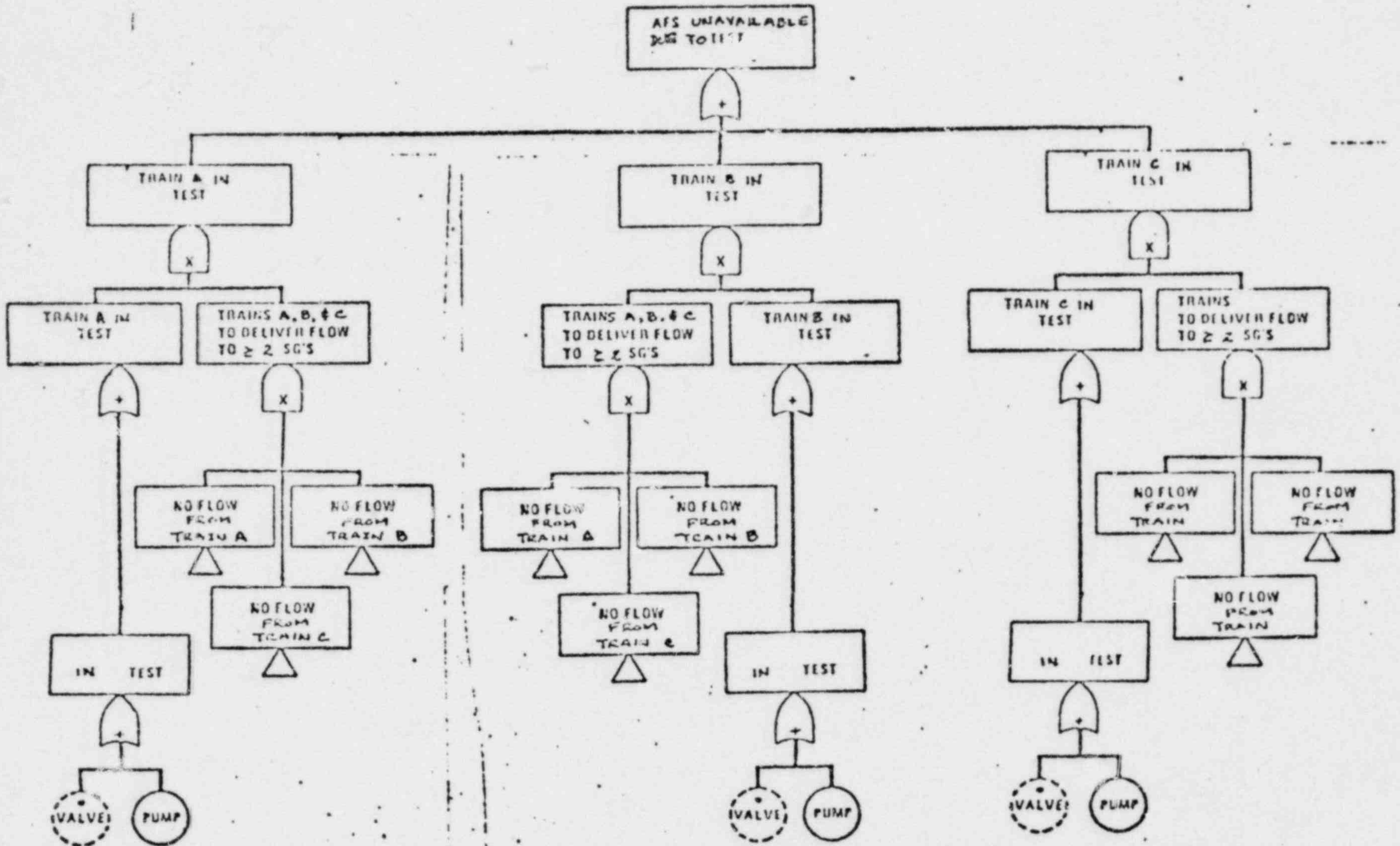


Fig. E-1 Maintenance Fault Tree



*VALVES WERE CONSIDERED BUT HAD NO IMPACT DUE TO TESTING DURING SHUTDOWN.

Fig. E-2 Test Fault Tree

APPENDIX F

COMMON CAUSE FAILURES
Hardware, Test and Maintenance, Human Errors

APPENDIX F

COMMON CAUSE FAILURES
Hardware, Test and Maintenance and Human Errors

F. Common Cause Hardware, Test and Maintenance and Human Error Analysis

Common cause analysis was performed both qualitatively and quantitatively; qualitatively to identify potential sources of common cause failures and quantitatively to indicate the limited effect that increased redundancy can have on the reliability of a system.

Qualitative Analysis - The identification of common or similar hardware, test, maintenance, human actions or physical links between redundant trains was the first step in this analysis. Based on the logic modeling (RBDs and FTs) the experience with other similar systems which use redundancy, the testing and maintenance plans, the operator interactions, the power supplies and service systems, the AFS Trains A and B can be classified as partly diverse as shown in Figure F.1, Train C is almost fully diverse from A and B. The major dependencies from the hardware viewpoint have been accounted for by considering the different initiating events which impact the power supplies for each train. The hardware dependencies are mostly outside the AFS components. These include check valves and blowdown valves on the steam generators.

As a final qualitative check, the potential for common cause failures as discussed in reference 6 were reviewed and are addressed below. Reference 6, listed seven "common cause" failures that occurred in 1975 AFS experience. These failures are discussed below as to the effect if they were to occur in Byron/Braidwood AFS:

A. Operator failed to open the valve from the condensate storage tank to the Train A and B pumps. The two AFS pump loops failed to be available on demand as required by technical specifications. Docket 50317-516.

   This failure indicates that a "single" valve provided condensate to the Train A and B AFS pumps. This appears to be a "single" failure point. The Byron /Braidwood AFS has separate supply lines and valves to each of the AFS pumps. Thus, a single valve closure will not cause AFS failure. Only a common cause failure, that of a multiple redundant inadvertant valve closure, will simulate this condition in the Byron/Braidwood AFS Train A and B design. In the Byron/Braidwood design Train C comes from an independent source of condensate. Also, the Essential Service Water (ESW) System can "automatically" supply water to Trains A and B should be condensate supply be unavailable.

B. Filters (in parallel) on suction side of three pumps plugged up with foreign material which restricted flow. Docket 50305-354.

   Byron/Braidwood AFS pumps have startup suction filters. Train A and B pumps have full flow test procedures that will detect any restricted flow including valve plugging.

C Condensate storage tank water level was intentionally drawn down below technical specification limits to maintain maximum steam generator blowdown. Failure to maintain water supply to multiple AFS pumps within specifications. Docket 50315-340.

Byron/Braidwood condensate storage tank for Trains A and B maintain a maximum of 500,000 gallons and, when the volume decreases to 200,000 gallons, the refill system can provide makeup water. The AFS requirement is 200,000 gallons. In addition, the other unit condensate storage tank with a maximum capacity of 500,000 gallons can be manually transferred to the AFS. A backup water supply is also automatically available from the essential service water system.

D. Condensate storage tank water level was intentionally drawn down below technical specification limits because makeup water supply was dirty (high oxygen content). Failure to maintain water supply to multiple AFWS pumps with specifications. Docket 50247-449.

In the B/B design the multiple supplies of water reduce dependence on any one supply.

E. Two AFS pumps failed to start because of defective contol switches which failed to close contacts. Docket 50305-350.

This could happen in any AFS. Since the Train B pump is an automatically initiated diesel drive, the Train A pump is an automatically started electric drive, and Train C is a manually started electric drive; the control switching has elements of diversity. Thus, all AFS failure requires independent failures in diverse systems. Common switches and breakers in the 2 out of 4 Auxiliary Feedwater Actuation logic for Trains A and B appear to have the greatest potential for a common cause failure. Manual override of the logic can minimize this potential common cause failure in the Byron/Braidwood designs.

F. A breaker accidentally opened and interrupted power to the turbine overspeed protection (which tripped the reactor), and also interrupted power to an AFS lube oil pump preventing start of the related AFS pump. Docket 50305-361.

The Byron/Braidwood AFS lube oil pump is a direct mechanical drive from the diesel driven feed pump. Thus, this failure mode should have no effect on the Byron/Braidwood AFS reliability.

G. Two AFS valves were upgraded during the licensing process and were not seismically qualified because of oversight. Docket 50289-491.

All safety related valves on the Byron/Braidwood AFS are seismically qualified.

Quantitative Analysis - The method known as the Beta Factor Method of Reference 9 was used to quantitatively estimate the effect of common cause failures. Simply stated, the Beta factor method assumes that a fraction of the operationally independent failure probabilities of one loop of a redundant

system will result in the loss of all the redundant loops in that system. The hand calculations based on the point estimate uses a generic Beta Factor of $\beta = 3\times10^{-2}$. This Beta Factor is a median value based on an assumed range of 10^{-1} to 10^{-3}. The common cause failure probability, Q_{cc}, for a redundant system can be approximated by the failure probability of one loop of a redundant system, Q_{loop}, times β. The total failure rate is the sum of the common cause failure contributions added to the independent failures in redundant Trains. Beta factors are determined from data and are significantly different for specific equipment types. Contributions to the Beta factor include human error, manufacturing, design, maintenance, testing, quality assurance, and external events.

The equations used to calculate the failure probability on demand were developed for each "AND" gate in the fault tree. The system failure probability Q_f is determined from the following formulation:

$$Q_f = Q_{independent\ failures} + Q_{common\ cause\ failures} = Q_{if} + Q_{ccf}$$

$$Q_f = (Q_1(1-\beta) \times Q_2(1-\beta) \times \ldots Q_n(1-\beta)) + \beta \frac{Q_1+Q_2+\ldots Q_n}{n},$$

where n is the number of redundant Trains

$$Q_{independent\ failure} = (Q_1(1-\beta) \times (Q_2(1-\beta) \times \ldots Q_n(1-\beta)) + \text{other component failures}$$

$$Q_{ccf} = \beta_a \frac{Q_{a1} + Q_{a2}+\ldots Q_{an}}{n} + \beta_b \frac{Q_{b1} + Q_{b2}+\ldots Q_{bn}}{n} + \ldots$$

$$+ \beta_j \frac{Q_{j1} + Q_{j2}+\ldots Q_{jn}}{n}$$

where a, b ... j are the common mode contributions from each redundant system within the system fault tree.

Since $\beta = .1$ the contribution of $(1-\beta)$ in the $Q_{independent}$ failure probability has only a small impact on the numerical estimate and therefore only slightly decreases the independent estimate.

In general, the Beta Factor approach to common cause failure estimates has its greatest impact on system reliability for highly redundant and simultaneous operating systems. It limits the system failure probability to about 1 to 3 orders of magnitude less than the single Train estimate.

The calculations for the uncertainty ranges in the AFS reliability also include the uncertainties for Beta Factors for individual components as assessed from Table F.2. The types of common cause failures which have occurred include a high contribution from electronic sytems (Beta = .1(2)), diesel generators (Beta = .03(2)), rotating machines with great separation and equipment which must pass strict single point failure analyses (Beta = .01(3)), and diverse systems with a very low potential for common cause failures (Beta = .001 or zero). These estimates include the impact of human errors in testing and maintenance equipment problems, etc. The trees of Appendix D and E

demonstrate more precisely how these factors can be considered. Once the fault tree of Appendix B was completed, the minimal cut sets were found by MOCUS. By inspection, the relative common cause failure fraction ordered by factors of 3 according to Table F.1 from Ref. 13 were made. These are shown in the data table of Appendix A.

For this analysis, the following assumptions were made:

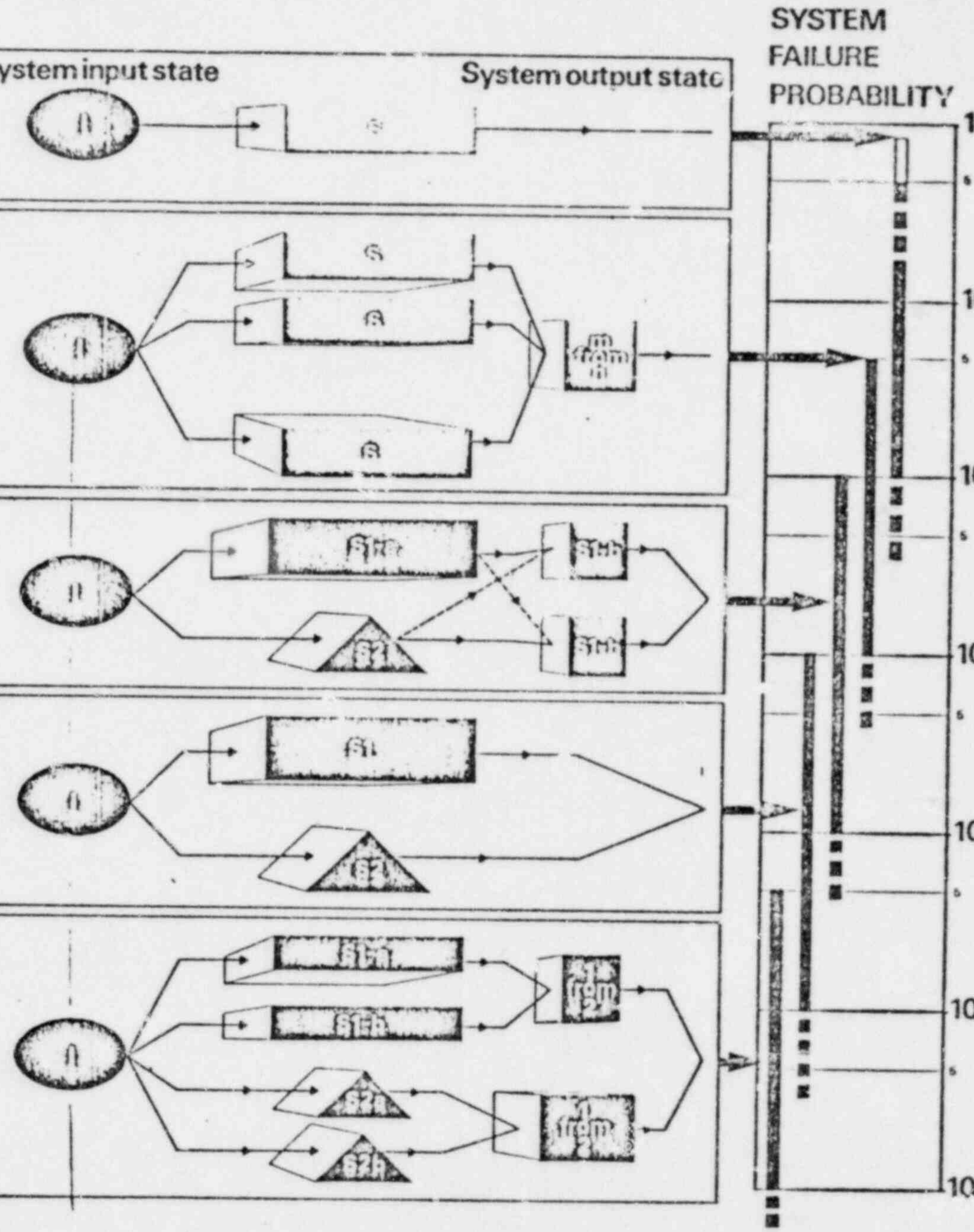
1. The valves, ESF signal, electric pump and human error were redundant and were considered with the common cause Beta factor.

2. The diesel drive and electric drive pump were diverse and thus subject to very low common cause Beta factor.

3. The major inter-Train common cause failures were considered separately for the three initiating events.

The common cause failure probability contributions to the AFS were calculated with the FT analysis code per inputs of Table A.IV and added to the independent failure probabilities. The results are shown in Sections 2 and 4.

System Design		
The simplest form of system is a single channel arrangement which will have a lower limit of probability of failure of approximately 10^{-3}. By introducing redundancy into the system to improve the reliability performance due to random independent failures the CMF problem becomes apparent, and limits the improvement.	**SINGLE CHANNEL SYSTEM**	
A system which has redundancy of its input channels only, might have a failure probability in a range from 5 10^{-2} to 10^{-3}. The inference of this arrangement is that the most significant contributors to failure probability are the components of the input measuring system, although the redundancy might also be extended to the other sub-systems to make some further reduction of failure probability possible.	**REDUNDANT SYSTEM**	
The logical development in the quest for greater system reliability is to reduce the CMF failure probability by introducing diversity into those parts of the system where identical redundancy has been previously considered as above. A typical system arrangement has functionally diverse measurements of two plant parameters, each of which is capable of initiating the required action and each of which has identical redundancy applied. A variation is to provide inter-connections between channels, but the intended improvement of system reliability due to random independent failures could be nullified by CMF possibilities that are introduced. Such a system would be capable of a failure probability in a range from 10^{-3} down to approximately 10^{-4}.	**PARTLY DIVERSE SYSTEM**	
For further reduction of system failure probability the independence of functionally diverse input measurements must be enhanced by applying equipment diversity and eliminating channel interconnections. Systems have therefore been evolved to give failure probabilities in a range from 10^{-3} to less than 10^{-5}. A common further development of this basic form is to include identical redundancy within each of the diverse sub-system.	**FULLY DIVERSE SYSTEM**	
The final stage in the evolution of redundancy system arrangement is to provide two separate and diverse output sub-systems such that they can each be operated by diverse input and intermediate sub-systems. These are therefore completely diverse and separate systems throughout, and might be expected to have failure probabilities in a range from 10^{-4} down to 10^{-6}. It is unlikely that for practical reasons, more than twofold diversity can be applied to each sub-system, as indicated on the diagram.	**TWO DIVERSE SYSTEMS**	



System Design Impact on Reliability

Table F.1 A Guide for Assessing the Impact of Common Mode Failures on System Reliability (Ref. 13)

TABLE F.2
POINT ESTIMATES OF β FOR VARIOUS COMPONENT TYPES

COMPONENT TYPE	FAILURE MODE	NO. OF COMPONENT FAILURES DUE TO COMMON CAUSES	TOTAL NO. OF COMPONENT FAILURES	β POINT ESTIMATE
DIESEL GENERATOR	FAIL TO START	7	50	0.14
	FAIL TO RUN	4	30	0.13
TRIP SYSTEM SENSOR CHANNEL	FAIL TO TRIP	14	153	0.09
VALVE	FAIL TO OPEN (CLOSE)	30	132	0.23
PUMP	FAIL TO START	2	14	0.14
	FAIL TO RUN	0	12	0.06[a]
PRESSURE, LEVEL, FLOW SWITCH	FAIL TO TRIP	41	77	0.53
AIRCRAFT ENGINE	FAIL TO RUN	136	1702	0.08

(a) OBTAINED FROM THE BINOMIAL DISTRIBUTION AT 50% CONFIDENCE,

$$(1 - \beta)^N = 1 - \gamma,$$

WHERE $N = 12$, $\gamma = 0.5$.

APPENDIX G

FAULT TREE ANALYSIS USING MOCUS AND STADIC

APPENDIX G

G. Fault Tree Analysis Using MOCUS and STADIC

Fault tree analysis is particularly useful in providing a schematic view of how the failures of primary events could lead to the failure of the top event. In this particular study, the top event is the failure to supply sufficient auxiliary feedwater to two of four steam generators within 20 mins of a LMFW, LMFW/LOOP, LMFW/LOAC events. To calculate the probabilities of these events, one can assign numerical probability values to the primary events in order to quantify the probability of faiure of the defined event.

The analysis consists of two basic steps. First, the minimal cut sets from the fault tree are determined. This is easily done with the MOCUS code (Ref. 10). An example of the cut sets determined using MOCUS is shown in Table G-1. These cut sets represent those events which could lead to the failure of three out of four steam generators to receive water and consequently, lead to the top event. From these cut sets one can formulate mathematical expressions which define the probability of the top event. The available failure rate data for the primary events have some degree of uncertainty and this should be accounted for in the quantification process. The STADIC code (Ref. 11) provides a fast and efficient method of doing this.

STADIC uses a Monte Carlo simulation technique to generate a pseudo-random sample statistical distribution for a user-defined output function. For example, in this study, one output function is the probability of the top event given the loss of main feedwater. The independent variables for this output function are the failure frequency rates of each primary event found in the minimal cut sets. Each variable exhibits random statistical variations represented by a particular probability distribution. STADIC generates a statistical distribution for the output function by selecting at random, values for each of the independent variables according to their assigned probability distributions and combining these distributions in accordance with the mathematical operations specified by the output function. A second set of randomly selected values for the independent variables is then chosen and a second evaluation of the function is made. This process is repeated several thousand times on the computer. The resulting values for the output function are then sorted and arranged in increasing order of magnitude. Confidence limits of the output distribution may then be determined directly from the ordered array. As an illustration of the method, we have in Table G-2 a list of equations defining the relationship of independent variables (QP) which lead to the top event. TRA1 and TRC2 give the mathematical operations leading to the failure of Trains A, B, and C, respectively. P1 and P12 represent all combinations of independent failures of primary events in the fault tree. P13 accounts for the common mode failures in Trains A and B. P14 accounts for the common mode failures within Train C such as failure of all the condensate booster pumps. P15 accounts for the common mode failures within Train B and P16 considers the common mode failures common to all Trains. Each common mode pair is defined by the lowest pair member mathematically. For example, pair 51, 33, and 7 is represented by 7 which is for check valve failure to close in the main feedwater system allowing backflow from the steam generators. This involves check valves affecting all Trains. Likewise, pairs 39-40, and 41-42 represent common faults in the diesel cooling

and lubrication which arise from the use of both internal and external systems (i.e. piping leaks. This type of fault would defeat the redundancy built in the diesel unit, and could prevent Train B from operating. XY(4) is the mathematical expression for the failure upon demand of the top event due to independent failures of the basic components; while XY(7) represents the failure per year of the top event given loss of main feedwater. The rest of the equations are self-explanatory.

TABLE G-1

MINIMAL CUT SETS FOR GATE G2

(LMFW)

CUT SETS WITH 1 COMPONENTS

NONE EXIST.

CUT SETS WITH 2 COMPONENTS

NONE EXIST.

CUT SETS WITH 3 COMPONENTS

1)	P6A	P6B	P6C
2)	P6A	P6B	P6D
3)	P6A	P6C	P6D
4)	P6B	P6C	P6D

CUT SETS WITH 4 COMPONENTS

1)	P6A	P6B	P4C	P3C
2)	P6A	P6B	P4D	P3D
3)	P6A	P6C	P4D	P3D
4)	P6B	P6C	P4D	P3D
5)	P6A	P4B	P6C	P3B
6)	P6A	P4B	P6D	P3B
7)	P6A	P4C	P6D	P3C
8)	P6B	P4C	P6D	P3C
9)	P4A	P6B	P6C	P3A
10)	P4A	P6B	P6D	P3A
11)	P4A	P6C	P6D	P3A
12)	P4B	P6C	P6D	P3B
13)	P6A	P6B	P5C	P3C
14)	P6A	P5B	P6C	P3B
15)	P5A	P6B	P6C	P3A
16)	P6A	P6B	P5D	P3D
17)	P6A	P5B	P6D	P3B
18)	P5A	P6B	P6D	P3A
19)	P6A	P6C	P5D	P3D
20)	P6A	P5C	P6D	P3C
21)	P5A	P6C	P6D	P3A

* See Table A.IV.1 for notational symbols

TABLE 6-1 CONTINUED

CUT SETS WITH 4 COMPONENTS

22)	P6B	P6C	P5D	P3D
23)	P6B	P5C	P6D	P3C
24)	P5B	P6C	P6D	P3B

CUT SETS WITH 5 COMPONENTS

1)	P6A	P4B	P4C	P3B	P3C
2)	P6A	P4B	P4D	P3B	P3D
3)	P6A	P4C	P4D	P3C	P3D
4)	P6B	P4C	P4D	P3C	P3D
5)	P4A	P6B	P4C	P3A	P3C
6)	P4A	P6B	P4D	P3A	P3D
7)	P4A	P6C	P4D	P3A	P3D
8)	P4B	P6C	P4D	P3B	P3D
9)	P6A	P5B	P4C	P3B	P3C
10)	P5A	P6B	P4C	P3A	P3C
11)	P6A	P5B	P4D	P3B	P3D
12)	P5A	P6B	P4D	P3A	P3D
13)	P6A	P5C	P4D	P3C	P3D
14)	P5A	P6C	P4D	P3A	P3D
15)	P6B	P5C	P4D	P3C	P3D
16)	P5B	P6C	P4D	P3B	P3D
17)	P4A	P4B	P6C	P3A	P3B
18)	P4A	P4B	P6D	P3A	P3B
19)	P4A	P4C	P6D	P3A	P3C
20)	P4B	P4C	P6D	P3B	P3C
21)	P6A	P4B	P5C	P3B	P3C
22)	P5A	P4B	P6C	P3A	P3B
23)	P6A	P4B	P5D	P3B	P3D
24)	P5A	P4B	P6D	P3A	P3B
25)	P6A	P4C	P5D	P3C	P3D
26)	P5A	P4C	P6D	P3A	P3C
27)	P6B	P4C	P5D	P3C	P3D
28)	P5B	P4C	P6D	P3B	P3C
29)	P4A	P6B	P5C	P3A	P3C
30)	P4A	P5B	P6C	P3A	P3B
31)	P4A	P6B	P5D	P3A	P3D
32)	P4A	P5B	P6D	P3A	P3B
33)	P4A	P6C	P5D	P3A	P3D
34)	P4A	P5C	P6D	P3A	P3C
35)	P4B	P6C	P5D	P3B	P3D
36)	P4B	P5C	P6D	P3B	P3C
37)	P6A	P5B	P5C	P3B	P3C
38)	P5A	P6B	P5C	P3A	P3C
39)	P5A	P5B	P6C	P3A	P3B
40)	P6A	P5B	P5D	P3B	P3D
41)	P5A	P6B	P5D	P3A	P3D
42)	P5A	P5B	P6D	P3A	P3B
43)	P6A	P5C	P5D	P3C	P3D
44)	P5A	P6C	P5D	P3A	P3D
45)	P5A	P5C	P6D	P3A	P3C
46)	P6B	P5C	P5D	P3C	P3D
47)	P5B	P6C	P5D	P3B	P3D
48)	P5B	P5C	P6D	P3B	P3C

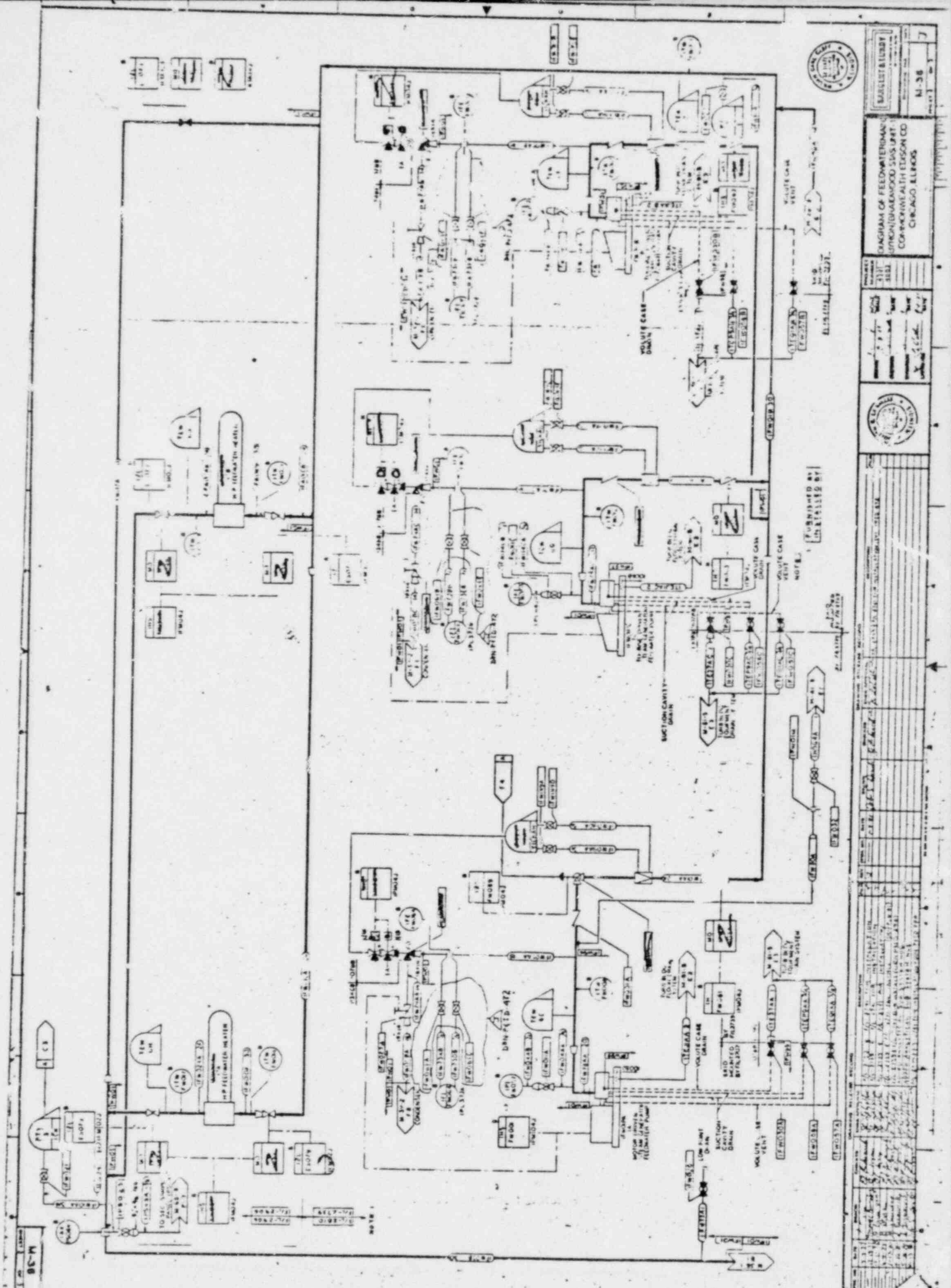
G-2 LOSS OF MAIN FEEDWATER EQUATIONS

```
      QP5=((1-BP17)*QP17)*((1-BP37)*QP37)
      TRA1=(QP18+(1-BP12)*QP12+(1-BP7)*QP7)
      TRA2=((1-BP8)*QP8)*(QP10*(1-BP10)+QP11*(1-BP11))
      TRA3=(QP22+(1-BP23)*QP23*(1-BP24)*QP24)*(QP25+QP28+
    C (1-BP27)*QP27+(1-BP26)*QP26)+QP14*QP15+QP16*QP17
      TRA4=4.*((1-BP13)*QP13)**3.
      TRB1=QP35+QP36+(1-BP34)*QP34+(1-BP30)*QP30+(1-BP43)*QP43+QP44
    C  +QP46+QP47+QP45
      TRB2=(((1-BP31)*QP31+(1-BP32)*QP32)*(1-BP29)*QP29)+((1-BP37)
    C *QP37*QP38)+((1-BP39)*QP39*(1-BP40)*QP40)+((1-BP41)*QP41*
    C  (1-BP42)*QP42)+4*(((1-BP33)*QP33)**3)
      TRC1=(1-BP51)*QP51+QP53+(1-BP50)*QP50+QP49+(1-BP52)*QP52+
    C (1-BP55)*QP55+(1-BP56)*QP56+(1-BP54)*QP54
      TRC2=((1-BP57)*QP57)**4
      P1=QP1*(1-BP2)*QP2
      P2=4*((1-BP6)*QP6)**3+4*((1-BP6)*QP6+(QP3*((1-BP4)*QP4
    C +QP5)))**3
      P3=TRC1*TRA1*TRB1
      P4=TRC1*TRA1*TRB2
      P5=TRC1*TRA2*(TRB1+TRB2)
      P6=TRC1*TRA3*(TRB1+TRB2)
      P7=TRC1*TRA4*(TRB1+TRB2)
      P8=TRC2*TRA1*TRB1
      P9=TRC2*TRA1*TRB2
      P10=TRC2*TRA2*(TRB1+TRB2)
      P11=TRC2*TRA3*(TRB1+TRB2)
      P12=TRC2*TRA4*(TRB1+TRB2)
      P13=BP12*QP12+BP23*QP23+BP9*QP9+BP10*QP9
      P14=BP54*QP54+BP57*QP57+BP50*QP50+BP52*QP52
      P15=BP39*QP39+BP41*QP41
      P16=BP7*QP7+BP6*QP6
      P17=(QP9+QP63)*(QP62+QP61)+QP63
      XY(1)=TRA1+TRA2+TRA3+TRA4+XLT
      XY(2)=TRB1+TRB2+XLT
      XY(3)=TRC1+TRC2+XLT
      XY(4)=(XY(1)*XY(2)+P17)*XY(3)
      XY(5)=XY(2)+P15
      XY(6)=XY(3)+P14
      XY(7)=XY(1)*XY(5)+P13+P17
      XY(8)=XY(7)*XY(6)+P16
      XY(9)=FLMFW*XY(4)
      XY(10)=FLMFW*XY(8)
      XY(11)=XY(4)+P1+P2
      XY(12)=XY(8)+P1+P2+BP2*QP2+BP6*QP6+BP17*QP17
      XY(13)=XY(11)*FLMFW
      XY(14)=XY(12)*FLMFW
```

APPENDIX H

BYRON/BRAIDWOOD

PIPING AND INSTRUMENT DRAWINGS FOR AUXILIARY FEEDWATER



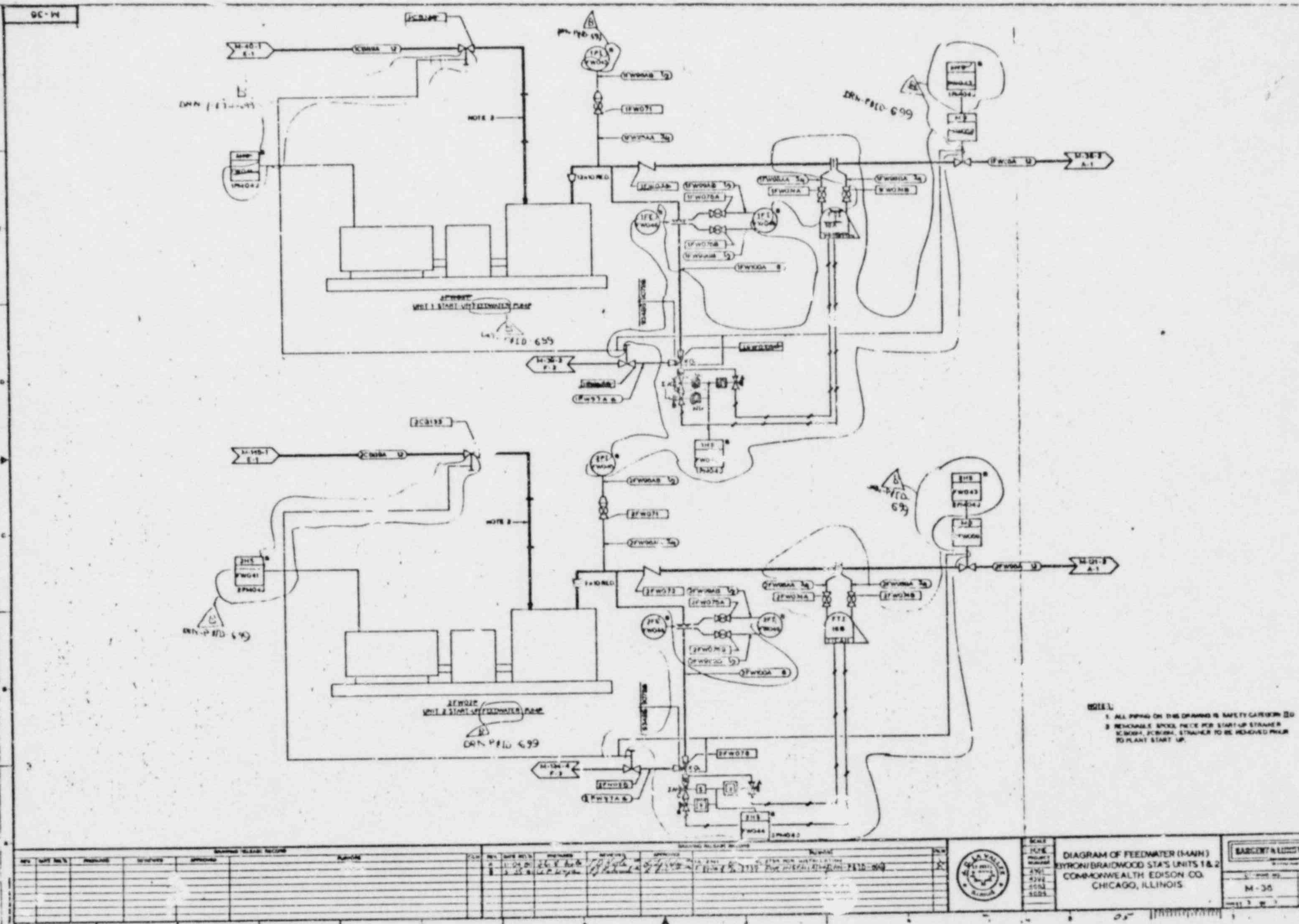


DIAGRAM OF FEEDWATER (MAIN)
BYRON/BRAIDWOOD STA'S UNITS 1 & 2
COMMONWEALTH EDISON CO.
CHICAGO, ILLINOIS
SARGENT & LUNDY
M-36
8
UNIT 1 START-UP FEEDWATER PUMP
UNIT 2 START-UP FEEDWATER PUMP
NOTE 2
DRN PFID 699
NOTES:
TO PLANT START UP.

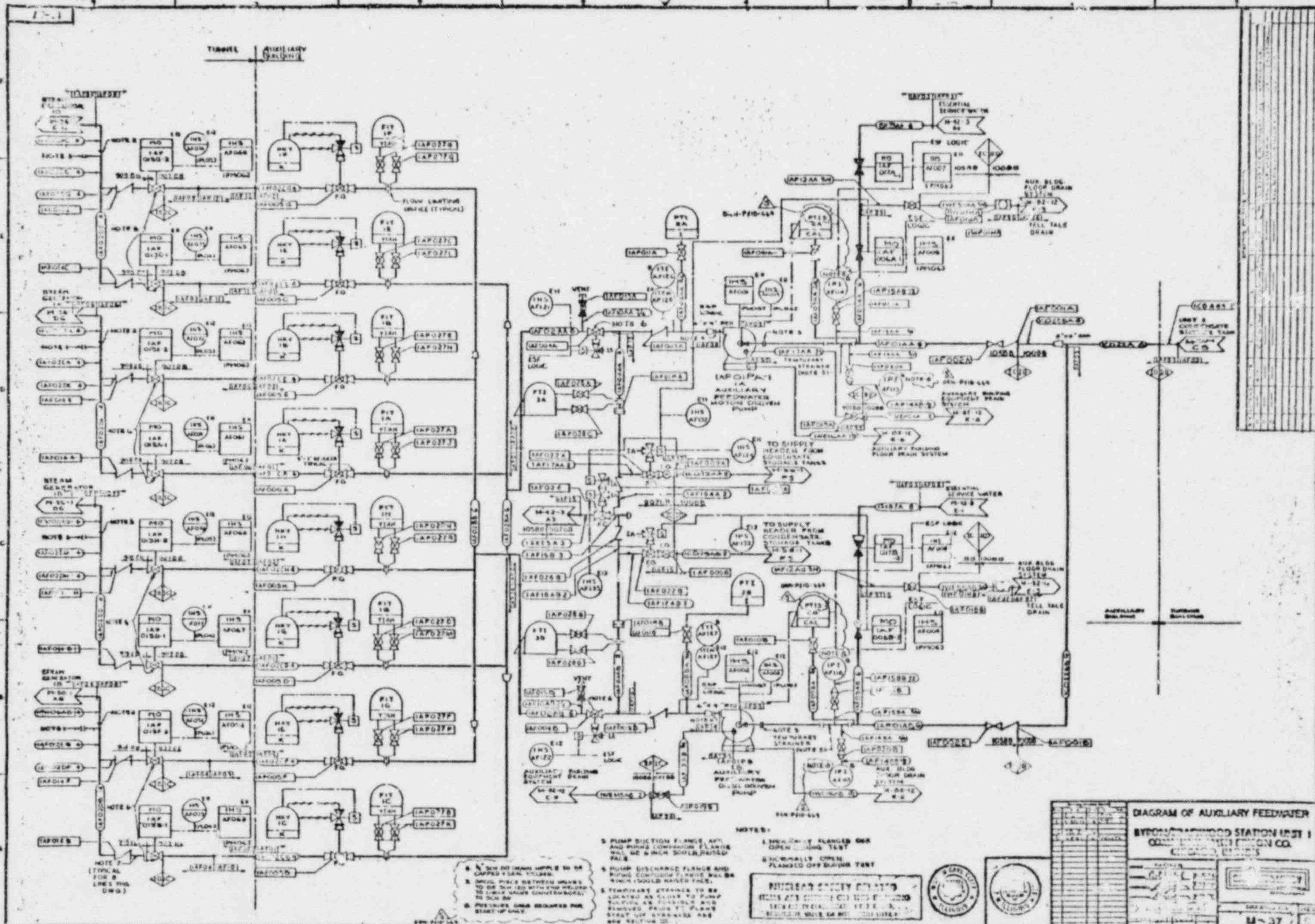
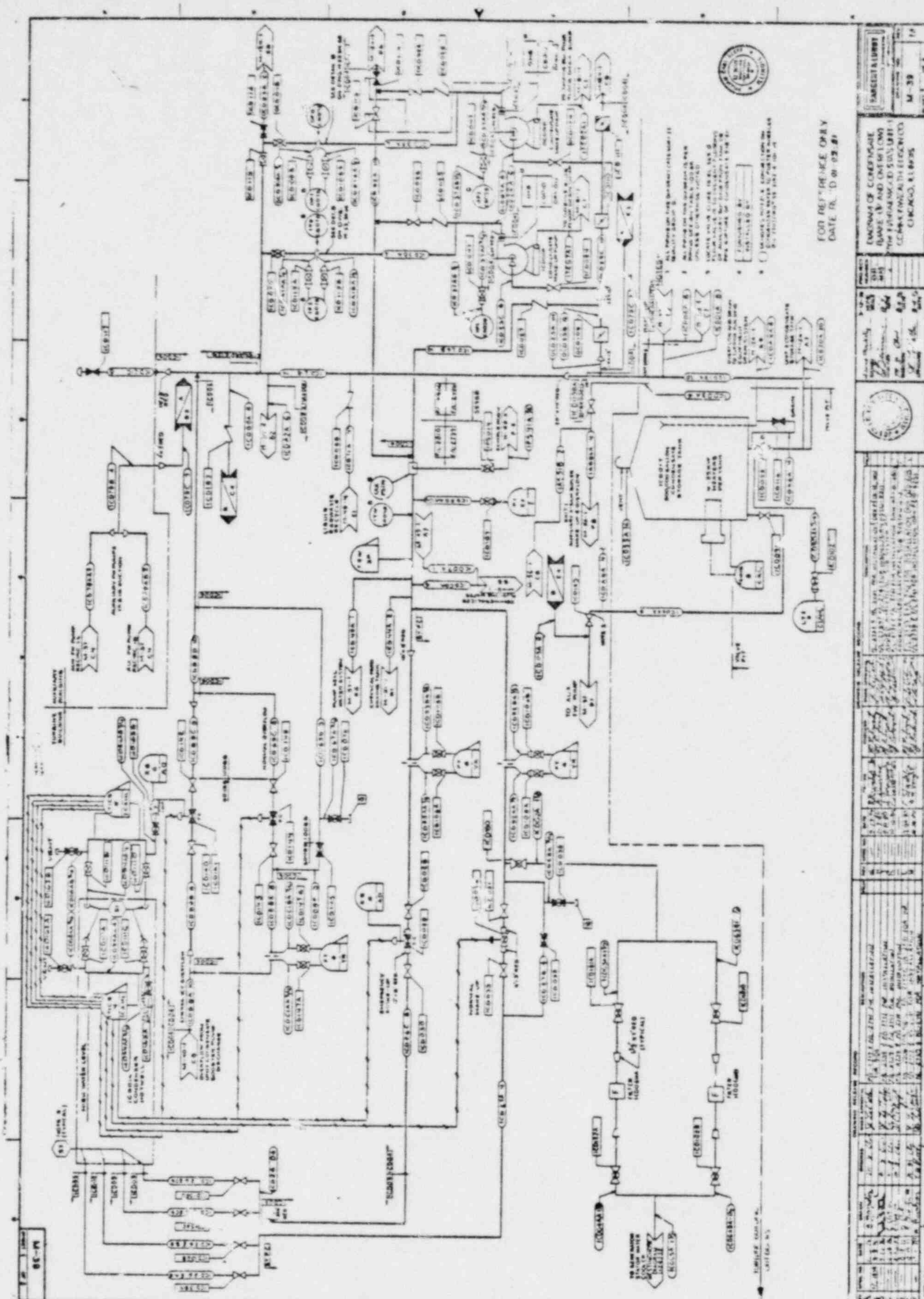


DIAGRAM OF AUXILIARY FEEDWATER
M-37
NUCLEAR SAFETY RELATED


M-39

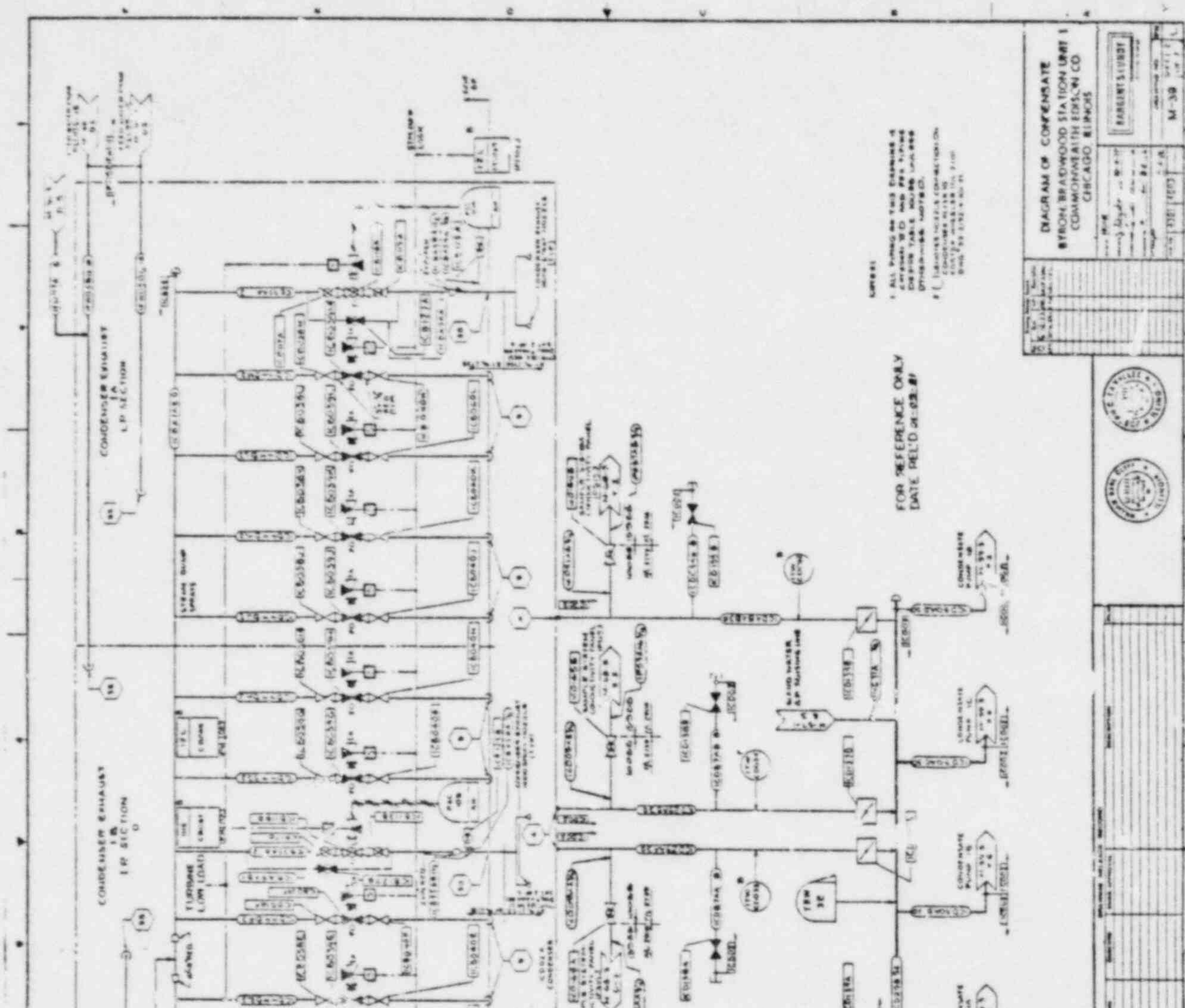


DIAGRAM OF CONDENSATE
BYRON/BRAIDWOOD STATION UNIT 1
COMMONWEALTH EDISON CO.
CHICAGO, ILLINOIS
SARGENT & LUNDY
M-39
FOR REFERENCE ONLY
CONDENSER EXHAUST
LP SECTION
TURBINE
CONDENSATE PUMP 1A
CONDENSATE PUMP 1B
CONDENSATE PUMP 1C

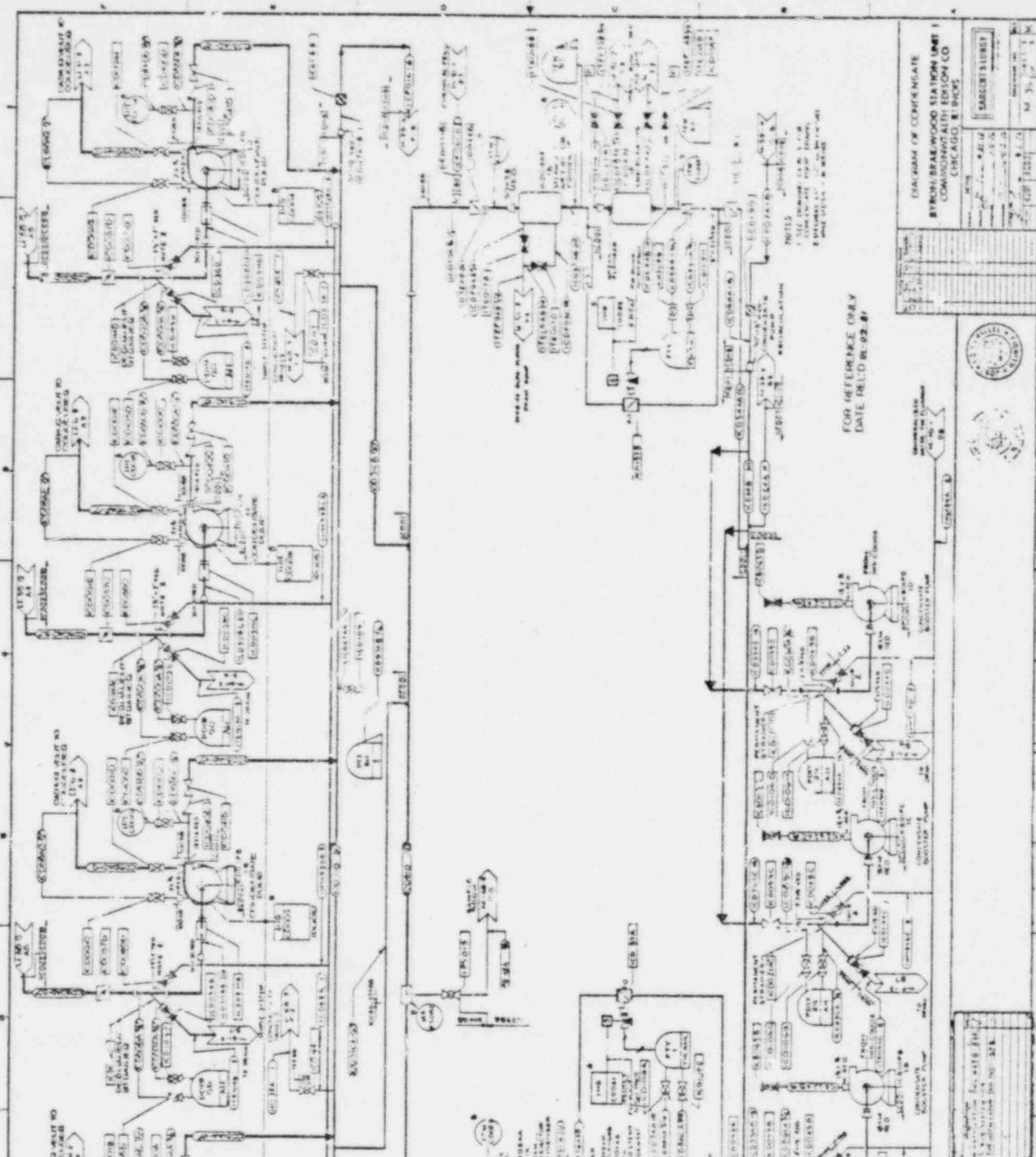


DIAGRAM OF CONDENSATE
COMMONWEALTH EDISON CO.
CHICAGO, ILLINOIS
FOR REFERENCE ONLY

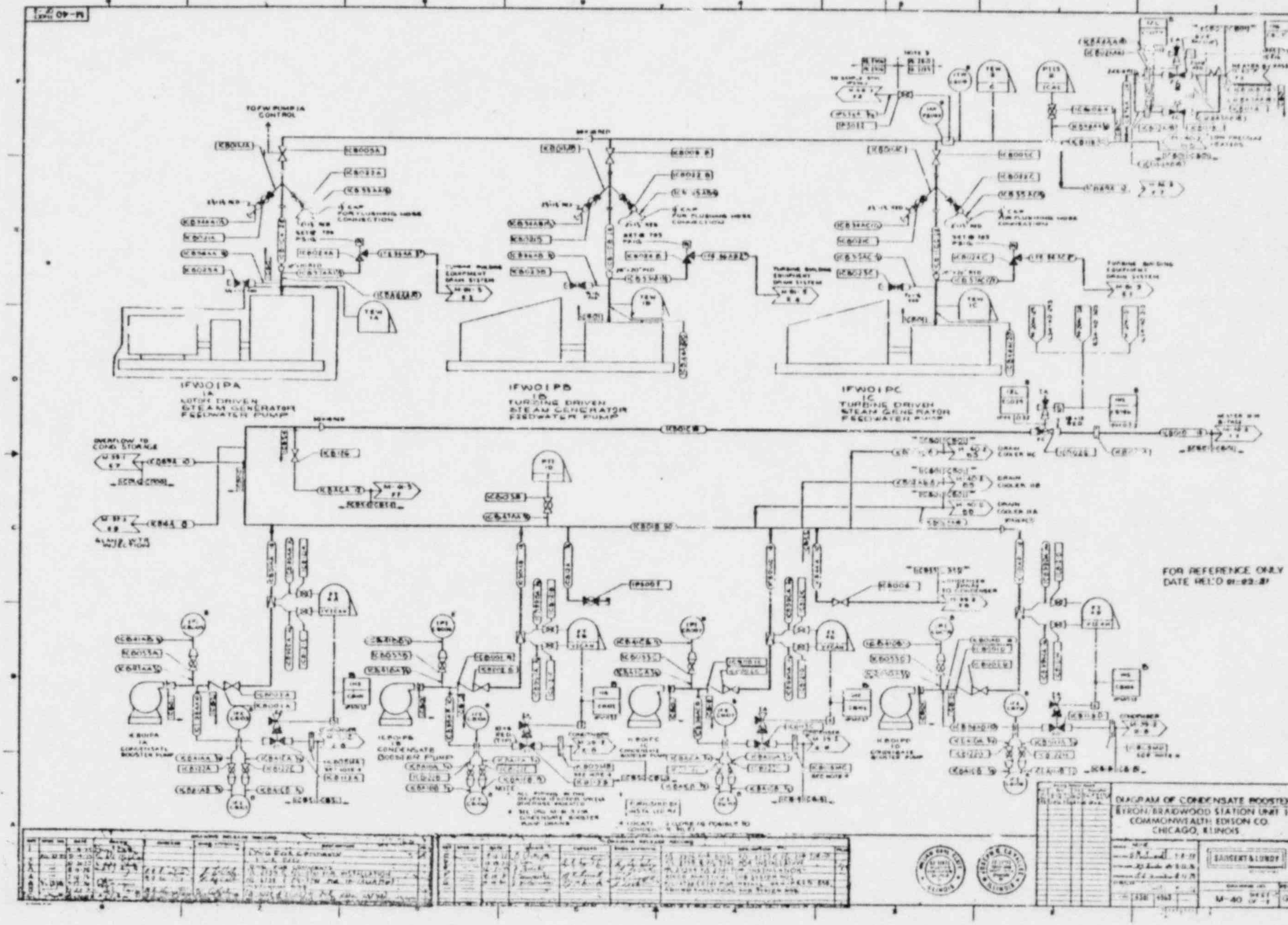


DIAGRAM OF CONDENSATE BOOSTER
BYRON/BRAIDWOOD STATION UNIT 1
COMMONWEALTH EDISON CO.
CHICAGO, ILLINOIS
SARGENT & LUNDY
M-40
FOR REFERENCE ONLY
DATE REC'D 01-23-81
1FW01PA
1A
MOTOR DRIVEN
STEAM GENERATOR
FEEDWATER PUMP
1FW01PB
1B
TURBINE DRIVEN
STEAM GENERATOR
FEEDWATER PUMP
1FW01PC
1C
TURBINE DRIVEN
STEAM GENERATOR
FEEDWATER PUMP
TO FW PUMP 1A CONTROL
OVERFLOW TO COND. STORAGE
CONDENSATE BOOSTER PUMP

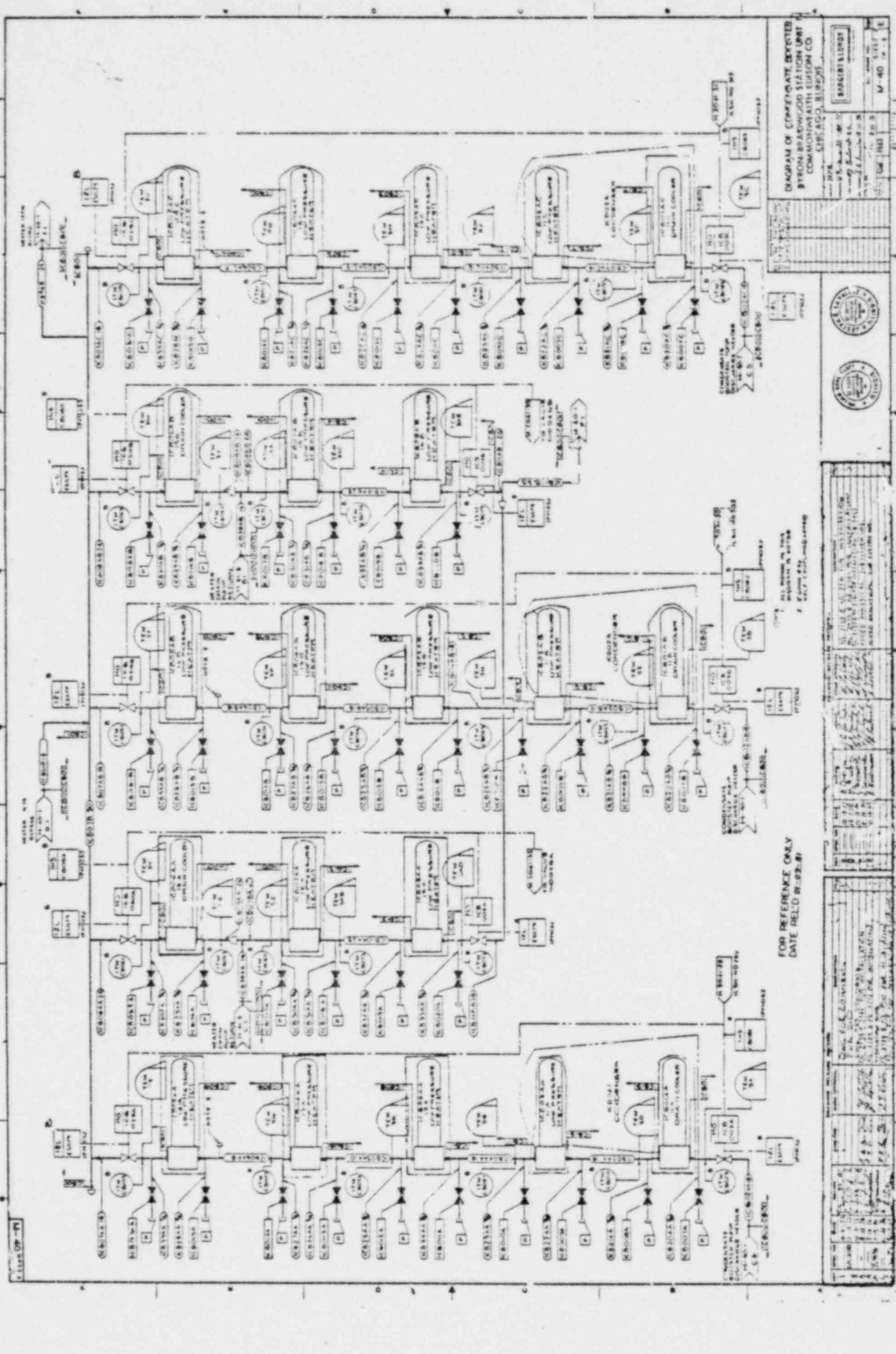
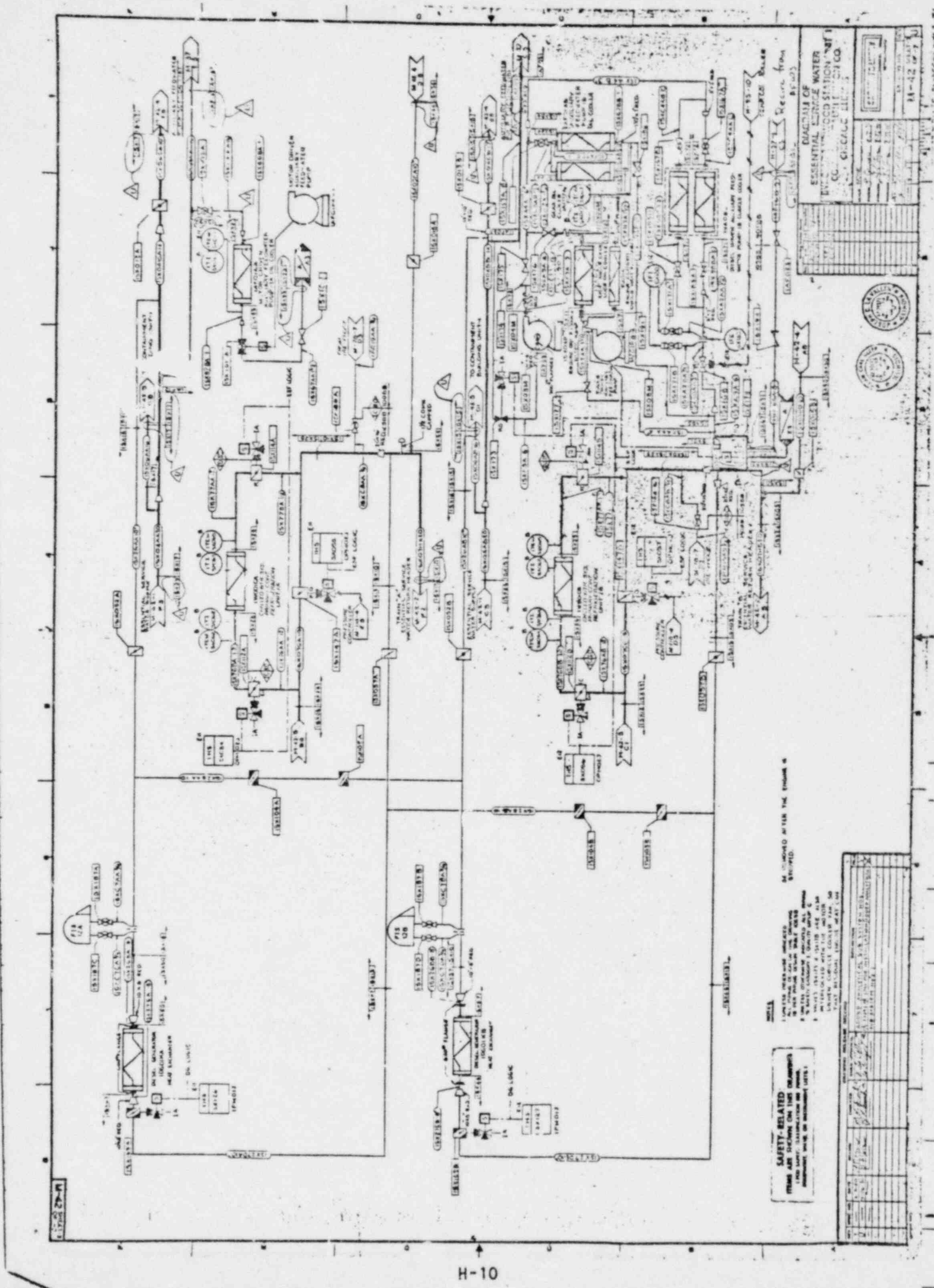
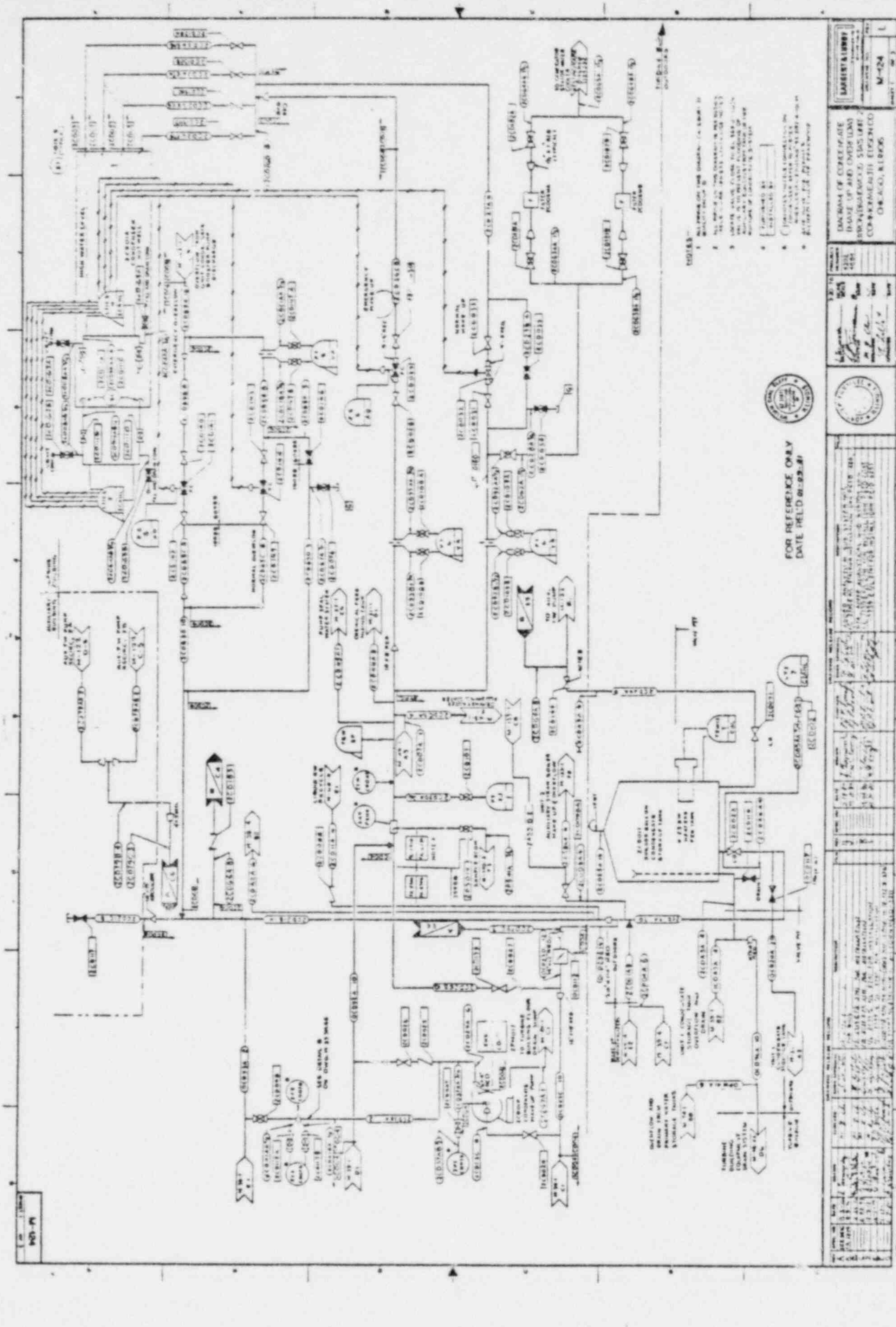


DIAGRAM OF CONDENSATE BOOSTER
BYRON/BRAIDWOOD STATION UNIT 1
COMMONWEALTH EDISON CO.
CHICAGO, ILLINOIS
SARGENT & LUNDY
M-40
FOR REFERENCE ONLY


SAFETY-RELATED
DIAGRAM OF ESSENTIAL SERVICE WATER
TO CONTAINMENT BUILDING UNITS


FOR REFERENCE ONLY
SARGENT & LUNDY
M-124
DIAGRAM OF CONDENSATE
MAKE UP AND OVERFLOW
BRAIDWOOD STATION UNIT 2
COMMONWEALTH EDISON CO
CHICAGO, ILLINOIS

APPENDIX I

BYRON/BRAIDWOOD ELECTRICAL DRAWINGS

A. Plant Power System

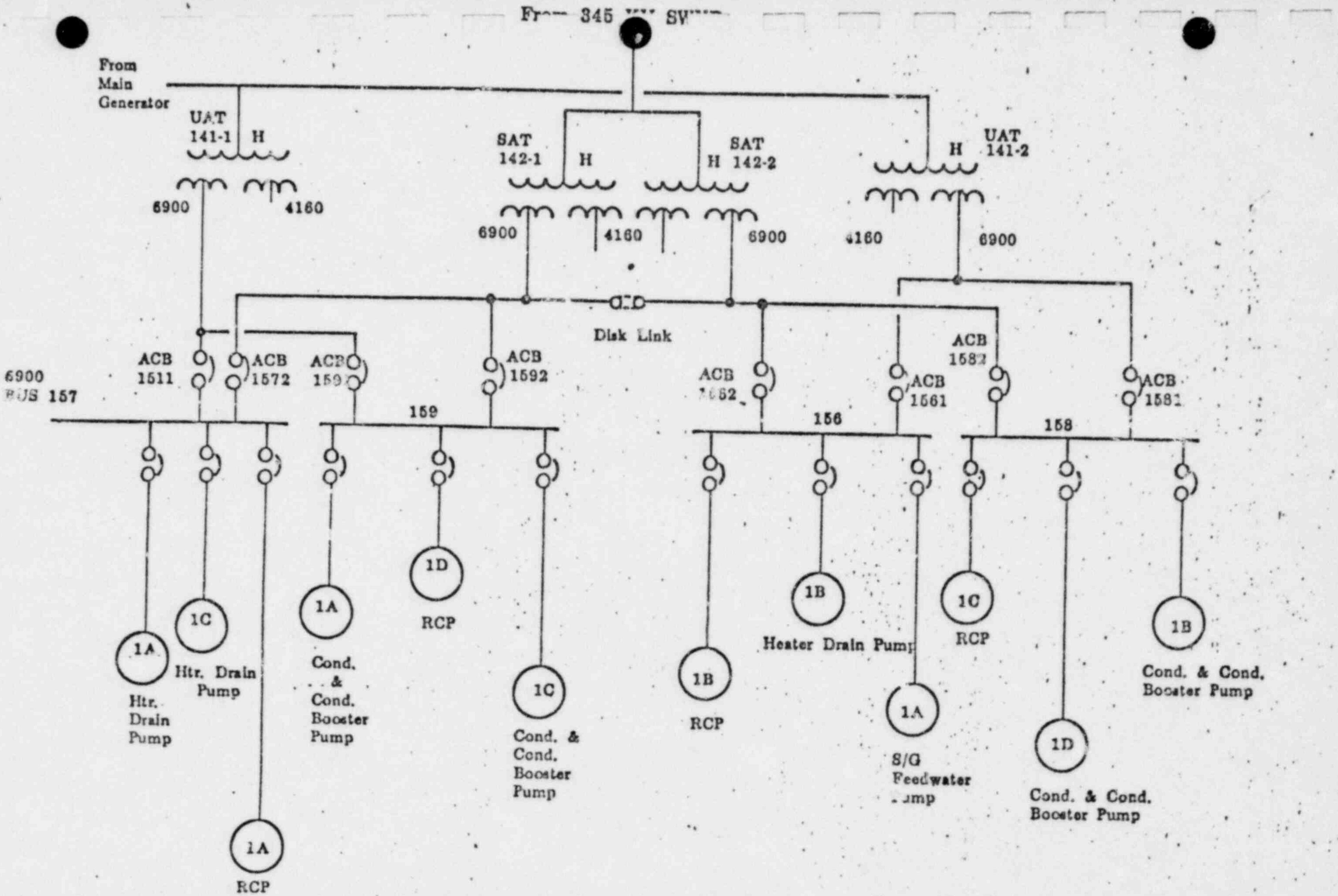


Figure 31-I-D4
6900 Volt Distribution System

Part II: Evaluation to the AFW System to Standard Review Plan Section 10.4.9 and Branch Technical Position ASB 10-1 (ESF Pump Trains Only)

Section 1 Evaluation to the Standard Review Plan Section 10.4.9

1.1 System components and piping have sufficient physical separation or shielding to protect the essential portions of the system from the effects of internally and externally generated missiles.

Response:

Byron/Braidwood system components and piping satisfy physical separation and shielding requirements relating to internally and externally generated missiles.

See FSAR Section:

3.5.1.1	Internally Generated Missiles (Outside Containment)
3.4.1.4	Missiles Generated by Natural Phenomena
3.5.1.5	Missiles Generated by Events Near the Site
3.5.1.6	Aircraft Hazards
3.5.2	Systems to be Protected
3.5.3	Barrier Design Procedure

1.2 The system satisfies the recommendations of Branch Technical Position ASB 3-1 with respect to the effects of pipe whip and jet impingement that may result from high or moderate energy piping breaks or cracks (in this regard the AFS is considered to be a high energy system).

Response:

The Auxiliary Feedwater System is not used for normal startup and shutdown at Byron and Braidwood and is therefore considered to be a moderate energy system and satisfies BTP ASB 3-1. See FSAR Postulated Rupture of Piping and FSAR Subsection 10.4.9.2, Auxiliary Feedwater System description.

1.3 The system and components satisfy design code requirements, as appropriate for the assigned quality group and seismic classifications.

Response

Byron/Braidwood station system and components satisfy design code requirements for assigned quality group and seismic classifications.

See FSAR Sections:

10.4.9.1 AFW System Design Basis
10.4.9.2 AFW System Description

1.4 The failure of non-essential equipment or components does not affect essential functions of the system.

Response:

Non-essential systems interfacing with the Auxiliary Feedwater System are:

1). The Condensate Storage Tank: Failure of the condensate storage tank or suction lines is accommodated by Essential Service Water Backup supply to each Auxiliary Feedwater pump suction.

2). Station Air System: Failure of the Air System is accommodated by failing air operated flow control valves open on loss of air. Therefore, failure of non-essential equipment does not affect the essential functions of the Auxiliary Feedwater System.

1.5 The system is capable of withstanding a single active failure.

Response:

The Auxiliary Feedwater system is capable of withstanding a single active failure.

See FSAR Sections:

7.2.2.2.3b	Single Failure Criteria (Electrical)
10.4.9.2	AFW System Description (Mechanical)
Table 10.4-3	Failure Modes - Effects Analysis.

1.6 The system possesses diversity in motive power sources such that system performance requirements may be met with either of the assigned power sources, e.g., a system with an a-c subsystem and a redundant steam/d-c subsystem.

Response:

The Auxiliary Feedwater System possesses diversity in motive power sources as stated in the FSAR.

See FSAR Sections:

10.4.9.1 AFW System Design Basis
10.4.9.2 AFW System Description
7.3.1.1.6 AFW System Operation
See also Section 3.0 of Part 1 of this response and the response to Question 10.36.

1.7 The system precludes the occurrence of fluid flow instabilities, e.g., water hammer, in system inlet piping during normal plant operation or during upset or accident conditions (see SRP Section 10.4.7).

Response:

Byron/Braidwood Auxiliary Feedwater System includes water hammer preventation capabilities.

See FSAR Sections:

10.4.7.3 Water Hammer Prevention Features
10.4.9.3.1 Auxiliary Feedwater System General Safety Evaluation.

1.8 Functional capability is assured by suitable protection during abnormally high water levels (adequate flood protection considering the probable maximum flood).

Response:

The functional capabilities of systems are assured as stated in FSAR Section 3.4; Water Level (Flood) Design.

1.9 The capability exists to detect, collect, and control system leakage and to isolate portions of the system in case of excessive leakage or component malfunctions.

Response:

Pump seal leakage is directed to the Auxiliary Building Equipment Drain System. Visual periodic inspections will provide indication of system leakage. The piping and valving in the Auxiliary Feedwater System is sufficiently diverse to allow component isolation for

malfunction and repair while still maintaining the essential functions of the Auxiliary Feedwater System.

1.10 Provisions are made for operational testing.

Response:

Provisions are made for operational testing of Byron/Braidwood Stations Auxiliary Feedwater as outlined in FSAR Subsection 10.4.9.4; AFW System Inspection and Testing Requirements.

1.11 Instrumentation and control features are provided to verify the system is operating in a correct mode.

Response:

Technical Specifications require verification by flow demonstration or valve position verification for proper operating alignment. Instrumentation is available to verify system alignment by either means from the control room.

See FSAR Subsection 10.4.9.5; AFW System Instrumentation Application.

1.12 The system is capable of automatically initiating auxiliary feedwater flow upon receipt of a system actuation signal.

Response:

The Auxiliary Feedwater System is capable of automatic initiation.

See FSAR Sections:

10.4.9.3.1	AFW System General Safety Evaluation
7.3	Engineered Safety Features Actuation System
7.2.1.1.2.e	Low-Low Steam Generator Water Level Trip.

1.13 The system satisfies the recommendations of Regulatory Guide 1.62 with respect to the system capability to manually initiate protective action by the auxiliary feedwater system.

Response:

The commitment to comply with the intent of Regulatory Guide 1.62 is found in FSAR Appendix A, page A-1.62-1.

1.14 The system design possesses the capability to automatically terminate auxiliary feedwater flow to a depressurized steam generator, and to automatically provide feedwater to the intact steam generator.

Response:

The flow orifices in the Auxiliary Feedwater lines automatically limit flow to a depressurized steam generator and insure flow to unaffected steam generators.

See FSAR Subsection 10.4.9.3.1; AFW System General Safety Evaluation.

1.15 The system possesses sufficient auxiliary feedwater flow capacity so that a cold shutdown can be achieved.

Response:

FSAR Subsection 10.4.9.1, AFW System Design Basis states a minimum of 200,000 gallons is necessary to cooldown to Residual Heat Removal System initiation. FSAR Chapter 16.0 Subsection 3.7.1.3 of the Technical Specification states the Limiting Condition for Operation for the Condensate Storage Tank. This limit is based on a sufficient volume of water in the condensate storage tank to meet the Design Basis Requirements for cooldown to RHR System initiation. In addition, sufficient water volume is available from the essential Service Water System to achieve cold shutdown.

1.16 The applicant's proposed technical specifications are such as to assure the continued reliability of the AFW during plant operation, i.e., the limiting conditions for operation and the surveillance testing requirements are specified and are consistent with those for other similar plants.

Response:

Byron/Braidwood Technical Specifications have been developed from the Standard Technical Specifications for Westinghouse plants.

Section 2: Branch Technical Position ASB 10-1

2.1 The auxiliary feedwater system should consist of at least two full-capacity, independent systems that include diverse power sources.

Response:

Byron/Braidwood Auxiliary Feedwater System consists of one 100% capabity emergency AC powered motor driven pump and one 100% capacity diesel driven pump (capable of supplying water independent of AC power availability).

See FSAR Sections:

10.4.9.1 AFW System Design Basis
10.4.9.2 AFW System Description
10.4.9.3 AFW System Safety Evaluation

2.2 Other powered components of the auxiliary feedwater system should also use the concept of separate and multiple sources of motive energy. An example of the required diversity would be two separate auxiliary feedwater trains, each capable of removing the afterheat load of the reactor system, having one separate train powered from either of two a-c sources and the other train wholly powered by steam and d-c electric power.

Response:

Motor operated valves in each auxiliary feedwater train employ diversity in power supplies. All operated valves employ the same diversity of supplies. This diversity is covered in FSAR Section 10.4.a.3, AFW System Safety Evaluation. In addition, Chapter 8.0 of the FSAR specifically addresses electrical power from all aspects applicable to train separation and diversity of power. All motor operated and air operated valves are normally open with the exception of essential service water suction valves which are normally closed.

2.3 The piping arrangement, both intake and discharge, for each train should be designed to permit the pumps to supply feedwater to any combination of steam generators. This arrangement should take into account pipe failure, active component failure, power supply failure, or control system failure that could prevent system function. One arrangement that would be acceptable is crossover piping containing valves that can be operated by remote manual control from the control room, using the power diversity principle for the valve operators and actuation systems.

Response:

A common header in the suction to the AFW pumps exists where the supply lines from the condensate storage tank combine. The line then splits to supply suction to the individual AFW pumps. A common header exists where the auxiliary feedwater enters the steam generator. This is down stream of the flow control valves on the discharge of the pumps. See FSAR Figure 10.4-2. Any of the AFW pumps can be aligned to supply any of the steam generators by operating motor operated valves from the control room.

2.4 The auxiliary feedwater system should be designed with suitable redundance to offset the consequences of any single active component failure; however, each train need not contain redundant active components.

Response:

See response to Item 5 of the previous section.

2.5 When considering a high energy line break, the system should be so arranged as to assure the capability to supply necessary emergency feedwater to the steam generators, despite the postulated rupture of any high energy section of the system, assuming a concurrent single active failure.

Response:

See response to Item 2 of the previous section.

Part III: Evaluation of the AFW System to the Generic Short Term and Long Term Requirements (ESF Pump Trains Only)

Section 1: Response to Short-Term Recommendations

1.1 NRC Recommendation GS-1

The licensee should propose modificaitons to the Technical Specifications to limit the time that one AFW system pump and its associated flow train and essential instrumentation can be inoperable. The outage time limit and subsequent action time should be as required in current Standard Technical Specifications; i.e., 72 hours and 12 hours, respectively.

Response

Commonwealth Edison has determined that the recommended technical specification is not necessary.

The Byron/Braidwood Auxiliary Feedwater System consists of two 100% pump trains. The design basis of this system as discussed in Subsection 10.4.9.1 of the FSAR and as analyzed in Section 15.0, Loss of Normal Feedwater indicates the adequacy of this system. Either train is sufficient to supply the required flow.

Subsection 3.7.1.2 of the Technical Specifications (FSAR Chapter 16.0) requires two pumps to be operable though one pump train may be inoperable up to 7 days. This is the same redundancy requirement as the rest of the safeguards pumps and is based on the above cited analyses.

1.2 NRC Recommendation GS-2

The licensee should lock open single valves or multiple valves in series in the AFW system pump suction piping and lock open other single valves or multiple valves in series that could interrupt all AFW flow. Monthly inspections should be performed to verify that these valves are locked and in the open position. These inspections should be proposed for incorporation into the surveillance requirements of the plant Technical Specifications. See Recommendation GL-2 for the longer term resolution of this concern.

Response

The AFW pumps are supplied from the Condensate Storage Tank via either of two separate lines, both of which are normally aligned to supply the AFW system.

One line has a manual, locked open valve and a check valve in series. The other line has two manual, normally open valves in series. The lines combine downstream of these valves in the Turbine Building and this common line splits to supply the two AFW pumps in the auxiliary building through a manual, normally open valve and a check valve in series in each pump supply line.

The AFW system status is monitored continuously in the Control Room by the Engineered Safeguards Display Monitor.

The Technical Specifications require monthly system operation surveillance which automatically test the suction lineup for the pumps. Due to the redundant nature of the normal supply lines the recommended Technical Specification change is not required.

1.3 NRC Recommendation GS-3

The licensee has stated that it throttles AFW system flow to avoid water hammer. The licensee should reexamine the practice of throttling AFW system flow to avoid water hammer.

The licensee should verify that the AFW system will supply on demand sufficient initial flow to the necessary steam generators to assure adequate decay heat removal following a loss of main feedwater flow and a reactor trip from 100% power. In cases where this reevaluation results in an increase in initial AFW system flow, the licensee should provide sufficient information to demonstrate that the required initial AFW system flow will not result in plant damage due to water hammer.

Response

It is not necessary to throttle the AFW flow to avoid water hammer in the feedwater lines.

The AFW flow is directed to an upper nozzle in the steam generator. This line is separate from the main feedwater nozzle. The line utilizes a tempering flow line which maintains a minimum flow to prevent water hammer due to admission of unheated feedwater.

1.4 NRC Recommendation GS-4

Emergency procedures for transferring to alternate sources of AFW supply should be available to the plant operators. These procedures should include criteria to inform the operators when, and in what order, the transfer to alternate water sources should take place. The following cases should be covered by the procedures:

(1) The case in which the primary water supply is not initially available. The procedures for this case should include any operator actions required to protect the AFW system pumps against self-damage before water flow is initiated.

(2) The case in which the primary water supply is being depleted. The procedure for this case should provide for transfer to the alternate water sources prior to draining of the primary water supply.

Response

The alternate source of AFW supply is the Essential Service Water System which is supplied to each pump through separate suction lines with two motor operated valves in series in each line. These valves open automatically upon a low pressure at the suction of the AFW pumps in conjunction with a low low steam generator level signal.

The AFW pumps stop on a low suction pressure signal. They restart automatically when suction pressure becomes available. A low suction alarm is initiated prior to the change over to warn operator prior to automatic change over occurring.

In addition to this automatic action, operating procedures (normal and abnormal) outline specific manual actions which must be taken by the operators to ensure adequate water supply to the AFW system to address the cases listed in the NRC recommendation.

1.5 NRC Recommendation GS-5

The as-built plant should be capable of providing the required AFW flow for at least two hours from one AFW pump train, independent of any ac power source. If manual AFW system initiation or flow control is required following a complete loss of ac power, emergency procedures should be established for manually initiating and controlling the system under these conditions. Since the water for cooling of the lube oil for the turbine-driven pump bearings may be dependent on ac power, design or procedural changes shall be made to eliminate this dependency as soon as practicable. Until this is done, the emergency procedures should provide for an individual to be stationed at the turbine-driven pump in the event of the loss of all ac power to monitor pump bearing and/or lube oil temperatures. If necessary, this operator would operate the turbine-driven pump in an on-off mode until ac power is restored. Adequate lighting powered by direct current (dc) power sources and communications at local stations should also be provided if manual initiation and control of the AFW system is needed. (See Recommendation GL-3 for the longer term resolution of this concern.)

Response

The as-built system at Byron and Braidwood Stations is capable of supplying the required AFW flow to the steam generators for two hours independent of any ac power source.

1.6 NRC Recommendation GS-6

The licensee should confirm flow path availability of an AFW system flow train that has been out of service to perform periodic testing or maintenance as follows:

(1) Procedures should be implemented to require an operator to determine that the AFW system valves are properly aligned and a second operator to independently verify that the valves are properly aligned.

(2) The licensee should propose Technical Specifications to assure that, prior to plant startup following an extended cold shutdown, a flow test would be performed to verify the normal flow path from the primary AFW system water source to the steam generators. The flow test should be conducted with AFW system valves in their normal alignment.

Response

(1) Administrative procedures are established that address the removal of equipment from service, the restoration of equipment to service and the realignment of equipment to normal service.

(2) The AFW system must be demonstrated to be operable prior to leaving Operational Mode 4 (Hot Shutdown). This requires a surveillance test that demonstrates flow to the steam generators. This requirement will be made part of the technical specifications.

1.7 NRC Recommendation GS-7

The licensee should verify that the automatic start AFW system signals and associated circuitry are safety-grade. If this cannot be verified, the AFW system automatic initiation system should be modified in the short-term to meet the functions requirements listed below. For the longer-term, the automatic initiation signals and circuits should be upgraded to meet safety grade requirements, as indicated in Recommendation GL-5.

(1) The design should provide for the automatic initiation of the AFW system flow.

(2) The automatic initiation signals and circuits should be designed so that a single failure will not result in the loss of AFW system function.

(3) Testability of the initiation signals and circuits shall be a feature of the design.

(4) The initiation signals and circuits should be powered from the emergency buses.

(5) Manual capability to initiate the AFW system from the control room should be retained and should be implemented so that a single failure in the manual circuits will not result in the loss of system function.

(6) The ac motor-driven pumps and valves in the AFW system should be included in the automatic actuation (simultaneous and/or sequential) of the loads to the emergency buses.

(7) The automatic initiation signals and circuits shall be designed so that their failure will

not result in the loss of manual capability to initiate the AFW system from the control room.

Response

The automatic initiation signals to the AFW system and the associated circuitry are safety-grade.

1.8 NRC Recommendation GS-8

The licensee should install a system to automatically initiate AFW system flow. This system need not be safety-grade; however, in the short-term, it should meet the criteria listed below, which are similar to Item 2.1.7 (a) of NUREG-0578. For the longer-term, the automatic initiation signals and circuits should be upgraded to meet safety-grade requirements, as indicated in recommendation GL-1.

(1) The design should provide for the automatic initiation of the AFW system flow.

(2) The automatic initiation signals and circuits should be designed so that a single failure will not result in the loss of AFW system function.

(3) Testability of the initiating signals and circuits should be a feature of the design.

(4) The initiating signals and circuits should be powered from the emergency buses.

(5) Manual capability to initiate the AFW system from the control room should be retained and should be implemented so that a single failure in the manual circuits will not result in the loss of system function.

(6) The ac motor-driven pumps and valves in the AFW system should be included in the automatic actuation (simultaneous and/or sequential) of the loads to the emergency buses.

(7) The automatic initiation signals and circuits should be designed so that their failure will not result in the loss of manual capability to initiate the AFW system from the control room.

Response

The design of the AFW system meets the criteria listed above.

Section 2: Response to Additional Short-Term Recommendations

2.1 NRC Recommendation

The licensee should provide redundant level indication and low level alarms in the control room for the AFW system primary water supply, to allow the operator to anticipate the need to make up water or transfer to an alternate water supply and prevent a low pump suction pressure condition from occurring. The low level alarm setpoint should allow at least 20 minutes for operator action, assuming that the largest capacity AFW pump is operating.

Response

Each auxiliary feedwater pump suction is equipped with a low suction pressure alarm. The alarm setpoint is sufficiently high to allow 20 minutes for the operator to take action prior to the pressure setpoint at which automatic suction switchover occurs if the low pressure condition is a result of low condensate storage tank level. Since the pumps are ultimately supplied by a common header from the condensate storage tank, this scheme provides a redundant indication of low condensate storage tank level. Additionally, the condensate storage tank is equipped with level indication in the control room and an associated low level alarm.

2.2 NRC Recommendation

The licensee should perform a 72 hour endurance test on all AFW system pumps, if such a test or continuous period of operation has not been accomplished to date. Following the 72 hour pump run, the pumps should be shutdown and cooled down and then restarted and run for one hour. Test acceptance criteria should include demonstrating that the pumps remain within design limits with respect to bearing/bearing oil temperatures and vibration and that pump room ambient conditions (temperature, humidity) do not exceed environmental qualification limits for safety-related equipment in the room.

Response

The recommended endurance test will be performed during the preoperation/startup testing period and will be documented.

2.3 NRC Recommendation

The licensee should implement the following requirements as specified by Item 2.1.7.b on page A-32 of NUREG-0578:

(1) Safety-grade indication of AFW flow to each steam generator should be provided in the control room.

(2) The AFW flow instrument channels should be powered from the emergency buses consistent with satisfying the emergency power diversity requirements for the AFW system set forth in Auxiliary Systems Branch Technical Position 10-1 of the Standard Review Plan, Section 10.4.9.

Response

The AFW flow to each steam generator is indicated in the control room, meets safety grade requirements and is powered from the ESF buses.

2.4 NRC Recommendation

Licensees with plants which require local manual realignment of valves to conduct periodic tests on one AFW system train and which have only one remaining AFW train available for operation should propose Technical Specifications to provide that a dedicated individual who is in communication with the control room be stationed at the manual valves. Upon instruction from the control room, this operator would re-align the valves in the AFW system from the test mode to its operational alignment.

Response

Local manual realignment of valves is not necessary to conduct periodic tests on the AFW system.

Section 3: Response to Long-Term Recommendations

3.1 NRC Recommendation GL-1

For plants with a manual starting AFW system, the licensee should install a system to automatically initiate the AFW system flow. This system and associated automatic initiation signals should be designed and installed to meet safety-grade requirements. Manual AFW system start and control capability should be retained with manual start serving as backup to automatic AFW system initiation.

Response

AFW system flow is automatically initiated.

3.2 NRC Recommendation GL-2

Licensees with plant designs in which all (primary and alternate) water supplies to the AFW system pass through valves in a single flow path should install redundant parallel flow paths (piping and valves).

Licensees with plant designs in which the primary AFW system water supply passes through valves in a single flow path, but the alternate AFW system water supplies connect to the AFW system pump suction piping downstream of the above valve(s), should install redundant valves parallel to the above valve(s) or provide automatic opening of the valve(s) from the alternate water supply upon low pump suction pressure.

The licensee should propose Technical Specifications to incorporate appropriate periodic inspections to verify the valve positions into the surveillance requirements.

Response

The ESW system supply connects downstream of the condensate supply valves. The valves automatically open on low pump suction pressure and low low steam generator level.

Periodic testing of the AFW systems is required by the Technical Specifications. This testing includes verification of valve positions.

3.3 NRC Recommendation GL-3

At least one AFW system pump and its associated flow path and essential instrumentation should automatically initiate AFW system flow and be capable of being operated independently of any ac power source for at least two hours. Conversion of dc power to ac power is acceptable.

Response

The diesel driven AFW system pump, its flow path and its essential instrumentation are capable of being operated independently of any ac power source for two hours.

3.4 NRC Recommendation GL-4

Licensees having plants with unprotected normal AFW system water supplies should evaluate the design of their AFW systems to determine if automatic protection of the pumps is necessary following a seismic event or a tornado. The time available before pump damage, the alarms and indications available to the control room operator, and the time necessary for assessing the problem and taking action should be considered in determining whether operator action can be relied on to prevent pump damage. Consideration should be given to providing pump protection by means such as automatic switchover of the pump suctions to the alternate safety-grade source of water, automatic pump trips on low suction pressure, or upgrading the normal source of water to meet seismic Category I and tornado protection requirements.

Response

The AFW pump circuitry is designed to trip the pumps on low suction pressure. Additionally, automatic pump suction switchover and automatic starting of the AFW pumps is provided.

3.5 NRC Recommendation GL-5

The licensee should upgrade the AFW system automatic initiation signals and circuits to meet safety-grade requirements.

Response

The automatic initiation signals and circuitry for the AFW system are safety grade.

Part IV: Evaluation of the Design Basis of the AFW System (ESF Pump Trains Only)

Question 1

a. Identify the plant transient and accident conditions considered in establishing AFWS flow requirements, including the following events:

1) Loss of Main Feed (LMFW)
2) LMFW 1/loss of offsite AC power
3) LMFW 1/loss of onsite and offsite AC power
4) Plant cooldown
5) Turbine trip with and without bypass
6) Main steam isolation valve closure
7) Main feedline break
8) Main steamline break
9) Small break LOCA
10) Other transient or accident conditions not listed above.

b. Describe the plant protection acceptance criteria and corresponding technical bases used for each initiating event identified above. The acceptance criteria should address plant limits such as:

1) Maximum RCS pressure (PORV or safety valve actuation)
2) Fuel temperature or damage limits (DNB, PCT, maximum fuel central temperature)
3) RCS cooling rate limit to avoid excessive coolant shrinkage
4) Minimum steam generator level to assure sufficient steam generator heat transfer surface to remove decay heat and/or cool down the primary system.

Response to 1.a

The Auxiliary Feedwater System serves as a backup system for supplying feedwater to the secondary side of the steam generators at times when the feedwater system is not available, thereby maintaining the heat sink capabilities of the steam generator. As an Engineered Safeguards System, the Auxiliary Feedwater System is directly relied upon to prevent core damage and system overpressurization in the event of transients such as a loss of normal feedwater or a secondary system pipe rupture, and to provide a means for plant cooldown following any plant transient.

Following a reactor trip, decay heat is dissipated by evaporating water in the steam generators and venting the generated steam either to the condensers through the steam dump or to the atmosphere through the steam generator safety valves

or the power-operated relief valves. Steam generator water inventory must be maintained at a level sufficient to ensure adequate heat transfer and continuation of the decay heat removal process. The water level is maintained under these circumstances by the Auxiliary Feedwater System which delivers an emergency water supply to the steam generators. The Auxiliary Feedwater System must be capable of functioning for extended periods, allowing time either to restore normal feedwater flow or to proceed with an orderly cooldown of the plant to the reactor coolant temperature where the Residual Heat Removal System can assume the burden of decay heat removal. The Auxiliary Feedwater System flow and the emergency water supply capacity must be sufficient to remove core decay heat, reactor coolant pump heat, and sensible heat during the plant cooldown. The Auxiliary Feedwater System can also be used to maintain the steam generator water levels above the tubes following a LOCA. In the latter function, the water head in the steam generators serves as a barrier to prevent leakage of fission products from the Reactor Coolant System into the secondary plant.

DESIGN CONDITIONS

The reactor plant conditions which impose safety-related performance requirements on the design of the Auxiliary Feedwater System are as follows for the Byron/Braidwood units.

- Loss of Main Feedwater Transient
  - Loss of main feedwater with offsite power available
  - Station blackout (i.e., loss of main feedwater without offsite power available)
- Secondary System Pipe Ruptures
  - Feedline rupture
  - Steamline rupture
- Loss of all AC Power
- Loss-of-Coolant Accident (LOCA)
- Cooldown

Loss of Main Feedwater Transients

The design loss of main feedwater transients are those caused by:

- Interruptions of the Main Feedwater System flow due to a malfunction in the feedwater or condensate system.
- Loss of offsite power or blackout with the consequential shutdown of the system pumps, auxiliaries, and controls.

Loss of main feedwater transients are characterized by a reduction in steam generator water levels which results in a reactor trip, a turbine trip, and auxiliary feedwater actuation by the protection system logic. Following reactor trip from a high initial power level, the power quickly falls to decay heat levels. The water levels continue to decrease, progressively uncovering the steam generator tubes as decay heat is transferred and discharged in the form of steam either through the steam dump valves to the condenser or through the steam generator safety or power-operated relief valves to the atmosphere. The reactor coolant temperature increases as the residual heat in excess of that dissipated through the steam generators is absorbed. With increased temperature, the volume of reactor coolant expands and begins filling the pressurizer. Without the addition of sufficient auxiliary feedwater, further expansion will result in water being discharged through the pressurizer safety and/or relief valves. If the temperature rise and the resulting volumetric expansion of the primary coolant are permitted to continue, then (1) pressurizer safety valve capacities may be exceeded causing overpressurization of the Reactor Coolant System, and/or (2) the continuing loss of fluid from the primary coolant system may result in bulk boiling in the Reactor Coolant System and eventually in core uncovering, loss of natural circulation, and core damage. If such a situation were ever to occur, the Emergency Core Cooling System would be ineffectual because the primary coolant system pressure exceeds the shutoff head of the safety injection pumps, the nitrogen overpressure in the accumlator tanks, and the design pressure of the Residual Heat Removal Loop. Hence, the timely introduction of sufficient auxiliary feedwater is necessary to arrest the decrease in the steam generator water levels, to reverse the rise in reactor coolant temperature, to prevent the pressurizer from filling to a water solid condition, and eventually to establish stable hot standby conditions. Subsequently, a decision may be made to proceed with plant cooldown if the problem cannot be satisfactorily corrected.

The blackout transient differs from a simple loss of main feddwater in that emergency power sources must be relied upon to operate vital equipment. The loss of power to the electric driven condenser circulating water pumps results in a loss of condenser vacuum and condenser dump valves. Hence, steam formed by decay heat is relieved through the steam generator safety valves or the power-operated relief valves. The calculated transient is similar for both the loss of main feedwater and the blackout, except that reactor coolant pump heat input is not a consideration in the blackout transient following loss of power to the reactor coolant pump bus.

The station blackout transient serves as the basis for the minimum flow required for the smallest capacity single auxiliary feedwater pump for the Byron/Braidwood units. The pump is sized so that it will provide sufficient flow against the steam generator safety valve set pressure (with 3% accumulation) to prevent water relief from the pressurizer. The same criterion is met for the loss of feedwater transient where ac power is available.

Secondary System Pipe Ruptures

The feedwater line rupture accident not only results in the loss of feedwater flow to the steam generators but also results in the complete blowdown of one steam generator within a short time if the rupture should occur downstream of the last non-return valve in the main or auxiliary feedwater piping to an individual steam generator. Another significant result of a feedline rupture may be the spilling of auxiliary feedwater to the faulted steam generator. Such situations can result in the injection of a disproportionately large fraction of the total auxiliary feedwater flow (the system preferentially pumps water to the lowest pressure region) to the faulted loop rather than to the effective steam generators which are at relatively high pressure. The system design must allow for terminating, limiting, or minimizing that fraction of auxiliary feedwater flow which is delivered to a faulted loop or spilled through a break in order to ensure that sufficient flow will be delivered to the remaining effective steam generator(s). The concerns are similar for the main feedwater line rupture as those explained for the loss of main feedwater transients.

Main steamline rupture accident conditions are characterized initially by plant cooldown and, for breaks inside containment, by increasing containment pressure and temperature. Auxiliary feedwater is not needed during the early phase of the transient but flow to the faulted loop will contribute to an excessive release of mass and energy to containment. Thus, steamline rupture conditions establish the upper limit on auxiliary feedwater flow delivered to a faulted loop. Eventually, however, the Reactor Coolant System will heat up again and auxiliary feedwater flow will be required to be delivered to the non-faulted loops, but at somewhat lower rates than for the loss of feedwater transients described previously. Provisions must be made in the design of the Auxiliary Feedwater System to limit, control, or terminate the auxiliary feedwater flow to the faulted loop as necessary in order to prevent containment overpressurization following a steamline break inside containment, and to ensure the minimum flow to the remaining unfaulted loops.

Loss of All AC Power

The loss of all ac power is postulated as resulting from accident conditions wherein not only onsite and offsite ac power is lost but also ac emergency power is lost as an assumed common mode failure. Battery power for operation of protection circuits is assumed available. The impact on the Auxiliary Feedwater System is the necessity for providing both an auxiliary feedwater pump power and control source which are not dependent on ac power and which are capable of maintaining the plant at hot shutdown until ac power is restored.

Loss-of-Coolant Accident (LOCA)

The loss-of-coolant accidents do not impose on the auxiliary feedwater system any flow requirements in addition to those required by the other accidents addressed in this response. The following description of the small LOCA is provided here for the sake of completeness to explain the role of the auxiliary feedwater system in this transient.

Small LOCA's are characterized by relatively slow rates of decrease in reactor coolant system pressure and liquid volume. The principal contribution from the Auxiliary Feedwater System following such small LOCAs is basically the same as the system's function during hot shutdown or following spurious safety injection signal which trips the reactor. Maintaining a water level inventory in the secondary side of the steam generators provides a heat sink for removing decay heat and establishes the capability for providing a buoyancy head for natural circulation. The auxiliary feedwater system may be utilized to assist in a system cooldown and depressurization following a small LOCA while bringing the reactor to a cold shutdown condition.

Cooldown

The cooldown function performed by the Auxiliary Feedwater System is a partial one since the reactor coolant system is reduced from normal zero load temperatures to a hot leg temperature of approximately 350° F. The latter is the maximum temperature recommended for placing the Residual Heat Removal System (RHRS) into service. The RHR system completes the cooldown to cold shutdown conditions.

Cooldown may be required following expected transients, following an accident such as a main feedline break, or during a normal cooldown prior to refueling or performing reactor plant maintenance. If the reactor is tripped following extended operation at rated power level, the AFWS is capable

of delivering sufficient AFW to remove decay heat and reactor coolant pump (RCP) heat following reactor trip while maintaining the steam generator (SG) water level. Following transients or accidents, the recommended cooldown rate is consistent with expected needs and at the same time does not impose additional requirements on the capacities of the auxiliary feedwater pumps, considering a single failure. In any event, the process consists of being able to dissipate plant sensible heat in addition to the decay heat produced by the reactor core.

Response to 1.b

Table Q10.53-1 summarizes the criteria which are the general design bases for each event, discussed in the response to Question 1.a, above. Specific assumptions used in the analyses to verify that the design bases are met are discussed in response to Question 2.

The primary function of the Auxiliary Feedwater System is to provide sufficient heat removal capability following reactor trip and to remove the decay heat generated by the core and prevent system overpressurization. Other plant protection systems are designed to meet short term or pre-trip fuel failure criteria. The effects of excessive coolant shrinkage are evaluated by the analysis of the rupture of a main steam pipe transient. The maximum flow requirements determined by other bases are incorporated into this analysis, resulting in no additional flow requirements.

Question 2

Describe the analyses and assumptions and corresponding technical justification used with plant condition considered in 1.a above including:

a. Maximum reactor power (including instrument error allowance) at the time of the initiating transient or accident.

b. Time delay from initiating event to reactor trip.

c. Plant parameter(s) which initiates AFWS flow and time delay between initiating event and introduction of AFWS flow into steam generator(s).

d. Minimum steam generator water level when initiating event occurs.

e. Initial steam generator water inventory and depletion rate before and after AFWS flow commences -- identify reactor decay heat rate used.

f. Maximum pressure at which steam is released from steam generator(s) and against which the AFW pump must develop sufficient head.

g. Minimum number of steam generators that must receive AFW flow; e.g., 1 out of 2? 2 out of 4?

h. RC flow condition -- continued operation of RC pumps or natural circulation.

i. Maximum AFW inlet temperature.

j. Following a postulated steam or feedline break, time delay assumed to isolate break and direct AFW flow to intact steam generator(s). AFW pump flow capacity allowance to accommodate the time delay and maintain minimum steam generator water level. Also identify credit taken for primary system heat removal due to blowdown.

k. Volume and maximum temperature of water in main feedlines between steam generator(s) and AFWS connection to main feedline.

l. Operating condition of steam generator normal blowdown following initiating event.

m. Primary and secondary system water and metal sensible heat used for cooldown and AFW flow sizing.

n. Time at hot standby and time to cooldown RCS to RHR system cut in temperature to size AFW water source inventory.

Response to 2

Analyses have been performed for the limiting transients which define the AFWS performance requirements. These analyses have been provided for review in the Applicant's FSAR. Specifically, they include:

- Loss of Main Feedwater (Station Blackout)
- Rupture of a Main Feedwater Pipe
- Rupture of a Main Steam Pipe Inside Containment

In addition to the above analyses, calculations have been performed specifically for the Byron/Braidwood units to determine the plant cooldown flow (storage capacity) requirements. The Loss of All AC Power is evaluated via a comparison to the transient results of Blackout, assuming an available auxiliary pump having a diverse (non-AC) power supply. The LOCA analysis, as discussed in response 1.b, incorporates the systems flow requirements as defined by other transients, and therefore is not performed for the purpose of specifying AFWS flow requirements. Each of the analyses listed above are explained in further detail in the following sections of this response.

Loss of Main Feedwater (Blackout)

A loss of feedwater, assuming a loss of power to the reactor coolant pumps, was performed in FSAR Subsection 15.2.7 for the purpose of showing that for a station blackout transient the peak RCS pressure remains below the criterion for Condition II transients and no fuel failures occur (refer to Table Q10.53-1). Table Q10.53-2 summarizes the assumptions used in this analysis. The analysis assumes that the plant is initially operating at 102% (calorimetric error) of the Engineered Safeguards design (ESD) rating shown on the table, a very conservative assumption in defining decay heat and stored energy in the RCS. The reactor is assumed to be tripped on low-low steam generator water level, allowing for level uncertainty. The FSAR shows that there is a considerable margin with respect to filling the pressurizer.

This analysis establishes the capacity of the smallest single pump and also establishes train association of equipment so that this analysis remains valid assuming the most limiting single failure.

Rupture of Main Feedwater Pipe

The double ended rupture of a main feedwater pipe downstream of the main feedwater line check valve is analyzed in FSAR, Subsection 15.2.8. Table Q10.53-2 summarizes the assumptions used in this analysis. Reactor trip is assumed to occur when the faulted steam generator is at the low-low level setpoint (adjusted for errors). This conservative assumption maximizes the stored heat prior to reactor trip and minimizes the ability of the steam generator to remove heat from the RCS following reactor trip due to a conservatively small total steam generator inventory. As in the loss of normal feedwater analysis, the initial power rating was assumed to be 102% of the ESD rating. Total auxiliary feedwater flow of 459 gpm was assumed to be delivered to the three non-faulted steam generators 1 minute after reactor trip. No operator action is assumed until an alternate source of water supply for the auxiliary feedwater supply is needed. The criteria listed in Table Q10.53-1 are met.

This analysis establishes the capacity of single pumps, establishes requirements for layout to preclude indefinite loss of auxiliary feedwater to the postulated break, and establishes train association requirements for equipment so that the AFWS can deliver the minimum flow required in 1 minute assuming the worst single failure.

Rupture of a Main Steam Pipe Inside Containment

Because the steamline break transient is a cooldown, the AFWS is not needed to remove heat in the short term. Furthermore, addition of excessive auxiliary feedwater to the faulted steam generator will affect the peak containment pressure following a steamline break inside containment. This transient is performed at four power levels for several break sizes. Auxiliary feedwater is assumed to be initiated at the time of the break, independent of system actuation signals. The maximum flow is used for this analysis. Table Q10.53-2 summarizes the assumptions used in this analysis. At 10 minutes after the break, it is assumed that the operator has isolated the AFWS from the faulted steam generator which subsequently blows down to ambient pressure. The criteria stated in Table Q10.53-1 are met.

This transient establishes the maximum allowable auxiliary feedwater flow rate to a single faulted steam generator assuming all pumps operating, establishes the basis for runout protection, if needed, and establishes layout requirements so that the flow requirements may be met considering the worst single failure.

Plant Cooldown

Maximum and minimum flow requirements from the previously discussed transients meet the flow requirements of plant cooldown. This operation, however, defines the basis for tankage size, based on the required cooldown duration, maximum decay heat input and maximum stored heat in the system. As previously discussed in response 1.a., the auxiliary feedwater system partially cools the system to the point where the RHRS may complete the cooldown, i.e., 350° F in the RCS. Table Q10.53-2 shows the assumptions used to determine the cooldown heat capacity of the auxiliary feedwater system.

The cooldown is assumed to commence at the maximum ESD power, and maximum trip delays and decay heat source terms are assumed when the reactor is tripped. Primary metal, primary water, secondary system metal and secondary system water are all included in the stored heat to be removed by the AFWS. See Table Q10.53-3 for the items constituting the sensible heat stored in the NSSS.

This operation is analyzed to establish minimum tank size requirements for auxiliary feedwater fluid source which are normally aligned.

Question 3

Verify that the AFW pumps in your plant will supply the necessary flow to the steam generator(s) as determined by items 1 and 2 above considering a single failure. Identify the margin in sizing the pump flow to allow for pump recirculation flow, seal leakage and pump wear.

Response 3

See Table Q10.53-4.

TABLE Q10.53-1

CRITERIA FOR AUXILIARY FEEDWATER SYSTEM DESIGN BASIS CONDITIONS

CONDITION OR TRANSIENT	CLASSIFICAITON*	CRITERIA	ADDITIONAL DESIGN CRITERIA
Loss of Main Feedwater	Condition II	Peak RCS pressure not to exceed design pressure + 10%. No consequential fuel failrues	
Station Blackout	Condition II	(same as LMFW)	Pressurizer does not fill 1 single motor driven auxiliary feed pump feeding 2 SGs.
Feedline Rupture	Condition IV	10 CFR 100 dose limits. Containment design pressure not exceeded	Core does not uncover
Loss of all A/C Power	N/A	Note 1	Same as blackout
Steamline Rupture	Condition IV	Containment design pressure not exceeded 100 CFR 100 dose limits.	
Loss-of-Coolant	Condition III	10 CFR 100 dose limits 10 CFR 50 PCT limits	
	Condition IV	10 CFR 100 dose limits 10 CFR 50 PCT limits	
Cooldown	N/A		100°F/hr 557° F to 350° F

*Ref: ANSI N18.2 (This information provided for those transients performed in the FSAR.)

Note 1 Although this transient establishes the basis for AFW pump powered by a diverse power source, this is not evaluated relative to typical criteria since multiple failures must be assumed to postulate this transient.

TABLE Q10.53-2

SUMMARY OF ASSUMPTIONS USED IN AFWS DESIGN VERIFICATION ANALYSES

Transient	Loss of Feedwater (station blackout)	Cooldown	Main Feedline Break	Main Steamline Break (containment)
a. Max reactor power	102% of ESD rating (102% of 3579 MWt)	3651 MWt	102% of ESD rating (102% of 3579 MWt)	0, 30, 70, 102% of rated (percent of 3425 MWt)
b. Time delay from event to rod-motion	29.3	2 sec	19	Variable
c. AFWS actuation signal/time delay for AFWS flow	lo-lo SG level/ 1 minute	N/A	low-low SG level/ 1 minute	Assumed immediately @ 0 sec (no delay)
d. SG water level at time of reactor trip	low-low SG level/ 37.3% NR span	N/A	(low-low SG level) 37.3% NR span	N/A
e. Initial SG inventory	59,400 lbm* 99,000 lbm/SG	65,074 lbm at 543.3°F	49,500 lbm* 88,830 lbm faulted intact 78,570 lbm	Consistent with power
• Rate of change before & after AFWS actuation	See FSAR Figure 15.2-10 and attached Figure 10.53.2-1	N/A	Turnaround at 4200 sec see attached Figure 10.53.2-2	N/A
Decay heat	ANS + 20%	N/A	ANS + 20%	ANS + 20%
f. AFW pump design pressure	1236 psia	1226 psia	1236	N/A
g. Minimum # of SGs which must receive AFW flow	3 of 4	N/A	3 of 4	N/A
h. RC pump status	Tripped at reactor trip	Tripped	All operating	All operating
i. Maximum AFW temperature	95°F	100°F	95°F	440°F
j. Operator action	none	N/A	Note 2	10 minutes
k. MFW purge volume/ S/G and temperature	100 ft^3/443°F	150 ft^3/ 440°F	100 ft^3/443°F	800 ft^3/loop (for dryout time)

TABLE Q10.53-2 (Continued)

SUMMARY OF ASSUMPTIONS USED IN AFWS DESIGN VERIFICATION ANALYSIS

Transient	Loss of Feedwater (station blackout)	Cooldown	Main Feedline Break	Main Steamline Break (containment)
l. Normal blowdown	none assumed	none assumed	none assumed	none assumed
m. Sensible heat	see cooldown	Table 2-2	see cooldown	N/A
n. Time at standby/time	2 hr/4 hrs	2 hr/4 hrs	2 hr/4 hrs	N/A
o. AFW flow rate	750 gpm - constant (min. requirement)	variable	459 gpm - constant	794 gpm (constant) to broken SG.

Note (1) assumed when alt. source of water is required for aux. feed.

* Low-low level mass adjusted for uncertainties.

10.53-35

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TABLE Q10.53-3

SUMMARY OF SENSIBLE HEAT SOURCES

Primary Water Sources (initially at ESD power temperature and inventory)

RCS fluid
Pressurizer fluid (liquid and vapor)

Primary Metal Sources (initially at ESD power temperature)

Reactor coolant piping, pumps and reactor vessel
Pressurizer
Steam generator tube metal and tube sheet
Steam generator metal below tube sheet
Reactor vessel internals

Secondary Water Sources (initially at ESD power temperature and inventory)

Steam generator fluid (liquid and vapor)
Main feedwater purge fluid between steam generator and AFWS piping

Secondary Metal Source (initially at ESD power temperature)

All steam generator above tube sheet, excluding tubes.

TABLE Q10.53-4

AUXILIARY FEEDWATER FLOW TO STEAM GENERATORS FOLLOWING AN ACCIDENT/TRANSIENT WITH SELECTED SINGLE FAILURE - GPM[1]

	ACCIDENT/TRANSIENT PUMP FAILURE	DIESEL FAILURE	MD PUMP FAILURE
1.	Loss of Main FW	160 gpm	160 gpm
2.	Feedline Rupture	160	160
3.	Blackout	214	214
4.	Cooldown	-	-
5.	Main Steamline Rupture (min)	165	165
6.	Main Steamline Rupture (max)	380	380

Notes:

(1) Items 1 through 5 are minimum expected flows to intact loops; item 6 is maximum possible flow to the faulted loop. With the maximum flow to the faulted steam generator, 650 gpm is available to be delivered to the intact steam generators. This exceeds the requirement of 160 gpm to two of the three remaining steam generators by a sufficient amount to allow full recirculation of the pump during the transient.

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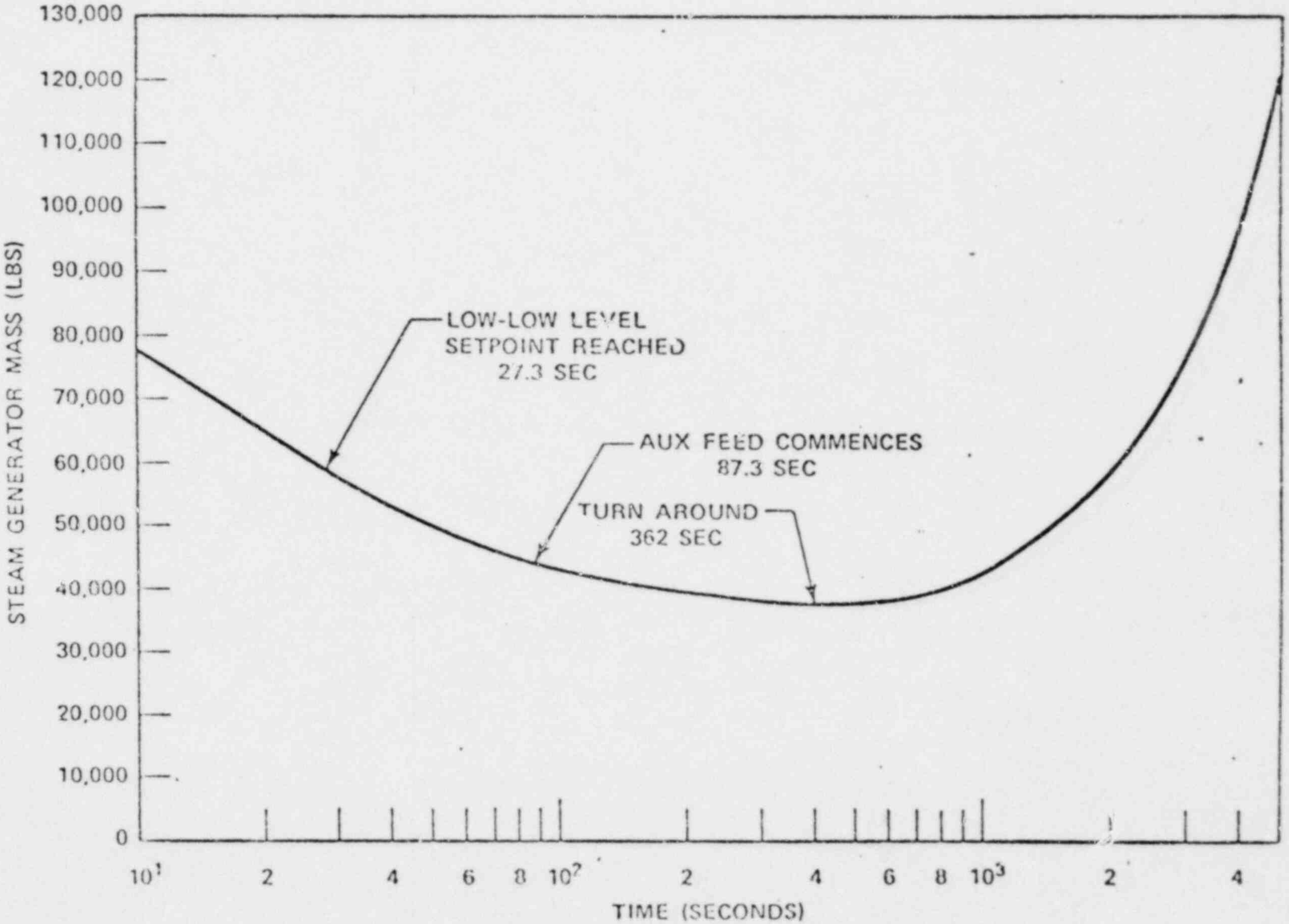
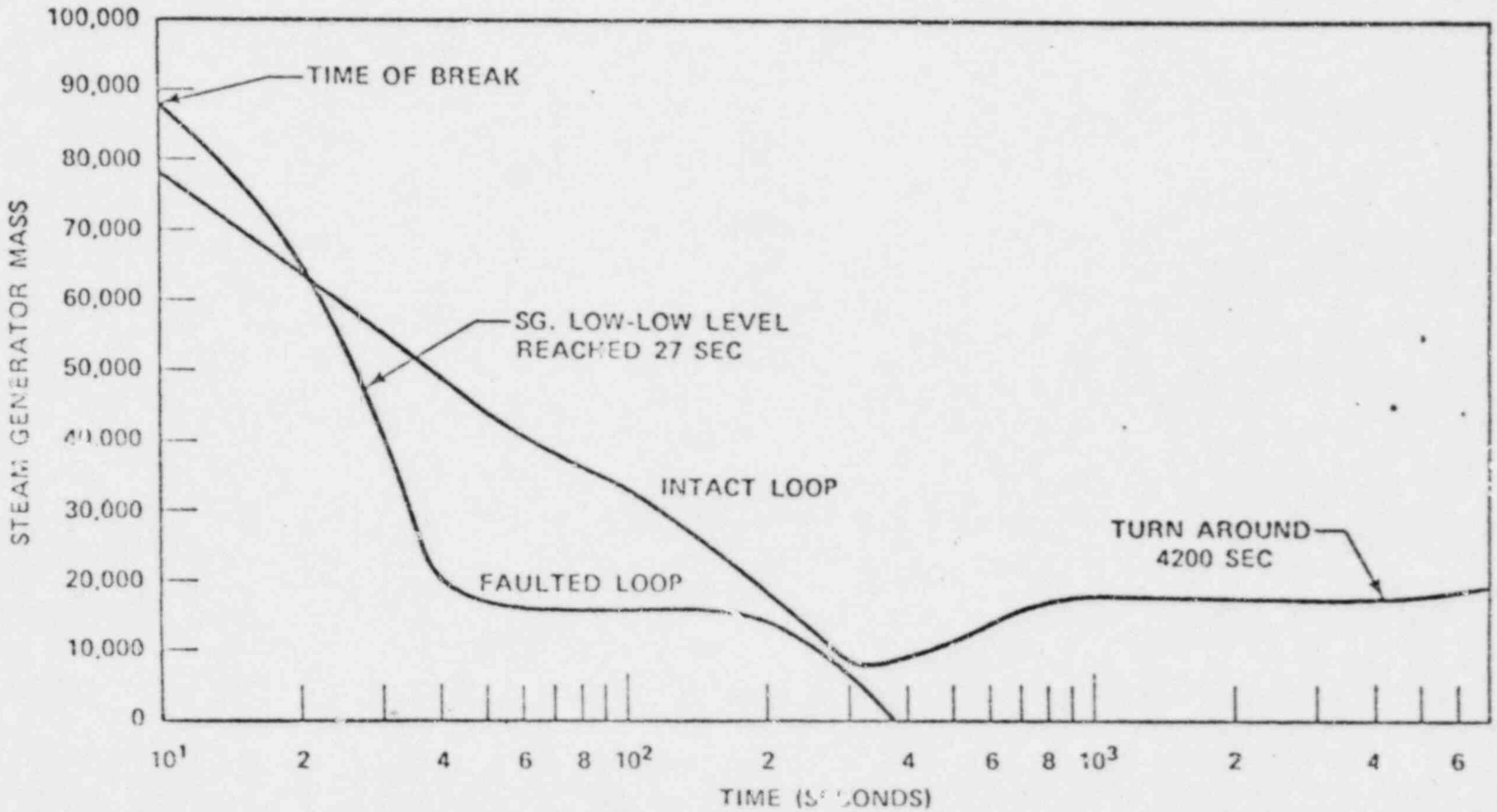


BYRON/BRAIDWOOD STATIONS
FINAL SAFETY ANALYSIS REPORT

Figure Q10.53-1
Steam Generator Mass vs Time Loss of Feedwater
Offsite Power Not Available
Loss of Feedwater at Time Zero

Q10.53-38

B/B-FSAR



Q10.53-39

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Figure Q10.53-2

Steam Generator Mass vs Time
Feedline Rupture Offsite Power Available

QUESTION 040.14

"Potential Problem with Containment Electrical Penetration Assemblies

"Recent operating experience at Millstone Unit No. 2 has shown that the deterioration of the epoxy insulation between splices has caused electrical shorts between conductors within a containment electrical penetration assembly. Indicate what tests and/or analysis that have been performed to demonstrate the acceptability of the design in this regard. Provide whatever information is required to perform an independent evaluation of this aspect of the electrical penetration design."

RESPONSE

It should be noted that the Byron/Braidwood Unit 1 electrical penetrations are being procured from Conax Corporation, whereas the Byron/Braidwood Unit 2 electrical penetrations are being procured from Bunker Ramo (Amphenol SAMS).

1. Conax Corporation Penetration Assemblies

   Conax does not utilize an epoxy insulation system in the design or manufacture of its containment electrical penetration assemblies.

   The Conax penetration design utilizes solid copper conductors which pass through the assembly without any internal splices. The conductors are continuously insulated with a polyimide (Kapton) film and mechanically sealed at both ends of the stainless steel tube (module) using thermoplastic (polysulfone) sealants. The modules are then mechanically sealed within the penetration's header plate.

   The penetration is designed with the provision of periodically monitoring the condition of all seals by internally pressurizing the assembly with nitrogen gas. Conax does not require the penetration assembly to be continuously pressurized during operation since the nitrogen gas does not contribute to the function of the penetration. Conax has performed extensive testing per IEEE-317 on similarly designed penetration assemblies (with and without nitrogen pressure) with satisfactory results.

   Therefore, the Conax penetration design is not susceptible to the failure experienced at Millstone Unit 2.

2. Bunker Ramo (Amphenol SAMS) Penetration Assemblies

The Bunker Ramo penetration assemblies are similar to those at Millstone Unit 2 in that they depend upon a glass epoxy sealant and a dry nitrogen pressure environment to ensure adequate functioning of electrical safety-related equipment and containment leaktightness. The penetration model (type) is identified by the manufacturer as a Unitized Header Assembly. The penetration module conductors (transition connector pins) embedded in the epoxy do not have an insulation jacket. The glass epoxy sealing material provides pin-to-pin insulation.

The manufacturer, however, has provided instructions so that the required dry nitrogen pressure will be maintained at all times during shipping, storage and installation of the electrical penetrations. This will prevent moisture from accumulating in the seals and the specified insulation resistance between adjacent conductors will be maintained to assure adequate functioning of electrical safety-related equipment.

Therefore, the Bunker Ramo penetration design is also acceptable in this regard.

QUESTION 040.143

"As explained in issue No. 1 of NUREG of 0138, credit is taken for all valves downstream of the Main Steam Isolation Valve (MSIV) to limit blowdown of a second steam generator in the event of a steam line break upstream of the MSIV. In order to confirm satisfactory performance following such a steam line break provide a tabulation and descriptive text (as appropriate) in the FSAR of all flow paths that branch off the main steam lines between the MISV's and the turbine stop valves. For each flow path originating at the main steam lines, provide the following information:

a) System identification

b) Maximum steam flow in pounds per hour

c) Type of shut-off valve(s)

d) Size of valve(s)

e) Quality of the valve(s)

f) Design code of the valve(s)

g) Closure time of the valve(s)

h) Actuation mechanism of the valve(s) (i.e., Solenoid operated, motor operated, air operated diagram valve, etc.)

i) Motive or power source for the valve actuating mechanism.

"In the event of the postulated accident, termination of steam flow from all systems identified above, except those that can be used for mitigation of the accident, is required to bring the reactor to a safe cold shutdown. For these systems describe what design features have been incorporated to assure closure of the steam shut-off valve(s). Describe what operator actions (if any) are required.

"If the systems that can be used for mitigation of the accident are not available or decision is made to use other means to shut down the reactor describe how these systems are secured to assure positive steam shut-off. Describe what operator actions (if any) are required.

"If any of the requested information is presently included in the FSAR text, provide only the references where the information may be found."

RESPONSE

The analysis of a main steamline break is presented in Subsection 15.1.5. The detailed description of main steamline isolation is in Chapter 10. Three lines branch off the main steam lines between the MSIV's and the turbine valves (ref. Figure 10.3-1).

A four inch line supplies approximately 16,000 lb/hr of steam to the gland steam system. A four inch motor-operated gate valve (GS001) is used for isolation. The valve is rated at 900 lbs. and is designed in accordance with ASTM standards.

This valve does not automatically close on a turbine trip but must be closed by the operator, if necessary.

A 28 inch line branches off each main steam header for the steam dump system and extraction to the second stage of the moisture separator reheater. The two 12 inch branch lines supply approximately 790,000 lb/hr. to the moisture separator reheater. Two 10 inch motor-operated gate valves (MS009A/B/C/D) on each moisture separator are used for isolation. The valves are rated at 900 lbs. and are designed per ASTM standards. The valves must be closed by the operator.

All of the valves are Category II, Quality Group D.

QUESTION 130.20

"With reference to your response to Question 130.8 relate the criteria used to ensure the adequate number of masses or degrees of freedom against those contained in the SRP, Section 3.7.2-II.1.a.(4). In your response quantitatively compare the two criteria and assess conservatism of the FSAR design."

RESPONSE

An adequate number of masses and degrees of freedom were considered in the dynamic modeling to determine the response of Category I and applicable non-Category I structures. The criteria used, as described in Question 130.8, is in compliance with SRP Section 3.7.2-II.1.a.(4).

Containment

The dynamic characteristics for the containment structure stick model indicate the number of dynamic degrees of freedom meet the guidelines set forth in the SRP Section 3.7.2-II.1.a.(4). In the horizontal model, there are four modes with frequencies less than 33 cps and 26 dynamic degrees of freedom. In the vertical model, there is one mode with a frequency less than 33 cps and 13 dynamic degrees of freedom.

Shear Structure System

The mass point locations for the shear structure system type models are described in Subsection 3.7.2.3.3. The behavior of a shear structure eliminates certain degrees of freedom. Due to the predominance of shear deformation expected for horizontal excitation of a structure with shear walls connected to rigid concrete slabs, vertical translation and rotations about horizontal axes will be negligible.

Similarly for the vertical model, only vertical translation is expected to exist for vertical excitation.

Additional degrees of freedom will not significantly affect the response of the shear structure. Additional mass joints located at midstory, for instance, for the sole purposes of obtaining more degrees of freedom will not affect the response because the mass of the shear wall is much less than the mass which would be lumped at the slab.

QUESTION 130.23

"Provide details such as connections and/or other protective measures to secure that the siding on turbine building will be blown off at the specified pressure of 105 psf. Also, describe the provisions to prevent that the blown off siding or other parts of the building become tornado generated missiles."

RESPONSE

Turbine building siding is composed of double span blow-in/blow-out panels consisting of face and liner sheets interconnected by subgirts. There are no fasteners used at the lap joints between the adjoining face sheets to permit blow-in or blow-out (see Figure Q130.23-1).

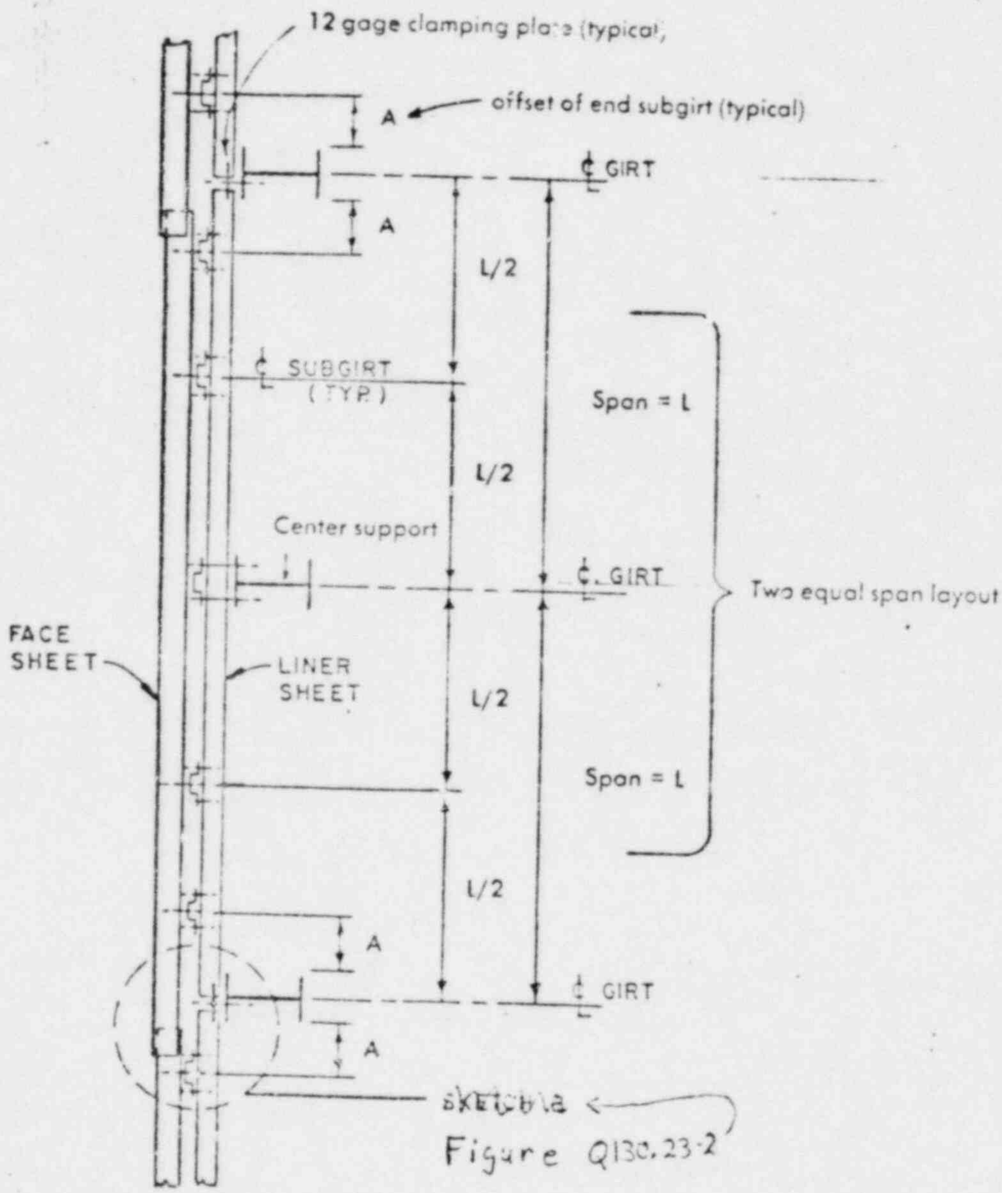
The liner sheet is fastened to the girt at center support with screws penetrating the subgirt. At panel end supports a 3-inch wide 12 gauge plate clamps down both liner ends. No screws penetrate liner sheet (see Figure Q130.23-2).

The siding failure mechanism, as described below, is shown in Figures Q130.23-3 and Q130.23-4.

Under wind pressure, because of large displacement, the liner ends slip out of the clamping plate and start to wrap around the center support or bend away from outside supports for inward or outward pressures, respectively. After the siding panel buckles, it remains anchored to the center support, bent either outward or inward about the center support.

Data provided by the siding manufacturer indicates that failure of the siding will take place at pressures considerably lower than the specified pressure of 105 psf.

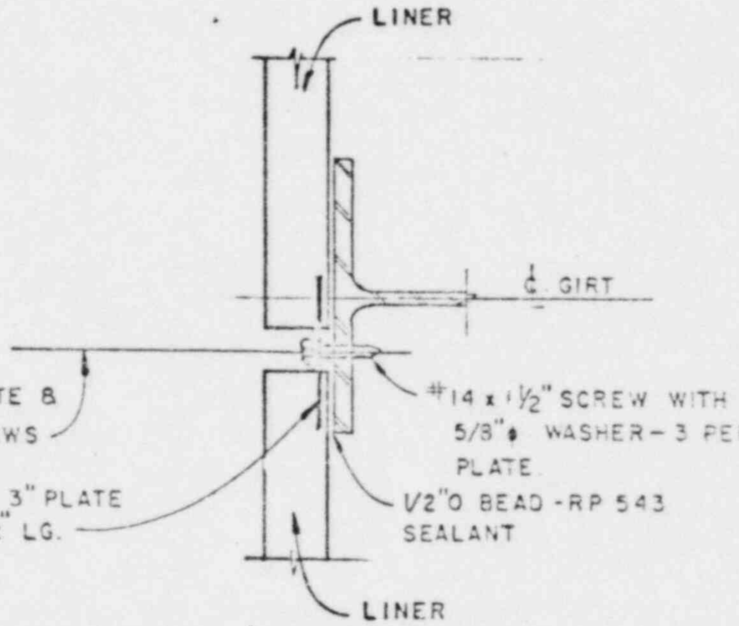
Girts, sag rods, doors, and windows are also permitted to fail. These items represent sources of potential tornado generated missiles, however, their effect on Category I structures will be less severe than the missiles defined in Subsection 3.5.1.4, for which Category I structures have been designed.



BYRON/BRAIDWOOD STATIONS
FSAR

FIGURE Q130.23-1
TURBINE BUILDING
SIDING

SKETCH A



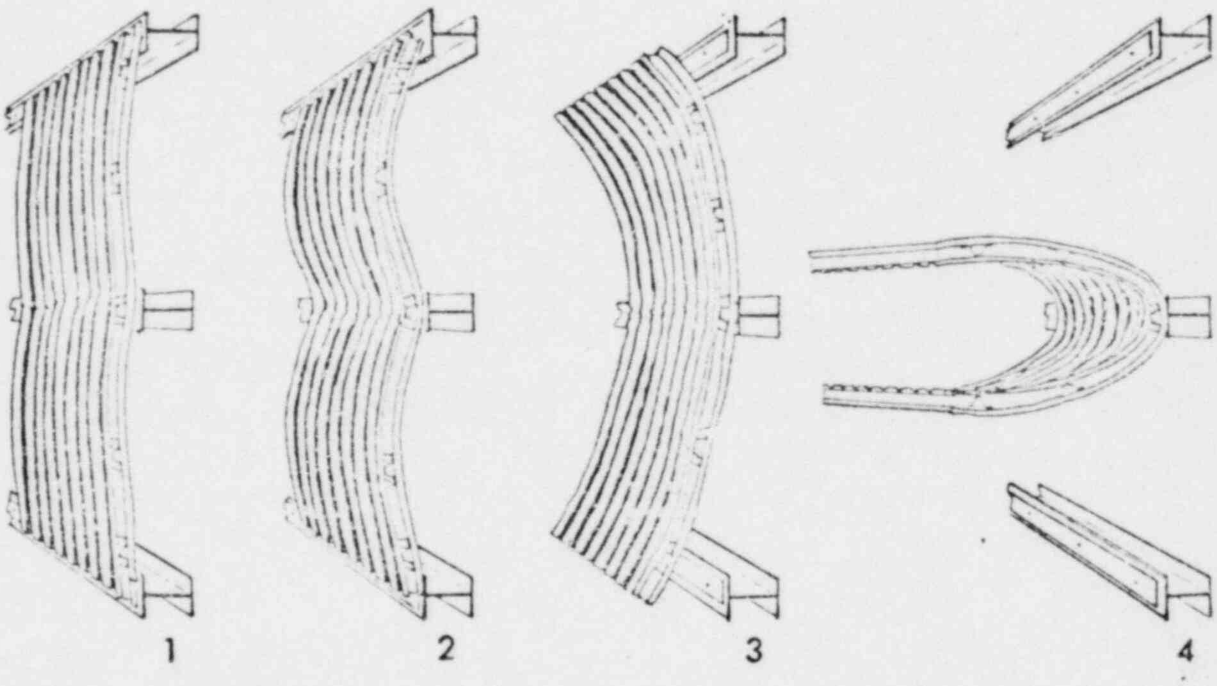
SKETCH B

BYRON/BRAIDWOOD STATION
FSAR

FIGURE Q130.2-1

TURBINE BUILDING
SIDING LINER SHEET
DETAIL

OUTWARD RELEASE MECHANISM



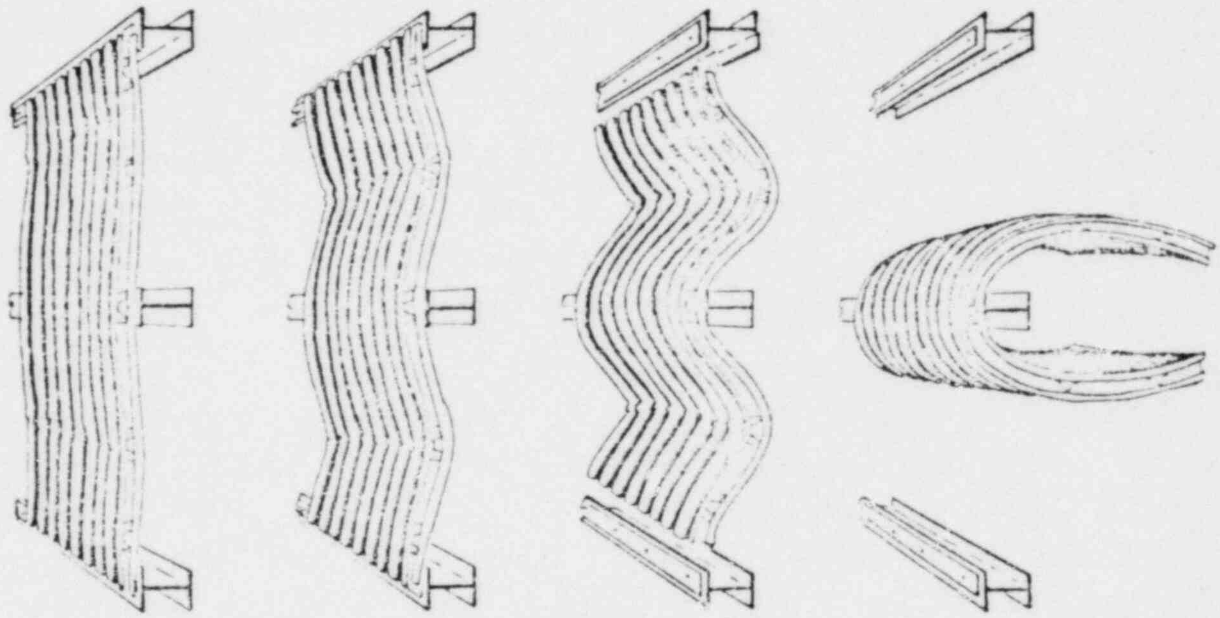
BYRON/BRAIDWOOD STATIONS
FSAR

FIGURE Q130.20-3

TURBINE BUILDING
SIDING OUTWARD
RELEASE MECHANISM

SKETCH C

INWARD RELEASE MECHANISM



SKETCH D

BYRON/BRAIDWOOD STATION
FSAR
FIGURE Q130.23-4
TURBINE Building SIDING
INWARD RELEASE
MECHANISM

QUESTION 130.25

"In FSAR Section 3.5.3 you indicated that the overall response of barriers to impactive and impulsive loads is based on FSAR Reference 7, 'Design of Missile Restraint Concrete Panels,' by J. M. Doyle, M. J. Klein and H. Shah. The method described in Reference 7 has not been reviewed and accepted by the Regulatory Staff. The method which is acceptable to the staff are specified in the SRP, Section 3.5.3. In order to ascertain that the method of design is sufficiently conservative and acceptable to the staff, you are requested to provide a comparison with the methods described in the SRP Section 3.5.3, by means of a design example or otherwise, to demonstrate conservativeness of the method selected for design of barriers."

RESPONSE

In Reference 7 of the FSAR, the maximum displacement, Y_m, of a structural element under the impact of a rigid missile of mass m is given by:

$$Y_m = \frac{m}{M} (D'^2 + (\frac{V_o}{\omega})^2)^{1/2} \quad (1)$$

where

M = equivalent mass of structural element

D' = penetration of missile into structure

C_o = missile velocity prior to impact

ω = circular frequency of structural element

The method specified in SRP Section 3.5.3 for treating this problem is given in Reference 1 cited at the end of this response.

Reference 1 specifies the equivalent static design load, F, as:

$$F = \frac{F_i}{K} \quad (2)$$

$$F_i = m \frac{V_o^2}{D'} \quad (3a)$$

$$t_i = \frac{2D'}{V_o} \quad (4)$$

$$K = \sqrt{\frac{2\mu - 1}{\pi t_1} T} + \frac{1 - \frac{0.5}{\mu}}{1 + \frac{0.7T}{t_1}} \qquad (5)$$

where

F_i = peak value of impact force developing between missile and target

t_1 = duration of impact

$T = \frac{2\pi}{\omega}$

μ = ductility ratio = $\frac{y_m}{\sigma_y}$

σ_y = static yield deflection, when element is idealized by an SDF dynamic system

It should be noted that in Equation (3a) above, the work of the missile on the target during the penetration process has not been evaluated consistently with the linear variation of contact force and velocity with time, which is the starting point of derivation in reference 1.

From the above assumptions, it follows that

$$F = F_i \left(1 - \frac{t}{t_1}\right)$$

$$V = V_o \left(1 - \frac{t}{t_1}\right)$$

$$\text{Impact Work} = \int_0^{t_1} F_i V_o \left(1 - \frac{t}{t_1}\right)^2 dt = \frac{1}{3} F_i V_o t_1$$

From Equation (4) $V_o t_1 = 2D'$

$$\text{Impact Work} = \frac{2}{3} F_i D' \qquad (6)$$

$$\text{Kinetic Energy of Missile} = 1/2\ m\ V_o^{\ 2} \qquad (7)$$

From Equations (6) & (7)

$$\frac{2}{3} F_i D' = \frac{1}{2} m\ V_o^{\ 2} \qquad (8)$$

When the evaluation is done consistently with these assumptions, Equation (2a) changes to Equation (3) by Equation (8).

$$F_i = \frac{3}{4} m \frac{V_o^2}{D'} \qquad (3)$$

The following examples are based on the use of Equation (3) To compare these two methods, a square reinforced concrete panel 26 ft long and 1.5 ft thick, with reinforcement details and material properties as summarized in Figure Q130.25-1 was considered. The results obtained by the two methods for the response of this panel to the impact of a rigid mass of the center are summarized below. The missile corresponds to the utility pole described in Table 3.5-3 of the FSAR, with

V_o = 241 ft/sec

$m = \frac{1490 \text{ lb sec}^2}{32.2 \text{ ft}}$

head-on contact area = 143.10 in^2

For this missile, the modified Petry formula with K_p = 0.0032 ft^3/lb* yields a missile penetration depth of D' = 5,976 inches. The panel is approximated as a Single Degree of Freedom System for response evaluation, with a mass of

$$M = 0.32 \left(\frac{\text{panel weight}}{g}\right)$$

where g = acceleration of gravity.

The period of the system is determined on the basis of a cracked section moment of inertia to be 0.0961 second. The yield force Q_y and the corresponding panel deflection σ_y are 358.5 kips and 0.680 inches, respectively. These values are based on replacement of the panel by an equivalent beam.**

---

* K_p as a function of concrete strength is obtained from the curve in Reference 2.

**The method for replacing the panel by an equivalent beam is described in Reference 7 of the FSAR.

The maximum displacement for Method 1, as outlined in FSAR Subsection 3.5.3, is obtained directly from Equation (1):

$$[y_{max}]_{method\ 1} = 1.37 \text{ in.}$$

To obtain the maximum displacement for Method 2 of SRP 3.5.3, trial and error procedure is used for Equations (2) through (5) to obtain the ductility parameter μ. This value is

$$[\mu]_{method\ 2} = 1.65$$

Therefore,

$$[y_{max}]_{method\ 2} = 1.65 \times 0.680 = 1.12 \text{ inches.}$$

The maximum displacement predicted by Method 1 of the FSAR exceeds 22% of the value obtained by corrected Method 2 of the SRP.

To provide an indication of the difference between the results from the two methods when other ductility ratios are used, the velocity of the missile is arbitrarily increased from 241 ft/sec to 364 ft/sec. This yields a missile penetration of 12 inches, which is 2/3 of the panel thickness (the maximum penetration permitted per the FSAR. Under this condition,

$$[y_{max}]_{method\ 1} = 2.08 \text{ inches}$$

$$[\mu]_{method\ 2} = 3.29$$

$$[y_{max}]_{method\ 2} = 3.29 \times 0.680 = 2.24 \text{ inches}$$

To summarize the results obtained by using the corrected Equation (3), values of $[y_{max}]$ Method 1/$[y_{max}]$ Method 2 are given below:

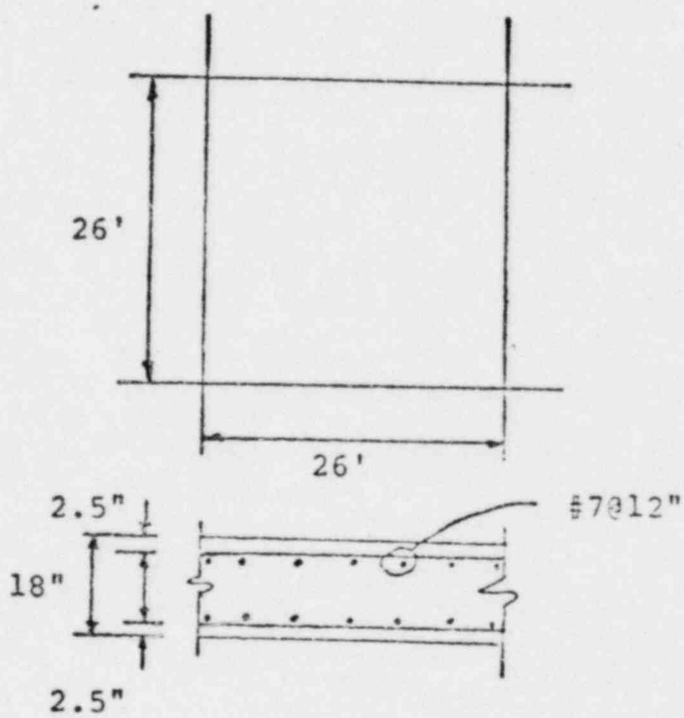
Missile Velocity ft/sec	Penetration Depth inches	Max. Deflection Ratio of the Two Methods
241	5.9	1.22
364	12	0.927

Since the maximum underestimation of Method 1 of the FSAR is less than 8%, the use of this method is considered acceptable.

REFERENCES

1. (From SRP 3.5.3): R. A. Williamson and R. R. Alvy, "Effect of Fragments Striking Elements," Homes and Narver, Inc., Revised November 1973.

2. R. P. Kennedy, "A Review of Procedures for the Analysis and Design of Concrete Structures to Resist Missile Impact Effects," Nuclear Engineering and Design 37 (1976) North-Holland Publishing Company.



f'_c = 3500 psi

f_y = 60 ksi

BYRON/BRAIDWOOD STATIONS
FSAR

FIGURE Q130.25-1

Missile - Re[illegible]
Concrete Panel

QUESTION 130.27

"Specify the procedures for prediciton of local damage of barriers consisting of composite sections. Indicate the compliance of the method selected with that contained in the SRP Section 3.5.3."

RESPONSE

Composite sections have not been used as barriers at the Byron and Braidwood Stations.

QUESTION 130.28

"Sargent & Lundy Report SL-3026, referenced in the FSAR, has not been reviewed and approved by the Regulatory Staff. Describe the contents of the report in an appropriate section of the FSAR or provide a copy of the report for review by the staff."

RESPONSE

Sargent & Lundy Report SL-3026, "Seismic Soil-Structure Interaction Analysis of Nuclear Power Plants," was transmitted on May 9, 1973 (letter from R. N. Bergstrom (S&L) to Dr. K. Kapur (AEC)). The report presents a general procedure for a coupled three-dimensional soil-structure interaction analysis using modal synthesis and finite element techniques.

QUESTION 130.29

"In Section 3.7.2.10 of the FSAR you stated that each individual floor framing beam of the buildings was designed statically for 1.5 times the acceleration corresponding to fundamental frequency of the beam from the applicable wall response spectrum. The position of the Regulatory staff states that the factor of 1.5 is to be applied to the peak acceleration of the applicable floor response spectra (see SRP Section 3.7.2-II.1.b.(3). Assess the impact of this apparent discrepancy between the two criteria and provide a justification for any deviation from the SRP."

RESPONSE

B/B-FSAR Subsection 3.7.2.10 is in conformance with provisions of the SRP. There is no deviation between the analysis and design method stipulated in the B/B-FSAR and in SRP Section 3.7.2-II.1.b, which permits the use of any rational and justifiable equivalent static load method. Justification is given below.

1. SRP Section 3.7.2-II.1.b.(3) applies only to the design of floor attached structures, equipment, and components and is based on a static load method which involves no anlaysis, i.e., no frequency calculation or modeling of the component.

2. The equivalent static load design method stated in B/B-FSAR Subsection 3.7.2.10 for design of floor framing is a more comprehensive and realistic method. It involves modeling each main floor framing-member, and determination of the fundamental frequency of the member, consideration of source of seismic excitation, and includes the effect of higher mode participation. The adequacy and conservatism of this method has been evaluated by comparison of results with a dynamic analysis for a typical floor framing. Results have been previously published from papers of the ASCE Spring Convention, Dallas, Texas, in April 1977 (Reference 1).

3. The justification for using the wall response spectrum instead of the floor response spectrum is as follows: The floor framing members are supported by steel columns. Columns are included in the vertical seismic model with the walls. Seismic response of floor framing members is given by the response of the supporting columns. Column response is given by the applicable wall spectra.

Therefore, it is appropriate to use the wall response spectra for the design of floor framing members. These wall response spectra adequately account for the amplification effect of any structures or components attached to the wall.

4. Since the floor framing member is modeled as a single-degree-of-freedom system, an amplification factor of 1.5 is used to account for higher mode participation. This factor is a conservative value. A typical steel floor framing member behaves close to a single-degree-of-freedom system, in which higher mode participation is insignificant. Use of 1.5 as the amplification factor for flexible beams having frequencies lower than 33 Hz will ensure the design adequacy of the equivalent static load method used herein.

REFERENCE

1. Y. L. Tien, V. Kumar, and S. J. Fang, "Design of Composite Floors For Vertical Seismic Loads," presented at ASCE Spring Convention, Dallas, Texas, April 1977, ASCE paper No. 2886.

QUESTION 130.30

"The statement in FSAR Section 3.7.2.8 that non-Category I structures are designed to prevent their failure under SSE conditions requires explanation. Describe what specific measures which have been taken to assure that failure of non-Category structures will be prevented not only under SSE condition but also under other loading conditions."

RESPONSE

Category II structures integrally connected to or located in the close vicinity of Category I structures are designed for Category I loads in order to prevent their failure on Category I structures. Table Q130.30-1 describes structures and elements under consideration and shows loads for which they are designed.

TABLE Q130.30-1

CATEGORY II STRUCTURES TO BE DESIGNED

FOR CATEGORY I LOADS

CATEGORY II STRUCTURES	INTERCONNECTED CATEGORY I STRUCTURES	CRITICAL CATEGORY II ITEMS DESIGNED FOR CATEGORY I LOADS	LOADS
1. Turbine Building	1. Auxiliary Building	1. Concrete floors	
2. Containment Building Buttress and Dome Enclosure	2. Containment Building	2. Columns 3. Crane girders 4. Roof girders 5. Vertical and horizontal bracing 6. Roof trusses 7. Purlins required for lateral support of roof girders 8. Tie rods 9. Connections to Auxiliary Bldg. at L row 10. Shear walls 11. Mat 12. Embedments to the Containment	SSE and Tornado
Train Shed	Fuel Handling Building	See Note	See Note

NOTE: Failure of the train shed under Safety Category I loadings will have no detrimental effect on adjacent Safety Category I structures. Only two roof girders from the train shed frame into the Fuel Handling Building will be permitted to fail. Consequences of their failure will be less critical than the failure due to missiles identified in Subsection 3.5.1.4 of the B/B-FSAR.

QUESTION 130.31

"The seismic responses for shear walls contained in Figures 3.7-57 thru 3.7-59 pertain to Braidwood Station only. Provide similar responses for the Byron Station."

RESPONSE

The seismic responses tabulated in Figures 3.7-57 through 3.7-59 pertain to Braidwood Station. The seismic responses for similar shear walls pertaining to Byron Station are tabulated in Tables Q130.31-1 through Q130.31-3.

TABLE Q130.31-1

BYRON STATION

SEISMIC RESPONSE (SSE) FOR WALLS AND COLUMN ROW L

AUXILIARY-FUEL HANDLING BUILDING

ELEVATION (ft)	SHEAR (kips)	MOMENT (k-ft)
477	2706	21646
451	5093	149645
439	15565	328238
426	16835	544536
401	20816	1045220
383	8784	1195894
367	8489	1342584
346	9687	1497007

TABLE Q130.31-1

BYRON STATION

SEISMIC RESPONSE (SSE) FOR WALLS AND COLUMN ROW Q

AUXILIARY-FUEL HANDLING BUILDING

ELEVATION (ft)	SHEAR (kips)	MOMENT (k-ft)
451	1739	45221
439	5612	111813
426	6243	192590
401	5725	331660
383	1958	565962
367	3628	429779
346	1369	451838

TABLE Q130.31-1

BYRON STATION

SEISMIC RESPONSE (SSE) FOR WALLS AND COLUMN ROW 30

AUXILIARY-FUEL HANDLING BUILDING

ELEVATION (ft)	SHEAR (kips)	MOMENT (k-ft)
451 ft 0 in.	3401	62419
439 ft 0 in.	8286	159401
426 ft 0 in.	8721	271789
401 ft 0 in.	10597	528867
376 ft 6 in.	9040	844375

QUESTION 130.34

"With reference to Section 3.8.1 of the FSAR: The 'ASME Boiler and Pressure Vessel Code' Section III, Division 2, 1973 is not entirely acceptable to the Regulatory staff. The SRP Section 3.8.1-II, describes the exceptions to the Code taken by the staff. Indicate your compliance with these exceptions or justify any deviations therefrom, if any."

RESPONSE

The exceptions described in SRP Section 3.8.1-II to the ASME Code, Section III, Division 2 are addressed in B/B-FSAR Subsection 3.8.1.5, where we have stated there is complete compliance with these exceptions.

QUESTION 130.35

"Section 3.8.1.4.7 of the FSAR implies that the only transient load that has been considered is that of thermal gradient. Indicate if what and how other transient loads have been considered in the design of the containment, (e.g., resulting from pipe break due to LOCA)."

RESPONSE

The transient thermal gradient described in Subsection 3.8.1.4.7 occurs as the containment wall heats up gradually after a LOCA in response to elevated containment atmospheric temperatures. Therefore, it is treated as a static load.

The seismic loads on the structure are transient and are determined from a dynamic analysis as indicated in Subsection 3.7.2.

LOCA pressures are transient; however, LOCA pressures are considered as a static loading because the rate of pressurization is gradual as shown in Figures 6.2-1 through 6.2-6. Pipe loads resulting from pipe breaks are transient. In design the bounding values of these loads are calculated on the basis of the collapse mechanism of the pipe (e.g., the pipe's plastic moment).

QUESTION 130.36

"with reference to FSAR Section 3.8.1.5.2 explain and justify the use of ACI-318-71 Section 11.4 for radial shear which is for nonprestressed concrete members. Explain why Section 11.5 of the ACI-318-71 was not used which is for shear stress for prestressed concrete members."

RESPONSE

Prestressed concrete sections typically have higher concrete shear strength than nonprestressed concrete sections. Several critical sections in the containment structure were evaluated for radial shear using both Section 11.4 and Section 11.5 of ACI-318-71 and it was found that the allowable concrete shear strength per Section 11.5 was, in general, at least 50% higher than that allowed under the provisions of Section 11.4. Therefore, Section 11.4 was used for reasons of conservatism and expeditious design.

QUESTION 130.37

"Section 3.8.1.7.2.2 of the FSAR and Appendix A indicates that structural acceptance test will meet the 'intent' of R.G. 1.18. This statement implies that there may be some deviations from the Regulatory Guide. Specify and justify any of such deviations with respect to the provisions of the R.G. 1.18."

RESPONSE

B/B-FSAR Subsection 3.8.1.7.2.2 and Appendix A will be revised to indicate compliance with the provisions of Regulatory Guide 1.18.

QUESTION 130.38

"Section 3.8.1.7.3.2 of the FSAR and Appendix A indicate that inservice tendon surveillance program will meet the 'intent' of R.G. 1.35. Such a statement implies that there may be some deviations from the Regulatory Guide. Furthermore, the present position of the Regulatory staff regarding inservice tendon surveillance program is stated in R.G. 1.35, April 1979 and R.G. 1.35.1, April 1979. Specify and justify any deviations in your inservice tendon surveillance program from the provisions of these Regulatory Guides."

RESPONSE

The inservice tendon surveillance program is presented in the Technical Specifications (B/B-FSAR Subsection 16.3/4.6.1.7) and will be changed to conform to the Regulatory staff position per Regulatory Guides 1.35 and 1.35.1, with the exception of the applicable acceptance criteria. The acceptance criteria, Subsection 3.8.1.7.3.2, will be modified as follows.

Tendon lift-off stress shall be adjusted to account for elastic losses which are a function of the original stressing sequence. These values will be compared with the average design values as stated below:

Hoop	140 ksi
Vertical	147 ksi
Dome	143 ksi

This acceptance criteria ensures the containment capacity through maintenance of tendon stress design values.

QUESTION 130.39

"Provide more information regarding the prestressing system used at the Byron/Braidwood plants. In your response indicate the system selected, type of strands, the ultimate tensile strength of wires and rods used, what are the allowable stresses in the tendons immediately after anchorage, state if tendons are grouted or ungrouted. Describe tendon sheathing, the trumpets, etc., and indicate the standard used for design of the prestressing system and any deviation therefrom."

RESPONSE

The maximum overstressing force in any tendon at the jacking end is limited to 80% of the guaranteed ultimate tensile strength (GUTS) of the tendon. The prestressing force in the tendon immediately after anchoring and prior to time dependent losses is 70% GUTS.

The trumplate assembly consists of a 20-1/2 inch by 20-1/2 inch by 3-3/4 inch bearing plate with a 7 inch diameter tube trumpet. Transition cones connect the trumpets to the semirigid sheathing which encloses the ungrouted tendons.

All other requested information is given in Section B.3 of Appendix B of the B/B-FSAR.

QUESTION 130.40

"Describe the corrosion protection provided for the tendons from the point of fabrication to the installed location and during the expected life-span of the system."

RESPONSE

A complete description of corrosion protection for the post-tensioning system used at the Byron/Braidwood Stations is given in Section B.3 of Appendix B.

QUESTION 130.41

"The loads and load combinations described in FSAR Section 3.8.2.3 and Table 3.8-7 require an explanation. It is the Regulatory staff's position that Subsection NE of the ASME Code Section III, Division I is not explicit with respect to the loads and load combinations which should be considered in the design of steel containments and metal MC components. Also Table 3.8-7 categories different load combinations in a different manner from that which is used in the ASME Code. The loads and load combination equations acceptable to the Regulatory staff are listed in the SRP Section 3.8.2 together with the allowable stresses. In view of the above you are requested to provide the following:

a) Correlate the loads and load combination equations with those listed in SRP Section 3.8.2.

b) Provide the allowable stresses for all load combination equations.

c) Specify the load combination equation for the flooded condition (the present load combinations in Table 3.8-7 seem not to include this condition).

d) Provide an explanation of all of the load symbols used (some like P or P_o are not explained and some such as loads under extreme environmental conditions are not listed).

e) Describe any deviations from the SRP Section 3.8.2 which have been used in design of metal portions of the containment and provide a justification therefrom."

RESPONSE

a) The load combinations for class MC components from FSAR Table 3.8-7 and SRP 3.8.2 correlate as shown in the following:

Load Comb No.	SRP 3.8.2	Load Comb No.	FSAR Table 3.8-7
(1)	$D + L + P_t + T_t$	(2)	$D + L + P_t + T_t$
(2)	$D + L + T_o + R_o$	(3)	$D + L + R_o + T_o + P_o + P_e$
(3)	$D + L + T_o + R_o + E$	(4)	$D + L + R_o + T_o + P_o + P_e + E$

(4) $D + L + T_a + P_a + R_a + E$	(8) $D + L + P_e + E + T_a + P_a + R_a$ or (10) $D + L + P_e + E + R_r + T_a + P_a + R_a$
(5) $D + L + T_e + P_e + R_e + E$	(4) $D + L + R_o + T_o + P_o + P_e + E$
(6) $D + L + T_a + R_a + P_a + E'$	(11) $D + L + P_e + R_r + T_a + P_a + R_a + E'$ or (12) $D + L + P_e + R_r + T_a + P_a + R_a + E'$
(7) $D + L + T_e + P_e + R_e + E'$	(7) $D + L + R_o + T_o + P_o + E'$
(8) $D + L + T_a + R_a + Y_r + Y_j$ $Y_m + E' + P_a$	(11) $D + L + P_e + R_r + T_a + P_a$ $R_a + E'$ or (12) $D + L + P_e + R_r + T_a + P_a + R_a + E'$
(9) $D + L + E + F_1$	None

b) The allowable stresses for the design of the class MC Components covered under Table 3.8-7 can be determined from either paragraphs NE-3131 (a), (b) and (d), or NE-3131 (c) of ASME, Section III, Division I Code. (The applicable edition of ASME Code is 1971, with coverage through the Summer Addenda, 1974. This is referenced in FSAR Subsection 3.8.2.5.1.) The allowable stresses for the corresponding load combination equations are shown in the following:

CLASS MC COMPONENTS - ALLOWABLE STRESSES

TABLE 3.8-7 LOAD COMBINATION EQUATION NO.	ASME, SECTION III, DIVISION I NE-3131 PARAGRAPH
1	NE-3131 (a), (b) and (d)
2	NE-3131 (a), (b) and (d)
3	NE-3131 (a), (b) and (d)
4	NE-3131 (a), (b) and (d)
5	NE-3131 (a), (b) and (d)
6	NE-3131 (a), (b) and (d)
7	NE-3131 (c)
8	NE-3131 (a), (b) and (d)
9	NE-3131 (a), (b) and (d)
10	NE-3131 (a), (b) and (d)
11	NE-3131 (c)
12	NE-3131 (c)

c) The SRP combination including post-LOCA flooding does not govern the design of the MC components. LOCA pressure produces larger loads on the penetrations because the surface elevation for the flood is less than 4 feet above the base mat.

d) Table 3.8-7 has been revised to include E' in the Extreme Environmental column. Also, in Table 3.8-7, P has been replaced with P' and P_p has been replaced with P_e. Table 3.8-4 will be revised to include definitions for P_e and R_r as shown below.

P_e = External Pressure on Containment, not considering P_a.

R_r = Loads Associated with High Energy Rupture of Piping Systems = $Y_m + Y_j + Y_r$.

e) There are no deviations from the SRP Section 3.8.2 in the design of the metal portions of the containment.

QUESTION 130.42

"The descriptive information of the internal structures is not in accordance with the provisions of R. G. 1.70 'Standard Format.' Provide sufficient information, illustrated by sketches, in FSAR Section 3.8.3, to allow the staff to perform a meaningful review of internal structures of the containment.

"Reference to Sections 1.2 and 3.9 of the FSAR is not acceptable since the sketches in Section 1.2 provide only general outline of the plant, and Section 3.9 is primarily devoted to equipment rather than structures. Description of the structural aspects of the internal structures should be included in FSAR Section 3.8.3."

RESPONSE

Containment building sections in the east-west and north-south directions, primary shield wall, and NSSS component enclosure plans are shown in new Figures 3.8-46 through 3.8-51. These figures, together with the descriptive information in Subsections 3.8.3.1.5 through 3.8.3.1.8 and Figures 3.9-4 through 3.9-9, define sufficient detail for a structural review of the containment internal structures.

QUESTION 130.43

"The descriptive information of Category I structures other than containment is not in accordance with the provisions of the R. G. 1.70. Provide sufficient information, illustrated by sketches in the FSAR Section 3.8.4 to enable the staff to perform a meaningful review. Referencing the FSAR Section 1.2 which illustrates the general layout of the plant is insufficient in detail for a structural review."

RESPONSE

A plan view of auxiliary-fuel handling building shear walls, elevation view, shear wall-slab diaphram connection (above grade and below grade), and a typical wall corner are shown in new Figures 3.8-52 through 3.8-56. These figures, along with the information in Subsection 3.8.4.4, provide sufficient detail for a structural review.

QUESTION 130.44

"Review of Table 3.8-11 reveals that the load combination equations used for the Byron River Screen House are not in conformance with those contained in SRP Section 3.8.4. Some of the loads are missing (tornado), some of the loads have load factors different from those in the SRP (it appears that equation 7 of the SRP is not considered), the symbols of the loads are not explained. Correlate the load combination equations in Table 3.8-11 with those contained in the SRP Section 3.8.4 and justify any deviations or exceptions taken from the Standard Review Plan."

RESPONSE

Comparison of the load combinations in FSAR Table 3.8-11* with those contained in the SRP Section 3.8.4 shows that the load combinations used for the Byron river screen house are in accordance with those contained in SRP Section 3.8.4 with the following exceptions:

1. Byron river screen house is not designed for tornado loading according to FSAR Table 3.8-11. The justification for deviating from SRP Equation 5 in Section 3.8.4 is found in the FSAR Table 3.2-1 Note 1 and Subsection 3.8.4.3:

   "The river screen house is not designed against the probable maximum flood and the design-basis tornado. The ultimate heat sink for the Byron Station consists of a combination of the river screen house and makeup wells. The makeup wells are designed for probable maximum flood and design-basis tornado."

2. Equations 6, 7, and 8 in Section 3.8.4 of the SRP are not applicable because they combine loads due to a postulated high-energy pipe break accident. There are no high-energy pipe lines within the Byron river screen house.

*Note: The Symbols of the loads in these tables are defined in Table 3.8-4 of the FSAR.

QUESTION 130.45

"Examination of Table 3.2-1 reveals that the isolation valve room is a seismic Category I structure. This item is missing in FSAR Section 3.8.4. Provide a description of this structure, the pertinent design and construction information and amend the FSAR, Section 3.8.4 accordingly."

RESPONSE

Refer to revised Subsection 3.8.4.1.4 of the B/B-FSAR.

QUESTION 130.46

"Section 3.8.5.5.2 of the FSAR implies that because of the depth of embedment, the foundation conditions, etc., you assumed that stability of Category I structures against overturning, sliding and buoyancy does not present a problem and an analysis for these conditions is not necessary. You are requested to either provide sufficient information to justify such an assumption or analyze the structures for the above conditions of stability, calculate the resulting factors of safety and compare them with those listed in the SRP Section 3.8.5."

RESPONSE

In Subsection 3.8.5.5.2 stability of the main building complex has been checked against overturning, sliding, and buoyancy in accordance with SRP Section 3.8.5. The FOS against overturning and sliding are 1.84 and 1.32, respectively, for governing SSE loads. These values are greater than the FOS of 1.1 given in SRP Section 3.8.5. Subsection 3.8.5.5.2 will be revised to reflect the factors of safety given above.

The essential service water cooling tower has also been checked for flotation; the FOS found was much greater than 1.1. Factors of safety for other Category I structures are listed in Subsection 3.8.5.5.2.

QUESTION 130.47

"With reference to FSAR Section 3.8.5.3 the extent of load combination equations in other parts of the FSAR is not sufficient. Some of the loads exerted on the foundation do not appear in other load combination equations to which reference has been made (e.g., buoyancy due to flooding). You are, therefore, requested to specify the loads and load combination equations for which foundations of Category I structures have been designed and correlate such loads with those listed in the SRP Section 3.8.5."

RESPONSE

In addition to the loads and loading combinations referred to in Tables 3.8-9, 3.8-10, and 3.8-11, the foundations of Category I structures are designed for the load combinations listed below. (These combinations are used to check stability of the structures against flotation, overturning, and sliding.)

a. D+H+E

b. D+H+W

c. D+H+E'

d. $D+H+W_t$

e. D+F'

In the above combinations, H is lateral earth pressure and F' is the buoyant force. These loads are the same as those listed in SRP Section 3.8.5.

QUESTION 130.54

"Your response to question 010.19 is not satisfactory. You are requested to state whether the liner plate of the spent fuel pool is:

a) considered to be a Category I structure, and

b) subject to Q/A and Q/C requirements of 10CFR, Part 50, Appendix B.

Furthermore, provide the criteria pertaining to the design and construction of the spent fuel pool liner plate."

RESPONSE

The spent fuel pool is comprised of reinforced concrete walls and mat lined with a 3/16-inch stainless steel plate. The reinforced concrete has been designed for the various hydrodynamic effects associated with the seismic loading conditions. The spent fuel pool liner plate is considered to be a Category I structure and is subject to quality assurance and quality control requirements of Appendix B of 10 CFR 50.

Continuous embedded plates (2 inch by 3/8 inch A36 with 5/8 inch diameter Nelson studs) were placed in the concrete pool walls in an approximate 6 foot 0 inch grid pattern. Embedded wide flanges (with 4 inch by 13 inch A36 with 5/8 inch diameter anchor bolts) were placed in a similar 6 foot 0 inch grid pattern in the concrete pool floor. Individual embedded plates (2 inch by 4 inch by 3/8 inch A36 with 1/2 inch diameter Nelson studs) were placed at 2 feet 0 inch on center within these wall and floor grid systems. The liner plate was subsequently attached to the embedments in 6 foot panels with a continuous groove weld at the seams to the continuous grid embedments and with plug welds at 2 feet 0 inch on center to the individual embedments. This anchorage system resulted in unsupported liner panels of 2 feet 0 inch by 2 feet 0 inch.

The liner plate and anchorage system have been designed for the forces resulting from long-term shrinkage of concrete, and a temperature rise to 150° F. from the 70° F. ambient temperature. The maximum compression force in the liner is calculated using the total strain of the long-term shrinkage of the concrete and the temperature rise. This compressive stress in the liner is limited to 0.95 F_y.

The maximum anchor force is obtained as follows. Before buckling, all panels are under the same compressive force and the anchor system is not carrying any load. When buckling occurs, the force in the buckled panel will be lower than that of the adjacent unbuckled panel. The difference between these forces, is resisted by the anchors. The maximum difference occurs when the force in the buckled panel is assumed to be zero. This assumption will lead to a maximum anchor force.

Additional seismic forces acting on the anchorage are found to be negligible when compared to the thermal loading. There are no attachments made directly to the liner plate; therefore, the only seismic forces acting on the anchorage would be from the self-weight excitation of the plate.

The installation of the liner plate was in accordance with the provisions as stated in Section B.7 of Appendix B of the B/B-FSAR.

QUESTION 212.85

"Identify all ECCS LOCA related instruments, valve and valve motors which are expected to be flooded following a postulated LOCA. For any ECCS or RHR valve motors which are submerged following a LOCA, evaluate the consequences of spurious activation of the valves."

RESPONSE

There are no ECCS LOCA-related instruments or valve operators which will become flooded during a postulated accident. Isolation valves RH8701A-1 and RH8701A-2 will be submerged, however, the valve motor-operator is located above the maximum predicted water level. The following air-operated valves will be submerged and inoperable but are not used for safe shutdown:

RC8037A/B/C/D - Loop drain header valves; fail closed.

RE9159A - Isolation valve to gas analyzer from reactor coolant drain tank; fail closed.

RE9160A - Isolation valve to waste gas compressor from reactor coolant drain tank; fail closed.

RY469 - Isolation valve to waste gas compressor from pressurizer relief tank; fail closed.

QUESTION 212.103

"Discuss non-compliance with Reg. Guide. 1.46 paragraphs 1.d and 2.d as indicated in the FSAR, Table 1.46."

RESPONSE

Appendix A has been revised to indicate that the applicant complies with Regulatory Guide 1.46.

QUESTION 212.127

"A change in the Westinghouse fuel rod internal pressure design criteria is in the process of being approved. This change will permit the internal fuel rod pressure to exceed system pressure. For some Condition III and IV overpower events, this will result in an increase in the number of rods normally expected to fail as a result of these events. This is due to the probability of a rod simultaneously being in DNB and exceeding system pressure. Subsequent ballooning and touching the adjacent rods follows, thereby causing more rods to go into DNB and fail. Therefore, for the Chapter 15 analyses of Condition III and IV events, confirm if this change in the fuel rod internal pressure design criteria has been factored into the number of rods predicted to fail."

RESPONSE

The NCR staff has completed its review of the revised Westinghouse fuel rod internal pressure design criteria and has decided on an acceptable amended criteria.

"The internal pressure of the lead fuel rod in the reactor will be limited to a value below that which could cause (1) the diametrical gap to increase due to outward cladding creep during steady-state operation, and (2) extensive DNB propagation to occur."

WCAP-8963, "Safety Analysis for the Revised Fuel Internal Design Basis," was found to be acceptable to support the conclusion that an insignificant number of additional DNB events would occur during transients and accidents as a result of operating with fuel rod pressure (1) greater than nominal system pressure, and (2) limited by the above criterion.

For all Condition III and IV overpower events, the number of rods that are assumed to fail is less than 10%. Therefore, the analyses for the Byron/Braidwood OFA amendment (Amendment 30) are bounded by the analysis presented in the WCAP. The results presented in the WCAP are based on the detailed probability analysis performed to determine the maximum extent of core damage that could lead to DNB propagation. It was shown that the propagation mechanism causes only a small incremental increase in the percentage of rods in DNB. In view of the conservative nature of the failure propagation scheme and the small percentage increase in the number of failed rods, the potential increase in site release is inconsequential.

Although this effect, resulting from the revised fuel rod internal pressure design criterion is small, it was factored into the number of rods predicted to fail.

QUESTION 212.135

"Clarify the statement which states that for an accidental depressurization of main steam system the DNB design limits are exceeded (FSAR 15.1.4.4) and provide a curve for DNBR vs. time for an inadvertent opening of steam generator relief or safety valve."

RESPONSE

For an accidental depressurization of the main steam system, the DNB design limits are met, not exceeded. The corrected text is found in the Optimized Fuel Assembly amendment (Amendment 30 to the B/B-FSAR Subsection 15.1.4.4).

At a meeting between the NRC, Westinghouse, and Commonwealth Edison on Septemer 17, 1981 at Byron Station, Westinghouse stated that it is not necessary to submit a curve for DNBR versus time for an inadvertent opening of a steam generator relief or safety valve. The justification for this statement will be provided in the response to Question 212.138.

QUESTION 212.137

"FSAR Table 1.3-2, Design Comparison, indicates that the boron injection tank has been deleted; however, Section 15.1.4.2.C references injection of concentrated boric acid solution. Explain this apparent discrepancy and any effects on the associated analyses."

RESPONSE

This question is essentially identical to Question 212.86. Refer to that response for a response to this question.

The concentrated boric acid solution (2000 ppm) referenced in Subsection 15.1.4.2.C is drawn from the refueling water storage tank (RWST). Chapter 15.0 was revised in Amendment 30.

QUESTION 212.144

"Provide an analysis for the loss of flow from two or more reactor coolant pumps or provide a justification, with bases, why this condition is not credible."

RESPONSE

The results of the loss of flow analysis are provided in Chapter 15.0 for the Optimized Fuel Assembly (OFA) amendment (Amendment 30). The partial loss of flow case (2/4) is discussed in Subsection 15.3.1 and the complete loss of flow case (4/4) is discussed in Subsection 15.3.2.

QUESTION 362.2

"For the test data presented in Tables 2.5-35 and 2.5-36, identify the source or location of the test specimens."

RESPONSE

The source of sand for these test specimens were block samples taken in test pits excavated within the essential service cooling pond. Three bulk samples were formed using sand material from the block samples. One bulk sample was formed to represent the low fines content portion of the Equality Formation (Test Series 1 in Table 2.5-35), typically found below elevation 585 feet MSL. A second bulk sample was formed to represent the medium fines content (Test Series 2 in Table 2.5-35) typically found near and slightly above elevation 585 feet, MSL. A third bulk sample was formed to represent the high fines content portion of the Equality Formation (Test Series 3 in Table 2.5-35), typically found above elevation 585 feet, MSL.

To obtain sufficient materials for testing, it was necessary to combine and mix materials from more than one block sample to form the bulk samples. Table Q362.2-1 details the block samples used to make the test specimens referenced in Tables 2.5-35 and 2.5-36.

Braidwood Figure 2.5-91 details the locations of test pits excavated within the Essential Service Cooling Pond. Logs of test pits are shown in Braidwood Figures 2.5-254 through 2.5-260.

Tables 2.5-35 and 2.5-36 of the Braidwood FSAR have been revised to identify the source or location of the test specimens.

TABLE 362.2-1

SOURCE OF SOILS FOR TEST SPECIMENS

LISTED IN BRAIDWOOD FSAR TABLES 2.5-35 AND 2.5-36

TEST PIT NUMBER	BLOCK SAMPLE NUMBER	ELEVATION OF BLOCK SAMPLE Ft., MSL	FINES CONTENT
HTP-5	21	586.7	21
HTP-3	32	585.6	13
HTP-3	37	580.8	1
HTP-3	38	584.4	1

QUESTION 362.3

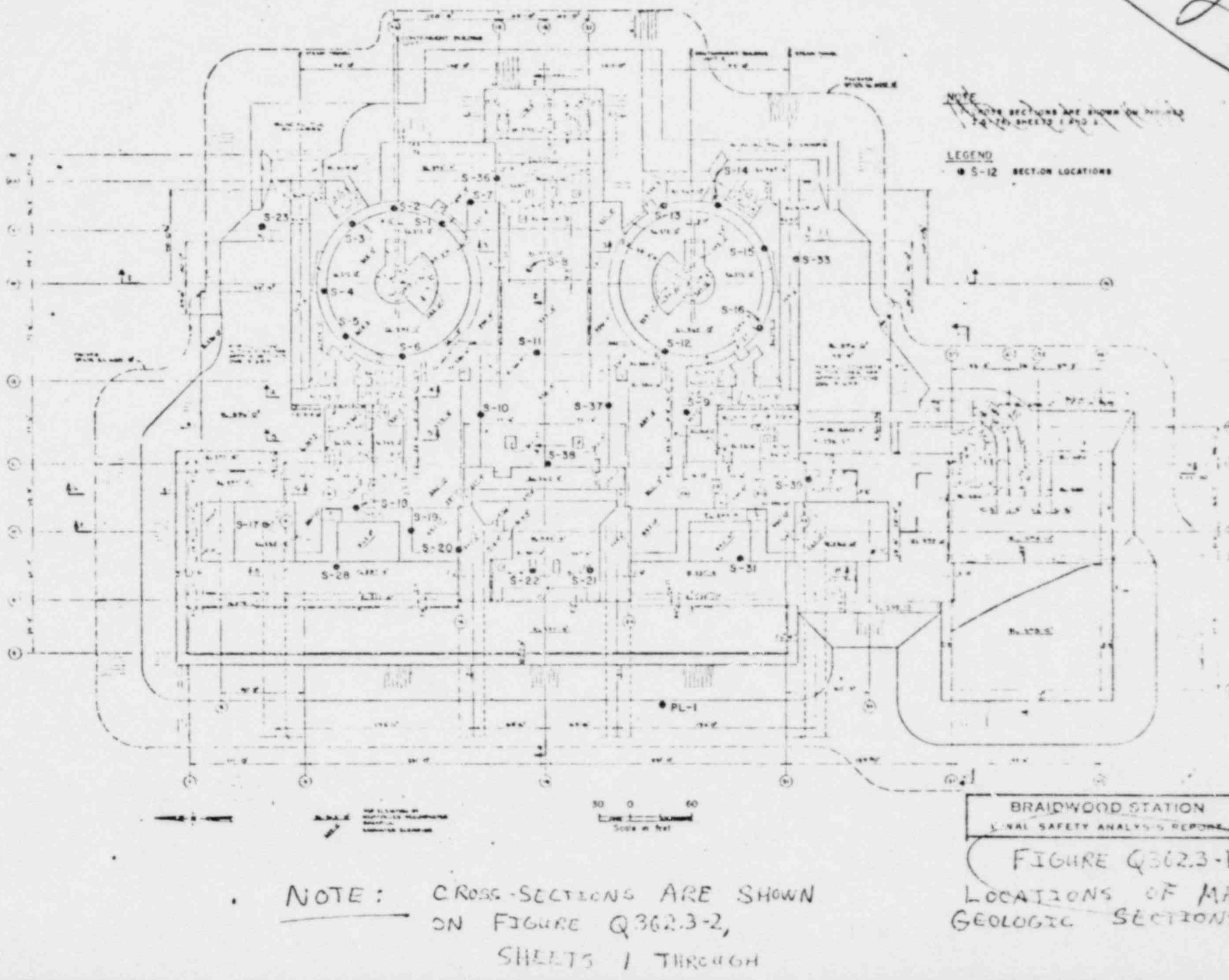
"Section 2.5.4.3 of the FSAR indicates that the foundation excavation was geologically mapped at 29 sections, whereas the sketches of only 3 sections are presented in the FSAR. Submit the detailed sketc' ~s of the remaining sections. The documentation related to the presence of coarse grained material (coarse sand, fine gravel, and cobbles) in the bottom of the Equality formation (fine sand stratum) is of concern because of its impact on the seepage from the Essential Service Cooling Pond."

RESPONSE

The geologic sections requested are shown in Figures Q362.3-2 (27 sheets) and Q362.3-4 (2 sheets). Locations of the sections are shown in Figures Q362.3-1 and Q362.3-3.

The effect of coarse grained materials is addressed in the response to Question 362.4.

Ivory



BRAIDWOOD STATION
FINAL SAFETY ANALYSIS REPORT

FIGURE Q362.3-1

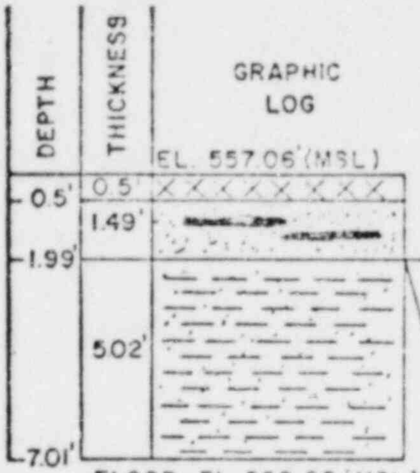
LOCATIONS OF MAPPED GEOLOGIC SECTIONS

NOTE: CROSS-SECTIONS ARE SHOWN ON FIGURE Q362.3-2, SHEETS 1 THROUGH

Q362.3-2

Every

SECTION 1
UNIT 1, EAST PORTION OF REACTOR PERIMETER WALL



FLOOR-EL. 550.05(MSL)
VERTICAL SCALE 1"=5'

DESCRIPTION

MUD MAT

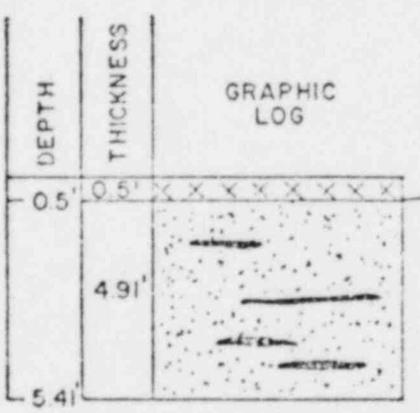
CARBONDALE FORMATION
FRANCIS CREEK SHALE MEMBER
CHANNEL SANDSTONE - Gray to grayish-brown, fine to medium grained, loosely cemented, micaceous, sandstone; thin to medium bedded, some cross-bedding, jointed, slightly calcareous, scattered coal seams up to 1-inch thick.

CONTROL POINT: S-1
COORDINATES: 69 R, 4's of 45°
ELEVATION: 555.07'(MSL)

SILTSTONE - Gray to light gray, slightly sandy, shaley, micaceous, finely laminated, blocky siltstone.

NOTES: LOCATION OF SECTION 1 IS SHOWN ON FIGURE Q362.3-1.

SECTION 2
UNIT 1, NE PORTION OF REACTOR PERIMETER WALL



VERTICAL SCALE 1"=5'

DESCRIPTION

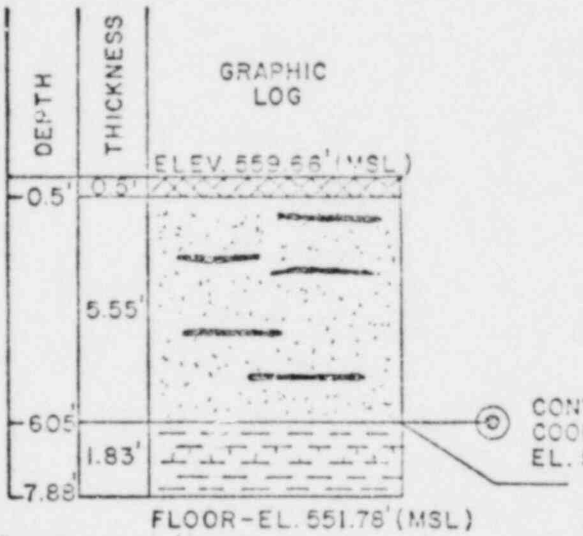
MUD MAT
CARBONDALE FORMATION
FRANCIS CREEK SHALE MEMBER
CHANNEL SANDSTONE - Gray to grayish-brown, fine to medium grained, loosely cemented, micaceous sandstone; thin to medium bedded, some cross-bedding, jointed, scattered coal seams up to 1/2-inch thick

NOTES:
1) NO CONTROL POINT WAS USED IN THIS SECTION AS IT WAS ALL CHANNEL SANDSTONE.
2) LOCATION OF SECTION 2 IS SHOWN ON FIGURE Q362.3-1.

BRAIDWOOD STATION
FSAR
FIGURE Q362.3-2
GEOLOGIC SECTIONS
(SHEET 1 OF 27)

Q362.3-3

SECTION 3
UNIT I, REACTOR PERIMETER NE WALL



MUD MAT

CARBONDALE FORMATION

FRANCIS CREEK SHALE MEMBER

CHANNEL SANDSTONE

Gray to gray-brown, fine to medium grained, slightly calcareous sandstone; loosely cemented, a few scattered clay layers up to 1/2" thick, coal seams to 1-inch thick, clay-filled joints, some cross-bedding.

CONTROL POINT S-3
COORDINATES: 67' R, 12' NW of 319°
EL. 553.61' (MSL)

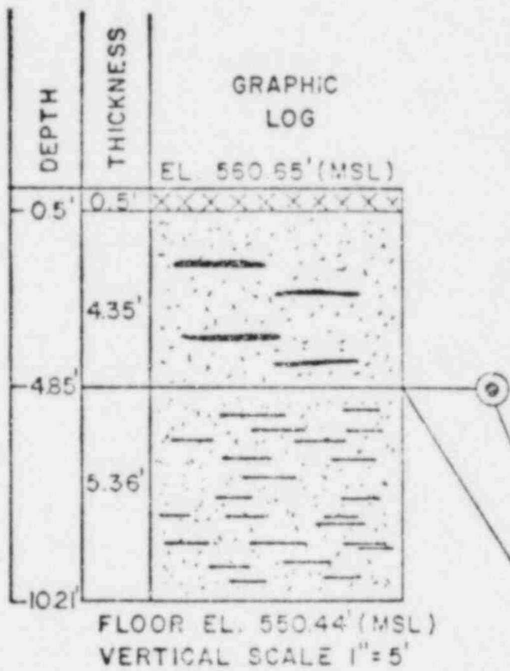
SILTSTONE -

Gray to light gray, slightly sandy, shaley, micaceous, blocky, siltstone; finely laminated, some coal slivers, calcareous in places.

NOTE: LOCATION OF SECTION 3 IS SHOWN ON FIGURE Q362.3-1.

BRAIDWOOD STATION
FSAR
FIGURE Q362.3-2
GEOLOGIC SECTIONS
(SHEET 2 OF 27)

SECTION 4
UNIT I, NORTH PORTION OF REACTOR PERIMETER WALL



DESCRIPTION

MUD MAT

CARBONDALE FORMATION

FRANCIS CREEK SHALE MEMBER

CHANNEL SANDSTONE - Gray to gray-brown, fine to medium grained, slightly cross-bedding, a few clay filled joints, clay layers up to ½" thick, coal seams up to 1" thick.

CONTROL POINT S-4
COORDINATES: 66 R, 2-6' WEST of 270°
ELEVATION: 555.80' (MSL)

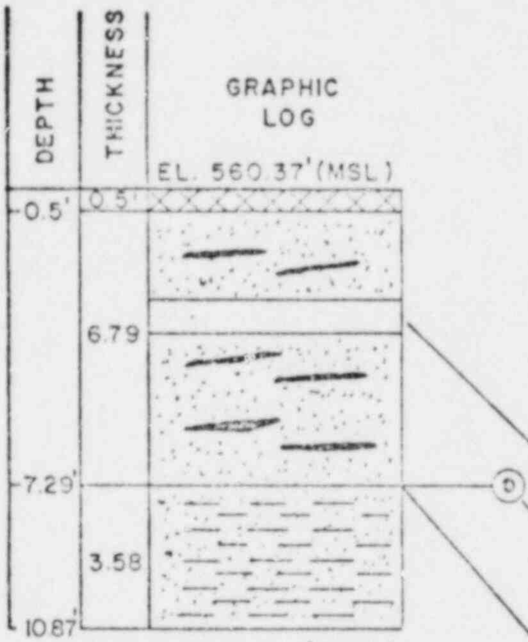
SILTSTONE - Gray to light gray, slightly sandy, shaley, micaceous, blocky siltstone; finely laminated, some coal slivers, calcareous in places.

NOTE: LOCATION OF SECTION 4 IS SHOWN ON FIGURE Q362.3-1.

BRAIDWOOD STATION
FSAR
FIGURE Q362.3-2
GEOLOGIC S[illegible]
(SHEET 3 OF 27)

SECTION 5

UNIT I, NORTHWEST PORTION OF REACTOR PERIMETER WALL



FLOOR EL. 549.5' (MSL)
VERTICAL SCALE 1" = 5'

DESCRIPTION

MUDMAT

CARBONDÁLE FORMATION
FRANCIS CREEK SHALE MEMBER
CHANNEL SANDSTONE - Gray, fine to medium grained, fairly well cemented sandstone; occasional cross-bedding, scattered joints; coal seams up to 1.5 inches thick. Petrified log; 0.8 ft. thick, brown, crumbly composed of lignite.

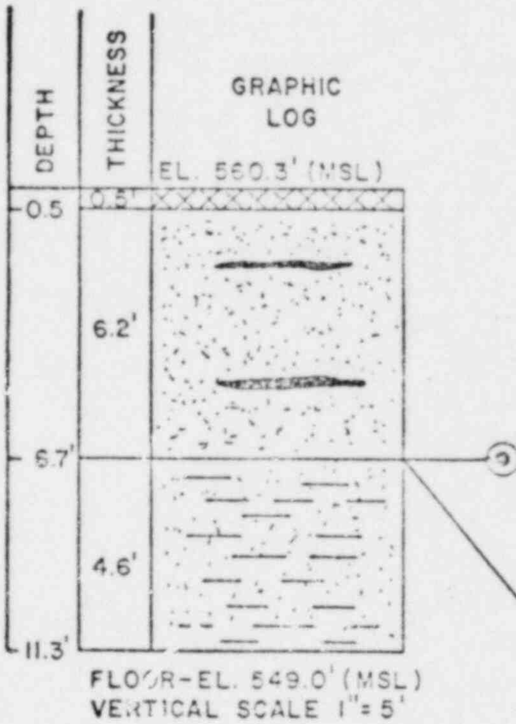
CONTROL POINT: S-5
COORDINATES: 57'R, 11'N of 216°
ELEVATION: 553.08'

SILTSTONE - Gray to lt. gray, slightly sandy, micaceous, shaley, siltstone; some laminations.

NOTE: LOCATION OF SECTION~~S~~ 5 IS SHOWN ON FIGURE Q362.3-1.

BRAIDWOOD STATION
FSAR
FIGURE Q362.3-2
GEOLOGIC SECTIONS
(SHEET 4 OF 27)

SECTION 6
UNIT I, WEST PORTION OF REACTOR PERIMETER WALL



DESCRIPTION

MUD MAT
CARBONDALE FORMATION
FRANCIS CREEK SHALE MEMBER
CHANNEL SANDSTONE - Gray, fine to medium grained, well-cemented sandstone; medium- to thickly-bedded, a few coal seams to 1/2" thick, scattered joints.

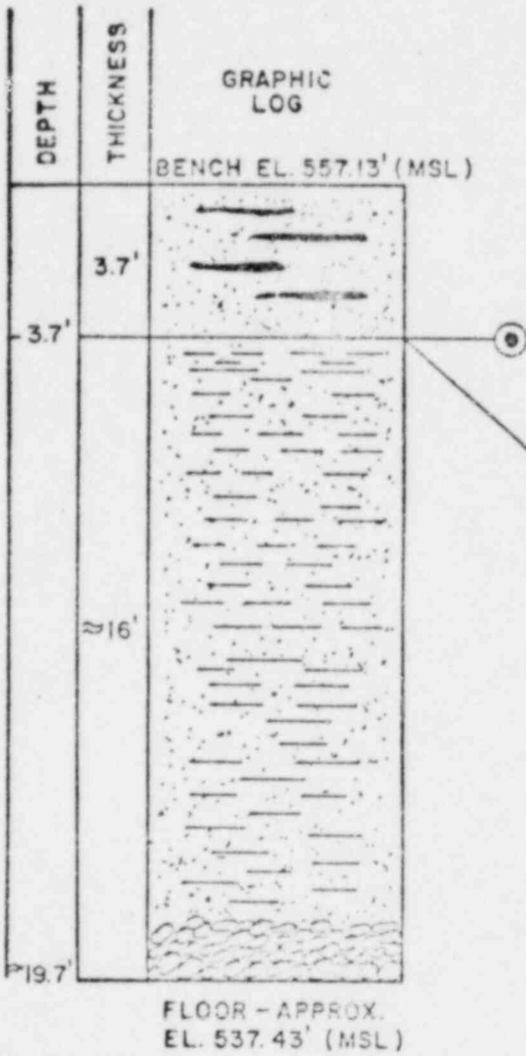
CONTROL POINT: S-6
COORDINATES: 68'R, 20'S of 199°
ELEVATION: 553.60' (MSL)

SILTSTONE - Gray to lt. gray, slightly sandy, micaceous, shaley siltstone; somewhat blocky.

NOTE: LOCATION OF SECTION 6 IS SHOWN ON FIGURE Q362.3-1,

BRAIDWOOD STATION
FSAR
FIGURE Q362.3-2
GEOLOGIC SECT
(SHEET 5 OF 27)

SECTION 7
UNIT 1, CIRCULATING COOLING WATER
EAST WALL



DESCRIPTION

CARBONDALE FORMATION

FRANCIS CREEK SHALE MEMBER

<u>CHANNEL SANDSTONE</u> - Gray, fine-grained, micaceous, somewhat friable, thin-bedded sandstone; abundant coal seams up to 1-inch thick; sandstone beds have an apparent dip of 4° to the south.

CONTROL POINT S-7
COORDINATES: 85'R, 4'S of 45°
EL. 553.43' (MSL)

<u>SILTSTONE</u> - Light gray, slightly sandy, shaley, micaceous siltstone; blocky, calcareous in some places, widely scattered thin coal seams, badly slaked in some areas.

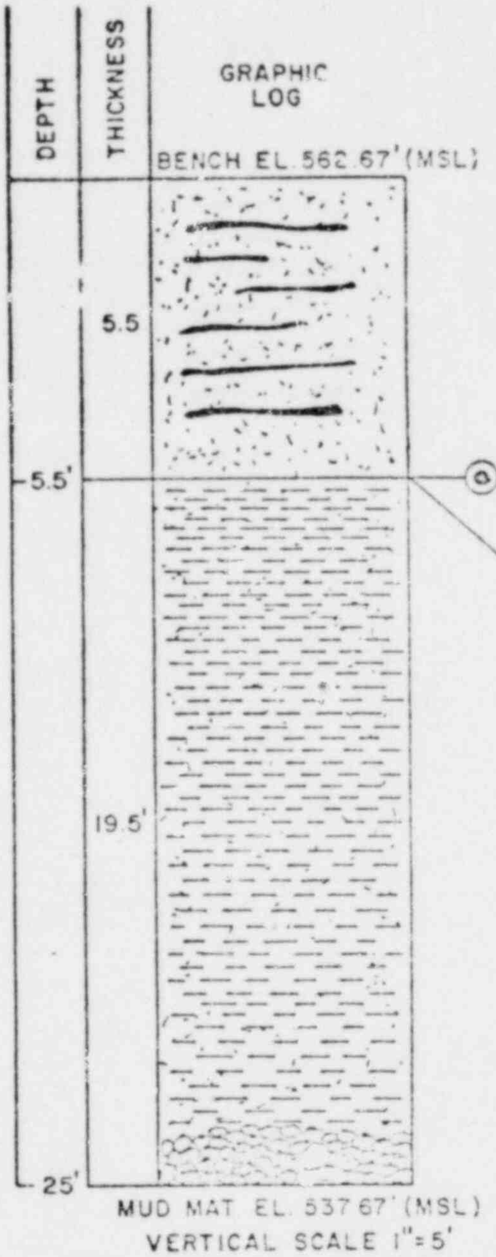
RUBBLE COVERING FLOOR

VERTICAL SCALE 1" = 5'

NOTE: LOCATION OF SECTION 7 IS SHOWN ON FIGURE Q362.3-1.

BRAIDWOOD STATION
FSAR
FIGURE Q362.3-2
GEOLOGIC SECTIONS
(SHEET 6 OF 27)

SECTION 8
EAST WALL OF AUXILIARY BUILDING



DESCRIPTION

CARBONDALE FORMATION
FRANCIS CREEK SHALE MEMBER
CHANNEL SANDSTONE - Gray, fine- to medium-grained, micaceous, thin-bedded sandstone; numerous coal seams to 1" thick. Channel sandstone pinches out to south of measured section.

CONTROL POINT S-8
COORDINATES: S 30 + 72, E 47 + 52
EL. 557.17' (MSL)

SILTSTONE - Light gray, sandy, shaley, micaceous, siltstone; calcareous in places, scattered thin coal seams, same slaking.

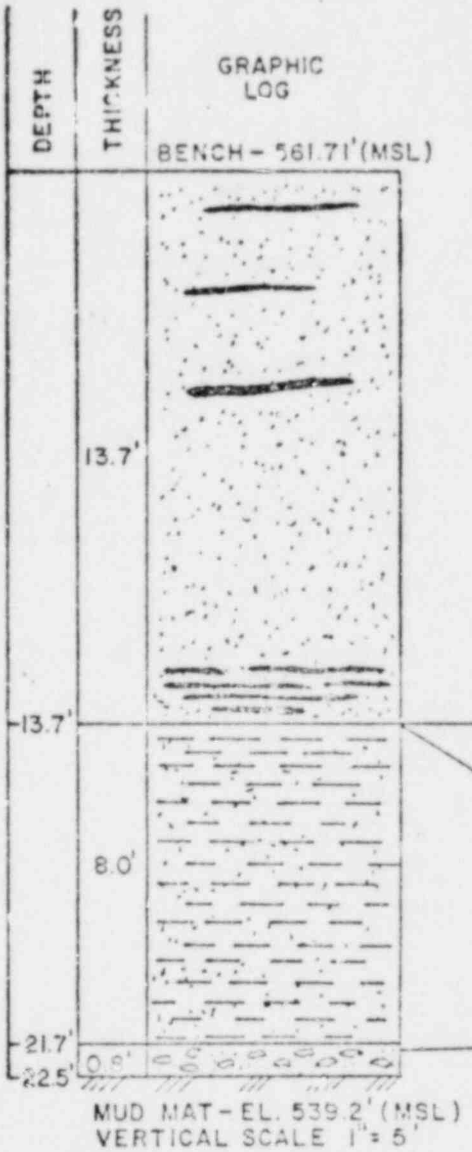
RUBBLE COVERING LOWER PORTION

NOTE: LOCATION OF SECTION 8 IS SHOWN ON FIGURE Q362.3-1.

BRAIDWOOD STATION
FSAR

FIGURE Q362.3-2
GEOLOGIC SECTIONS
(SHEET 7 OF 27)

SECTION 9
SOUTH WALL OF AUXILIARY BUILDING



DESCRIPTION

CARBONDALE FORMATION
FRANCIS CREEK SHALE MEMBER
CHANNEL SANDSTONE - Gray, fine to medium grained, micaceous, well cemented, thinly to thickly bedded sandstone; jointed, scattered coal seams to 2" thick, occasional cross-bedding.

Bottom 1-2' thin bedded with abundant coal seams

CONTROL POINT: S-9
COORDINATES: S 32+18, E 46+30.5
ELEVATION: 548.01'

SILTSTONE - Gray to light gray, slightly sandy, shaley, micaceous siltstone; blocky, finely laminated, jointed, calcareous in places.

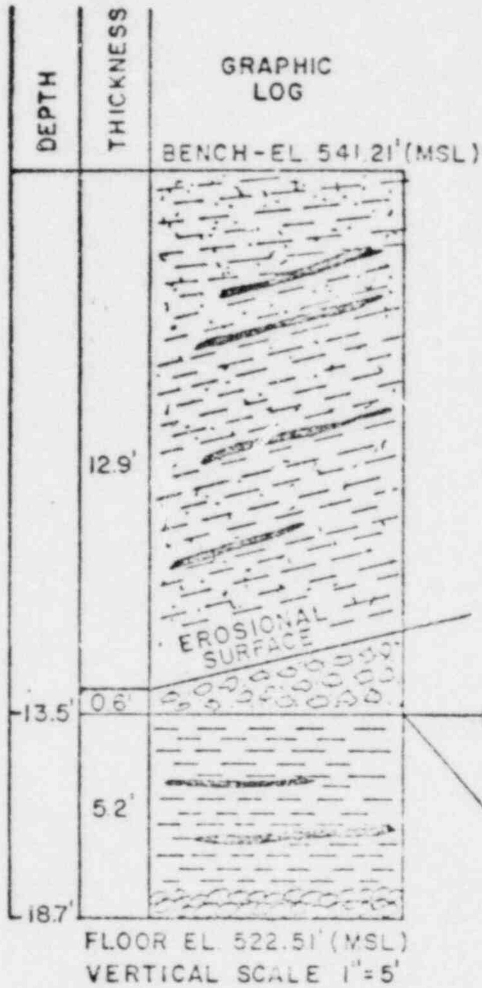
CONGLOMERATE - Siderite conglomerate, pebble 1/4" to 3", some chunks of coal, sandstone matrix. The siderite clasts are sub-angular to angular, indurated.

NOTE: LOCATION OF SECTION 9 IS SHOWN ON FIGURE Q362.3-1.

BRAIDWOOD STATION
FSAR

FIGURE Q362.3-2
GEOLOGIC SECT[illegible]
(SHEET 8 OF 27)

SECTION 10
NORTH WALL OF AUXILIARY BUILDING



DESCRIPTION

CARBONDALE FORMATION
FRANCIS CREEK SHALE MEMBER
SILTSTONE - Gray to light gray, shaley, micaceous siltstone, blocky, some portions calcareous, some areas laminated, scattered sand lenses 1 to 2 inches in thickness, coal seams to 1/2-inch. Bedding has an apparent dip of 15° to west.

CONGLOMERATE + COAL LENS - 0.4' coal lens: pebble conglomerate, pebbles 1/4 to 2-inch in diameter in sandstone matrix.

CONTROL POINT S-10
COORDINATES: S 30+24.33, E 46+23
ELEVATION: 527.71'

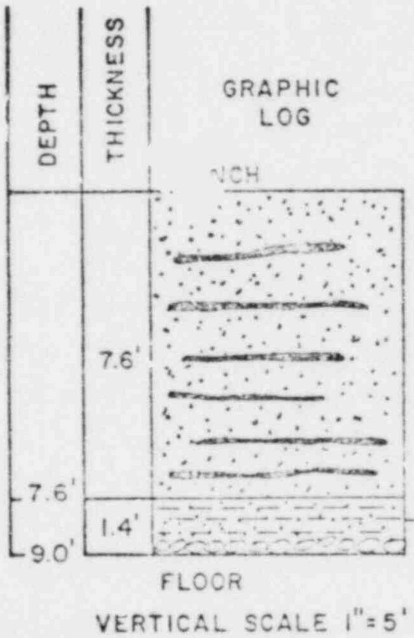
SILTSTONE, as above flat bedded.

RUBBLE

NOTE: LOCATION OF SECTION 10 IS SHOWN ON FIGURE Q362.3-1.

BRAIDWOOD STATION
FSAR
FIGURE Q362.3-2
GEOLOGIC SECTIONS
(SHEET 9 OF 27)

SECTION 13
UNIT 2, EAST PORTION OF REACTOR PERIMETER WALL



DESCRIPTION

CARBONDALE FORMATION
FRANCIS CREEK SHALE MEMBER
CHANNEL SANDSTONE - Gray, fine to mediu grained, micaceous, well cemented sand-stone; top 4 ft. medium bedded, thinly bedded in bottom 3 ft., abundant coal seams to 1¼-inch thick, sandstone friable in bottom 3 ft.

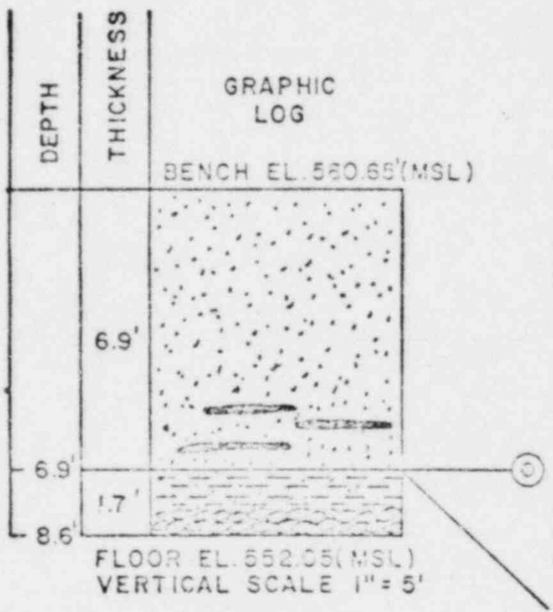
SILTSTONE - Gray, slightly sandy, shaley micaceous siltstone; occasionally laminated, calcareous in places, scatter-ed coal stringers, badly slaked, bottom portion rubble covered.

NOTES: 1) LOCATION OF SECTION 13 IS SHOWN ON FIGURE Q362.3-1.
2) CONTROL POINT COULD NOT BE LOCATED AFTER MAPPING BY SURVEYORS AND COULD NOT BE RESET AS WALLS WERE COVERED WITH PLASTIC SHEETING AND WIRE MESH.

BRAIDWOOD STATION
FSAR
FIGURE Q362.3-2
GEOLOGIC SECTIONS
(SHEET 10 OF 27)

SECTION 14

UNIT 2, EAST PORTION OF REACTOR PERIMETER WALL



DESCRIPTION

CARBONDALE FORMATION
FRANCIS' CREEK SHALE MEMBER
<u>CHANNEL SANDSTONE</u> - Gray, fine to medium-grained, micaceous, well cemented sandstone; medium to thinly bedded with thin shale lenses.

Bottom 1 to 2 ft. thinly bedded with abundant coal seams up to 1-inch thick.

CONTROL POINT: S-14
COORDINATES: 84' 3" R, 19' N of 216°
ELEVATION: 553.75' (MSL)

<u>SILTSTONE</u> - Gray, slightly sandy, micaceous, shaley siltstone; occasionaly laminated, calcareous in places, scattered coal stringers, jointed, badly slaked, rubble covering bottom portion.

NOTE: LOCATION OF SECTION 14 IS SHOWN ON FIGURE Q362.3-1.

BRAIDWOOD STATION
FSAR

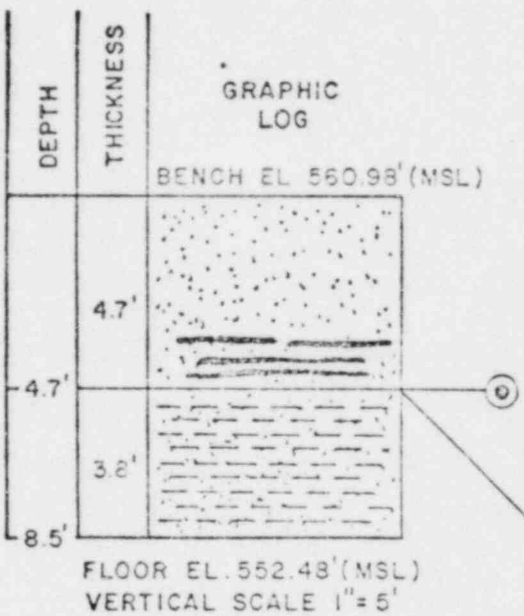
FIGURE Q362.3-2

GEOLOGIC [illegible]

(SHEET 11 OF 27)

SECTION 15

UNIT 2, SOUTHEAST PORTION OF REACTOR PERIMETER WALL (OUTSIDE)



DESCRIPTION

CARBONDALE FORMATION
FRANCIS CREEK SHALE MEMBER
CHANNEL SANDSTONE - Gray, fine to medium-grained, micaceous, well cemented sandstone; top 3 ft. thickly bedded, bottom 1 to 1.5 ft. thinly bedded with abundant coal seams, some cross-bedding.

CONTROL POINT: S-15
COORDINATES: 85'3" R, 13' SW of 234°
ELEVATION: 556.28' (MSL)

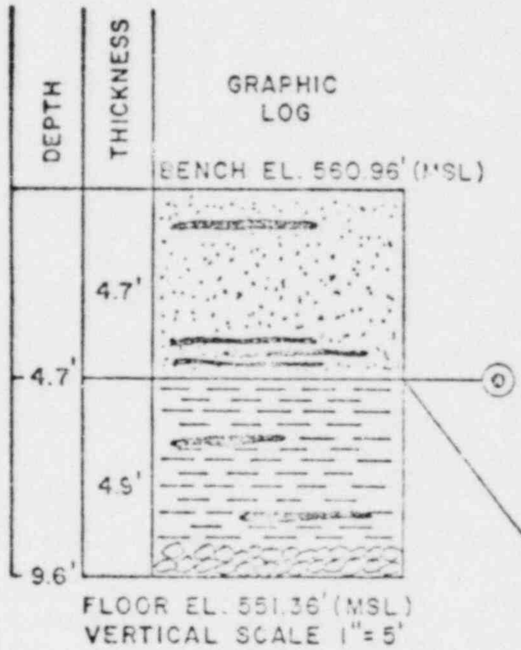
SILTSTONE - Gray, slightly sandy, micaceous, very shaley siltstone; occasionally laminated, calcareous in places, badly slaked, rubble covering bottom portion.

NOTE: LOCATION OF SECTION 15 IS SHOWN ON FIGURE Q362.3-1.

BRAIDWOOD STATION
FSAR
FIGURE Q362.3-2
GEOLOGIC SECTIONS
(SHEET 12 OF [illegible])

SECTION 16

UNIT 2, SOUTHWEST PORTION OF REACTOR PERIMETER WALL (OUTSIDE)



DESCRIPTION

CARBONDALE FORMATION
FRANCIS CREEK SHALE MEMBER
CHANNEL SANDSTONE - Gray, fine to medi[um] grained, micaceous, friable sandstone; top 3' thinly to medium bedded, jointed, scattered concretions and coal seams.

Bottom 1' thinly bedded with abundant coal seams to 1" thick, apparent dip of sandstone beds is 12° N.

CONTROL POINT: S-16
COORDINATES: 93'3" R, 28' WEST of 272°
ELEVATION: 556.26' (MSL)

SILTSTONE - Gray, micaceous, blocky, jointed, shaley, slightly sandy siltstone; badly slaked, some laminations are calcareous in places, scattered coal stringers. Rubble covering bottom of section.

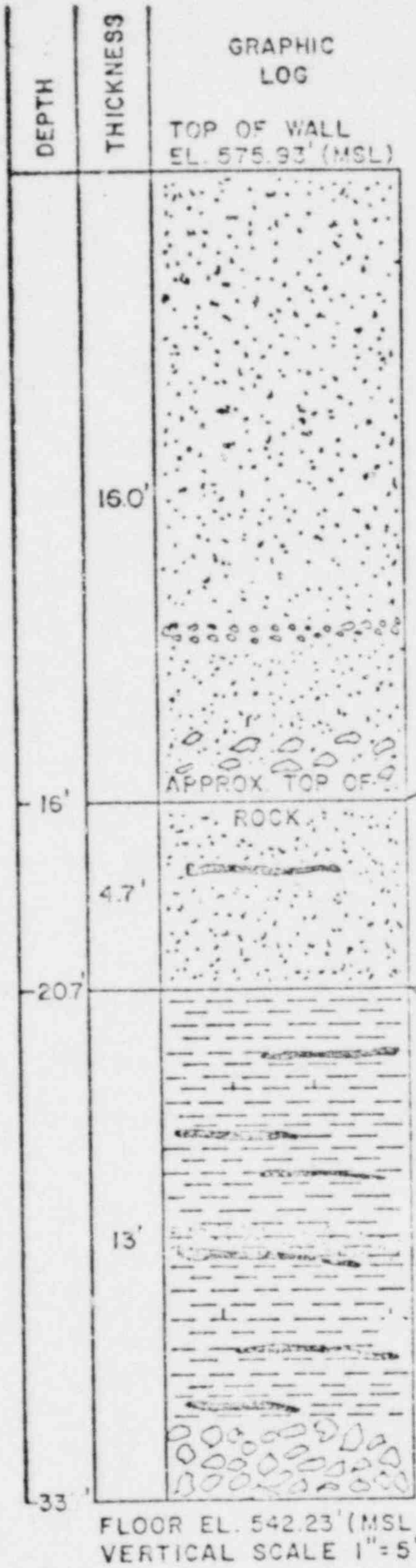
NOTE: LOCATION OF SECTION 16 IS SHOWN ON FIGURE Q362.3-1.

BRAIDWOOD STATION
FSAR

FIGURE Q362.3-2
GEOLOGIC SECTIONS

(SHEET 13 OF 27)

SECTION 17
UNIT 1, NORTH WALL OF TURBINE BUILDING (CIRCULATING WATER PIPE TRENCH)



DESCRIPTION

UNDIFFERENTIATED OVERBURDEN (2)
WEDRON FORMATION

CARBONDALE FORMATION
FRANCIS CREEK SHALE MEMBER
CHANNEL SANDSTONE - Gray, fine to medium grained, micaceous, well cemented, jointed sandstone; medium-to thickly-bedded, sparse to scattered coal seams.

CONTROL POINT: S-17
COORDINATES: S 28+05.5, E 44+87
ELEVATION: 555.23' (MSL)

SILTSTONE - Gray, slightly sandy, micaceous, shaley, blocky siltstone; calcareous in places; top 4' contains abundant coal seams up to 1/2" thick, badly slaked and weathered.

RUBBLE

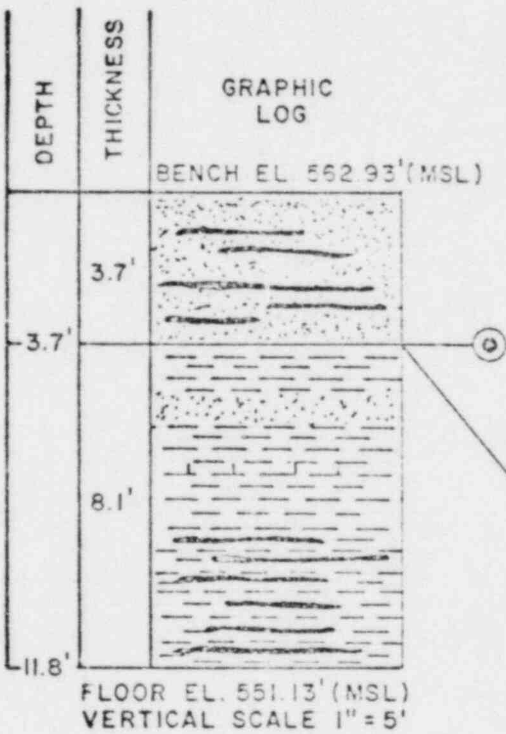
NOTES:
1) LOCATION OF SECTION 17 IS SHOWN ON FIGURE Q362.3-1
2) OVERBURDEN COULD NOT BE MAPPED DUE TO STEEPNESS OF WALL.

BRAIDWOOD STATIONS
FSAR
FIGURE Q362.3-2
GEOLOGIC SECTIONS
(SHEET 14 OF 27)

SECTION 18

UNIT 1, WEST WALL OF TURBINE BUILDING

(NORTH PORTION)



DESCRIPTION

CARBONDALE FORMATION
FRANCIS CREEK SHALE MEMBER
CHANNEL SANDSTONE - Gray, fine-to medium-grained, micaceous, well cemented, jointed sandstone; medium to thickly-bedded, scattered to abundant coal seams to 1" thick.

CONTROL POINT : S-18
COORDINATES : S 29 + 10, E 45 + 33
ELEVATION : 559.23'(MSL)

SILTSTONE - Gray, slightly sandy, micaceous, shaley, blocky siltstone; calcareous in places, badly slaked and weathered, abundant coal seams to 1/2" thick in bottom portion of section.

NOTE: LOCATION OF SECTION 18 IS SHOWN ON FIGURE Q362.3-1.

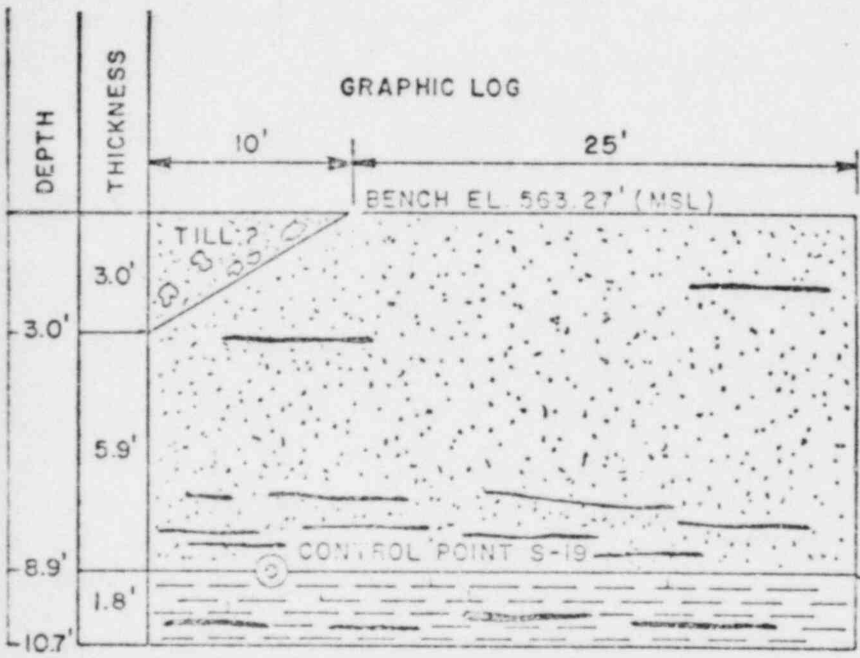
BRAIDWOOD STATION
FSAR

FIGURE Q362.3-2
GEOLOGIC SECTIONS

(SHEET 15 OF 27)

SECTION 19

UNIT I, NORTH WALL OF TURBINE BUILDING



FLOOR EL. 552.57'
VERTICAL SCALE 1"=5'

CARBONDALE FORMATION
FRANCIS CREEK SHALE MEMBER
CHANNEL SANDSTONE - Gray, fine to medium-grained, micaceous, well-cemented, jointed sandstone; medium to thickly-bedded, coal seams abundant near base.

COORDINATES: S 29 + 61, E 45 + 09
ELEVATION: 554.37' (MSL)

SILTSTONE - Gray, slightly sandy, micaceous, shaley, blocky siltstone; calcareous in places, badly slaked and weathered, coal seams to 1/2" thick in bottom portion of section.

NOTE: LOCATION OF SECTION 19 IS SHOWN ON FIGURE Q362.3-1.

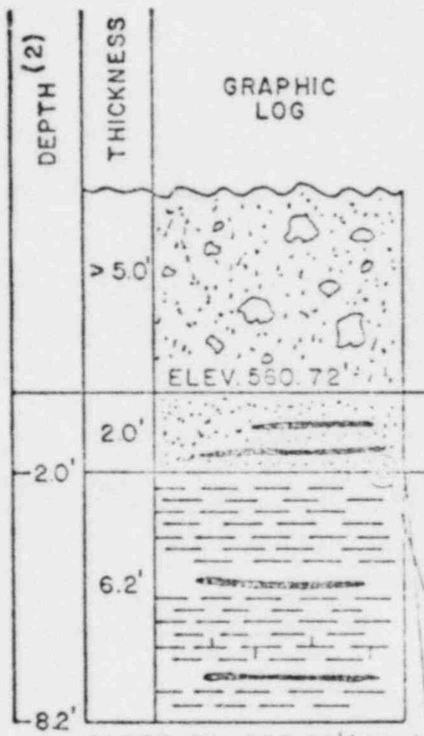
BRAIDWOOD STATION
FSAR

FIGURE Q362.3-2

GEOLOGIC SECT

(SHEET 16 OF 27)

SECTION 20
UNIT I, NORTH WALL OF TURBINE BUILDING



FLOOR EL. 552.52'(MSL)
VERTICAL SCALE 1" = 5'

DESCRIPTION

WEDRON FORMATION
TILL - Gray, silty, sandy clay till; mottled pink and brown. (Thickness not recorded)

CARBONDALE FORMATION
FRANCIS CREEK SHALE MEMBER
CHANNEL SANDSTONE - Gray, fine to medium-grained, micaceous, well cemented, medium bedded sandstone; scattered joints, scattered thin coal seams.

CONTROL POINT: S-20
COORDINATES: S 29 + 98, E 44 + 62
ELEVATION: 558.72' (MSL)

SILTSTONE - Light gray, slightly sandy, micaceous, shaley, blocky, laminated siltstone; calcareous in places, scattered coal stringer

NOTES: 1) LOCATION OF SECTION 20 IS SHOWN ON FIGURE Q362.3-1.
2) DEPTH SHOWN IS DEPTH BELOW WEDRON - CARBONDALE, CONTACT AS THICKNESS OF WEDRON WAS NOT RECORDED.

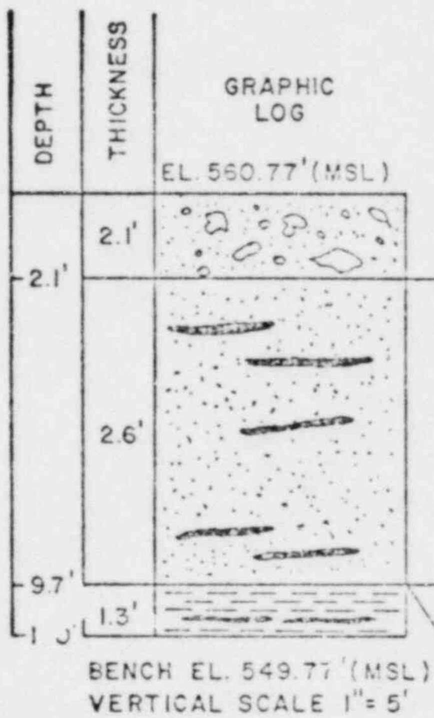
BRAIDWOOD STATION
FSAR

FIGURE Q362.3-2

GEOLOGIC SECTIONS

(SHEET 17 OF 27)

SECTION 21
UNIT 2, WEST WALL OF TURBINE BUILDING



DESCRIPTION

WEDRON FORMATION
TILL - Gray, silty, sandy clay with pink and brown mottling

CARBONDALE FORMATION
FRANCIS CREEK SHALE MEMBER
CHANNEL SANDSTONE- Gray, fine-to medium-grained, micaceous, well cemented, thin-to medium-bedded sandstone; scattered joints, scattered thin coal seams.

CONTROL POINT S-21
COORDINATES: S 31 + 41, E 44 + 65
ELEVATION: 551.07'

SILTSTONE- Light gray, slightly sandy, micaceous, shaley, blocky, laminated siltstone; calcareous in places, scattered coal stringers. The contact with the overlying channel sandstone has an apparent dip of 10° E.

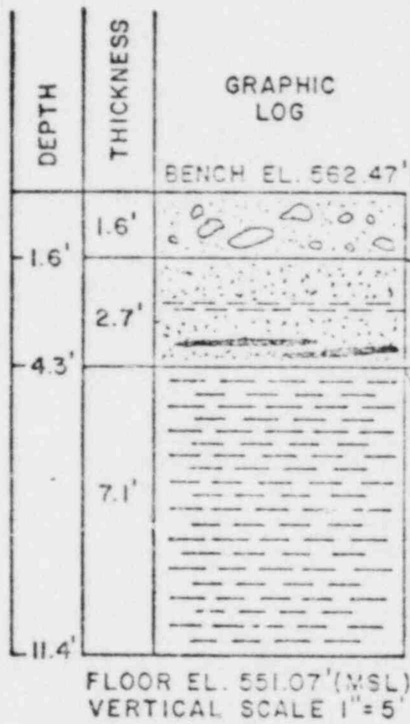
NOTE: LOCATION OF SECTION 21 IS SHOWN ON FIGURE Q362.3-1.

BRAIDWOOD STATION
FSAR

FIGURE Q362.3-2
GEOLOGIC SECTIONS
(SHEET 18 OF 27)

SECTION 22

UNITS 1&2 WEST WALL OF TURBINE BUILDING (BETWEEN UNITS 1 & 2)



DESCRIPTION

WEDRON FORMATION

TILL- Gray, silty, sandy, stiff, slightly gravelly clay; fine-to coarse sand, coarse gravel, pink and brown clasts.

CARBONDALE FORMATION

FRANCIS CREEK SHALE MEMBER

CHANNEL SANDSTONE- Gray, micaceous, fine-to medium-grained, poorly cemented sandstone; scattered shale lenses to 1", scattered coal lenses to 1" thick, siltstone thinly bedded

CONTROL POINT: S-22
COORDINATES: S 30+82, E 44+59
ELEVATION: 558.17' (MSL)

SILTSTONE- Gray, slightly sandy, micaceous, shaley, blocky siltstone; some laminations.

NOTE: LOCATION OF SECTION 22 IS SHOWN ON FIGURE Q362.3-1.

BRAIDWOOD STATION
FSAR

FIGURE Q362.3-2

GEOLOGIC SECTIONS

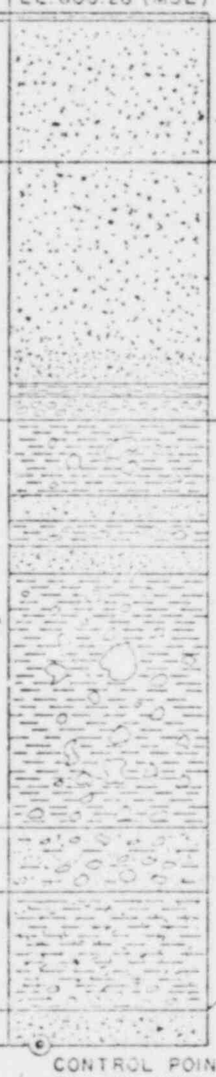
(SHEET 19 OF 27)

SECTION 28

NORTH PORTION OF WEST WALL OF EXCAVATION

GRAPHIC LOG

EL. 600.26' (MSL)



DEPTH	THICKNESS	DESCRIPTION
0.3'	0.3	TOPSOIL - Black, silty, very sandy, organic clay
5.7'	5.4'	PARKLAND SAND - Orangish-brown, very silty, very fine- to to fine-grained sand; slightly clayey in places, horizontal bedding.
15.6'	9.9'	EQUALITY FORMATION - Tan to brown, fine to medium grained, slightly silty sand; cross-bedded near base. sandy clay; lag gravel - fine to coarse grained gravel (EL. 584.6')
31.6'	15.6'	WEDRON FORMATION TILL - Brownish gray to gray, silty, slightly sandy, slightly gravelly clay, with occasional cobbles and boulders: till is blocky and jointed, scattered 1- to 2-inch lenses of sand and gravel, two 1-foot thick beds of silty fine-to-coarse-grained sand near top.
33.6'	2.4'	OUTWASH (?) - Reddish brown to gray and tan silty, gravelly, very sandy clay to slightly clayey sand; sand is fine to coarse grained; fine to coarse grained gravel, scattered cobbles.
38.1'	4.5'	TILL - Reddish brown to pink and gray, silty, sandy, slightly gravelly clay; occasional 1-2 inch sand pockets, gravelly towards base.
39.6'	1.5'	CARBONDALE FORMATION FRANCIS CREEK SHALE MEMBER CHANNEL SANDSTONE - Weathered sandstone, unweathered below control point.

CONTROL POINT: S-28
COORDINATES: S 28 + 83.5
E 44 + 43
ELEVATION: 560.66' (MSL)

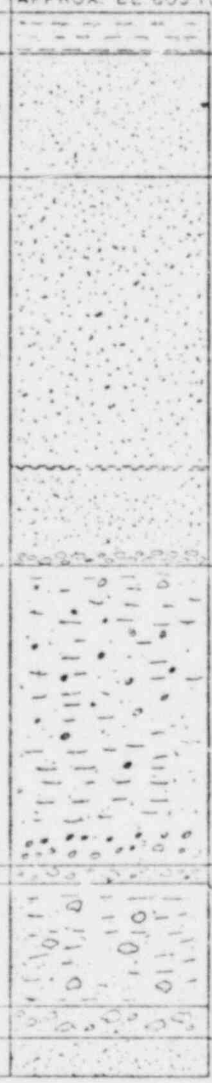
VERTICAL SCALE 1" = 5'

NOTE: LOCATION OF SECTION 28 IS SHOWN ON FIGURE Q362.3-1.

BRAIDWOOD STATION
FSAR
FIGURE Q362.3-2
GEOLOGIC SECTIONS
(SHEET 20 OF 27)

SECTION 31
SOUTH PORTION OF WEST WALL OF EXCAVATION

APPROX. EL. 603.1'(MSL)



DEPTH	THICKNESS	DESCRIPTION
	1.6'	TOPSOIL - Black, very sandy, silty clay with organics
1.6'	4.7'	PARKLAND SAND - Orange to orange-brown, loose, very silty, very fine to fine sand; some root fragments
6.3'	14.9'	EQUALITY FORMATION - Gray to light brown, loose, fine- to medium grained sand; cross-bedded in places. cross-bedding clay-layer approximately 4 feet from base basal lag gravel, abundant cobbles and boulders (EL. 581.9')
21.2'	11.5'	WEDRON FORMATION TILL - Pinkish to brownish gray, somewhat soft to loose, very sandy and silty clay; some gravel, top 1 to 2' very gravelly with abundant cobbles. Gravel most abundant near top and bottom.
32.7'	0.7'	OUTWASH - Grayish brown to light brown, loose, silty sand and gravel; sand mostly fine- to medium-grained, gravel fine to coarse grained.
33.4'	4.7'	TILL - Gray to brownish gray, hard, silty clay with some sand and gravel; scattered cobbles.
38.1'	1.2'	basal outwash of gravelly, fine to coarse sand with abundant cobbles and scattered boulders.
39.3'	1.5'	CARBONDALE FORMATION FRANCIS CREEK SHALE MEMBER CHANNEL SANDSTONE - weathered sandstone
40.8'		

APPROX. EL. 562.3'(MSL)

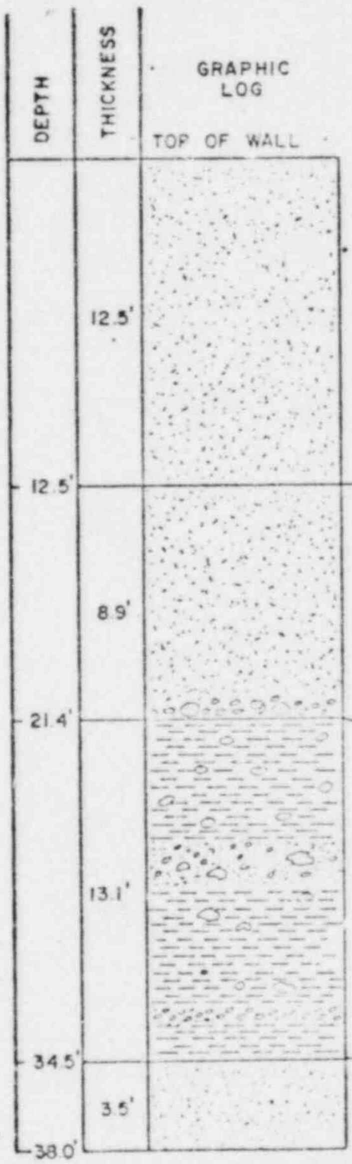
VERTICAL SCALE 1"=5'

NOTES: 1) APPROXIMATE LOCATION OF SECTION 31 IS SHOWN ON FIGURE Q362.3-1.
2) CONTROL POINT WAS DESTROYED BEFORE SURVEYING COULD BE PERFORMED; THE ELEVATION OF THE TOP OF THE WEDRON FORMATION WAS TAKEN FROM FSAR FIGURE 2.5-29.

BRAIDWOOD STATION
FSAR
FIGURE 362.3-2
GEOLOGIC SECTIONS
(SHEET 21 OF 27)

SECTION 33

UNIT 2, SOUTH EXCAVATION WALL
(SOUTH OF UNIT 2 REACTOR)



VERTICAL SCALE 1"=5'

DESCRIPTION

PARKLAND SAND - Orange to light brown, slightly silty to to silty, poorly graded very fine to fine sand; horizontal bedding.

EQUALITY FORMATION - Gray-brown to light brown, poorly graded, fine-to-medium-grained sand.

cross-bedded from 2 to 7 feet above bottom

scattered gravel and cobbles near base (approximate el. 577.0' (MSL)

WEDRON FORMATION
TILL - Gray to brownish gray, hard, silty, clayey, slightly gravelly sand and sandy silt to clayey silt.

OUTWASH - Light brown, loose, fine to coarse sand and fine gravel; scattered cobbles
TILL - Gray, pinkish gray, and brownish gray, hard, sandy, gravelly clayey silt to silty clay; very sandy in places

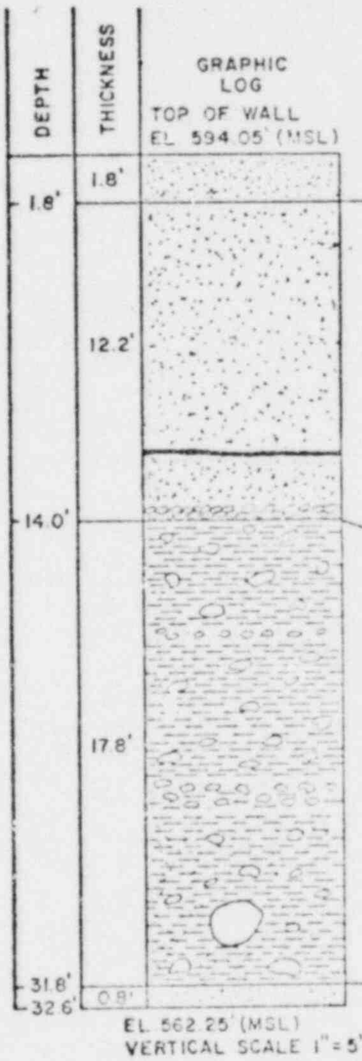
OUTWASH - Fine to coarse sand with gravel.

CARBONDALE FORMATION
FRANCIS CREEK SHALE MEMBER
CHANNEL SANDSTONE - Upper 2-feet weathered, gray, friable, fine-to-medium grained sandstone; weathered to about 36.5'. Floor covered with rubble.

NOTES:
1) CONTROL POINT WAS DESTROYED PRIOR TO SURVEYING. ELEVATION OF TOP OF WEDRON FORMATION IS FROM FSAR FIGURE 2.5-29.
2) APPROXIMATE LOCATION OF SECTION 33 IS SHOWN ON FIGURE Q362.3-1.

BRAIDWOOD STATION
FSAR
FIGURE Q362.3-2
GEOLOGIC SECTIONS
(SHEET 22 OF 27)

SECTION 36
UNIT I, EAST WALL OF EXCAVATION



DESCRIPTION

PARKLAND SAND - Orange to light orangish-brown, loose, slightly silty, very fine- to fine-grained sand; horizontally bedded.

EQUALITY FORMATION - Light brown to gray, loose, poorly graded, fine- to medium-grained sand; cross-bedded in some areas.

clay layer approximately 1 inch thick

lag gravel - sand and gravel with cobbles, layer approximately 0.5 feet thick (EL. 580.85')

WEDRON FORMATION

TILL - Gray to brownish gray, hard, sandy, gravelly, silty clay to clayey silt with scattered cobbles; very sandy in places

OUTWASH - sand and gravel; approximately 0.5' thick

OUTWASH- gray to light brown, loose, soft, silty, clayey, gravelly sand with scattered cobbles, sand is fine to coarse grained, gravel predominataby fine grained.
TILL - Gray to brownish gray, hard, gravelly, sandy, silt clay to clayey silt; scattered cobbles and boulders with glacial striations, a few joints.

CONTROL POINT: S-36
COORDINATES: S 30+59, E 48+37
ELEVATION: 563.05' (MSL)

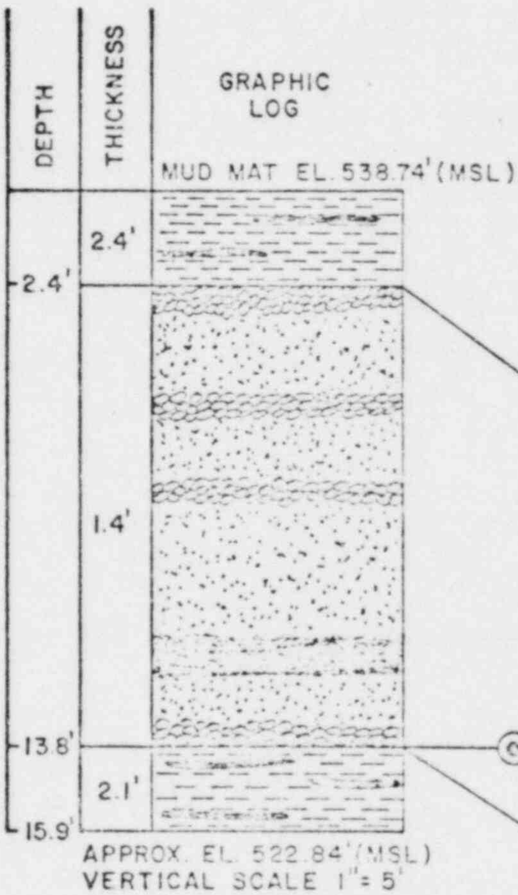
CARBONDALE FORMATION
FRANCIS CREEK SHALE MEMBER
CHANNEL SANDSTONE - Weathered sandstone; unweathered below elevation 562.25' (MSL)

NOTE: LOCATION OF SECTION 36 IS SHOWN OF FIGURE Q362.3-1.

BRAIDWOOD STATION
FSAR
FIGURE Q362.3-2
GEOLOGIC SECTION
(SHEET 23 OF 27)

SECTION 37

UNIT 2 SOUTH WALL OF AUXILIARY BUILDING



DESCRIPTION

CARBONDALE FORMATION
FRANCIS CREEK SHALE MEMBER
SILTSTONE- Gray, slightly sandy, micaceous, shaley, blocky, laminated siltstone; calcareous in places, scattered lenses of coal, sandstone, and siderite pebbles.

SANDSTONE- Gray, well cemented, thin to thick bedded, fine-to medium-grained sandstone alternating with coal beds to 1.5 feet thick and conglomerate beds with gravel and cobble sized siderite concretions in sandstone matrix; a few thin siltstone beds.

CONTROL POINT: S-37
COORDINATES: S 31 + 53.75, E 46 + 03.58
ELEVATION: 529.94' (MSL)

SILTSTONE - As above siltstone

NOTE: LOCATION OF SECTION 37 IS SHOWN ON FIGURE Q362.3-1.

BRAIDWOOD STATION
FSAR

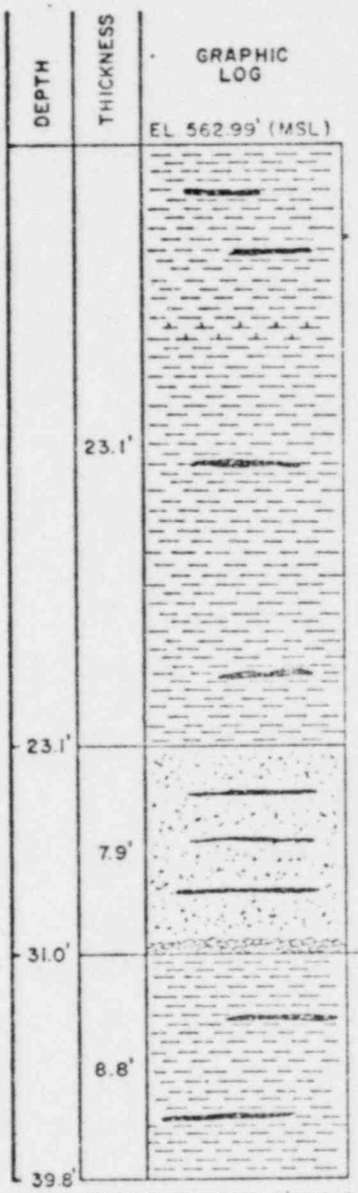
FIGURE Q362.3-2

GEOLOGIC SECTIONS

(SHEET 24 OF 27)

SECTION 38

UNITS 1 & 2-WEST WALL OF AUXILIARY BUILDING



FLOOR EL. 523.19' (MSL)
VERTICAL SCALE 1" = 5'

DESCRIPTION

CARBONDALE FORMATION
FRANCIS CREEK SHALE MEMBER
SILTSTONE - Gray, slightly sandy, micaceous, shaley, blocky, laminated siltstone; calcareous in places, scattered lenses of coal, sandstone and siderite pebbles

SANDSTONE - Gray, thin to thick bedded, fine to medium sandstone; some coal beds to 0.5' thick, thin siltstone layers, basal conglomerate composed of siderite pebbles ranging in size from 0.25 to 3 inches, conglomerate is well cemented. This sandstone pinches out to the north of the measured section.

SILTSTONE - as upper siltstone

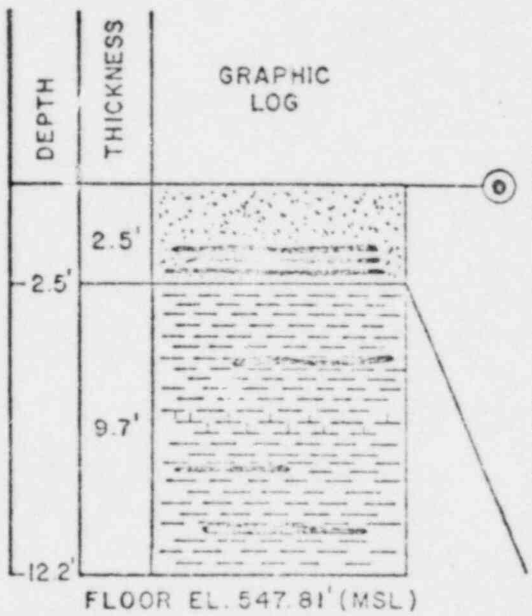
CONTROL POINT: S-38
COORDINATES: S 31 + 02.17, E 45 + 60.17
ELEVATION: 531.99' (MSL)

NOTE: LOCATION OF SECTION SHOWN ON FIGURE Q362.3-1.

BRAIDWOOD STATION
FSAR
FIGURE Q362.3-2
GEOLOGIC SECTIONS
(SHEET 25 OF 27)

SECTION 39

UNIT 2 SOUTH END OF TURBINE ROOM



DESCRIPTION

CONTROL POINT: S-39
COORDINATES: S 33+65, E 45+53
ELEVATION: 560.01'

CARBONDALE FORMATION
FRANCIS CREEK SHALE MEMBER

CHANNEL SANDSTONE - Brownish-gray, fine- to medium-grained, loosely cemented, micaceous, slightly silty, thin- to medium bedded sandstone; occasional cross-beds, abundant coal seams to 4-inches thick, thickly bedded near top.

SILTSTONE - Greenish-gray, slightly sandy, micaceous, laminated, shaley, blocky siltstone; slightly calcareous in places, widely scattered coal stringers.

NOTE: LOCATION OF SECTION 39 IS SHOWN ON FIGURE Q362.3-1.

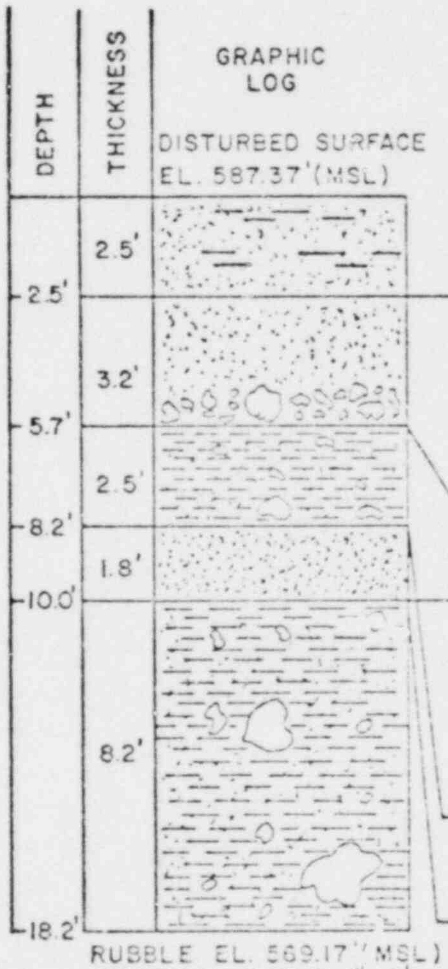
BRAIDWOOD STATION
FSAR

FIGURE Q362.3-2
GEOLOGIC SECTIONS

(SHEET 26 OF 27)

SECTION PL-1

PIPELINE SECTION - EAST WALL OF TRENCH



DESCRIPTION

PARKLAND SAND- Orangish-brown to orange, silty to very silty, very fine-to medium-grained sand.

EQUALITY FORMATION

SAND- Brown to gray-brown, slightly silty, fine-to medium-grained sand; cross-bedded, sand coarser toward base.

LAG GRAVEL-Clayey sand and gravel; boulders to 1' in diameter (581.67')

WEDRON FORMATION

TILL - Brown to gray-brown, silty, clayey sand; scattered gravel, cobble-to pea-sized clasts of reddish brown clay, less clay near top.

CONTROL POINT: PL-1
COORDINATES: 20' N COLUMN LINE 26
15' W COLUMN LINE A
ELEVATION: 577.37'(MSL)

OUTWASH - Orange brown to brown, slightly silty, medium- to fine-grained sand.

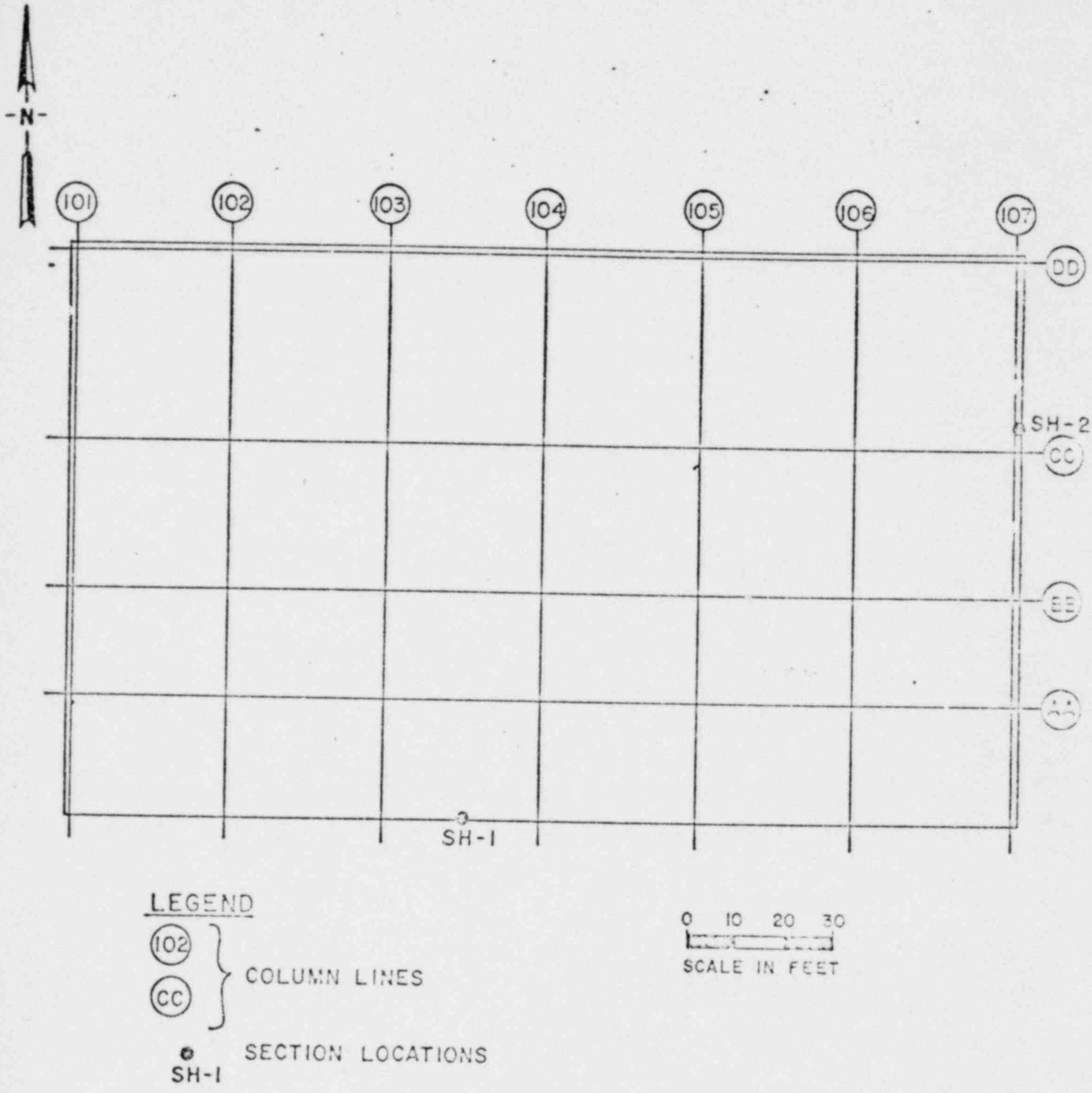
TILL - Gray to gray-brown, hard, slightly sandy, silty clay; scattered gravel, cobbles, and boulders up to 1' in diameter.

NOTE: LOCATION OF SECTION PL-1 IS SHOWN ON FIGURE Q362.3-1.

BRAIDWOOD STATION
FSAR

FIGURE Q362.3-2
GEOLOGIC SECTIONS
(SHEET 27 OF 27)

Ivory



BRAIDWOOD STATION
FSAR
FIGURE Q362.3-3

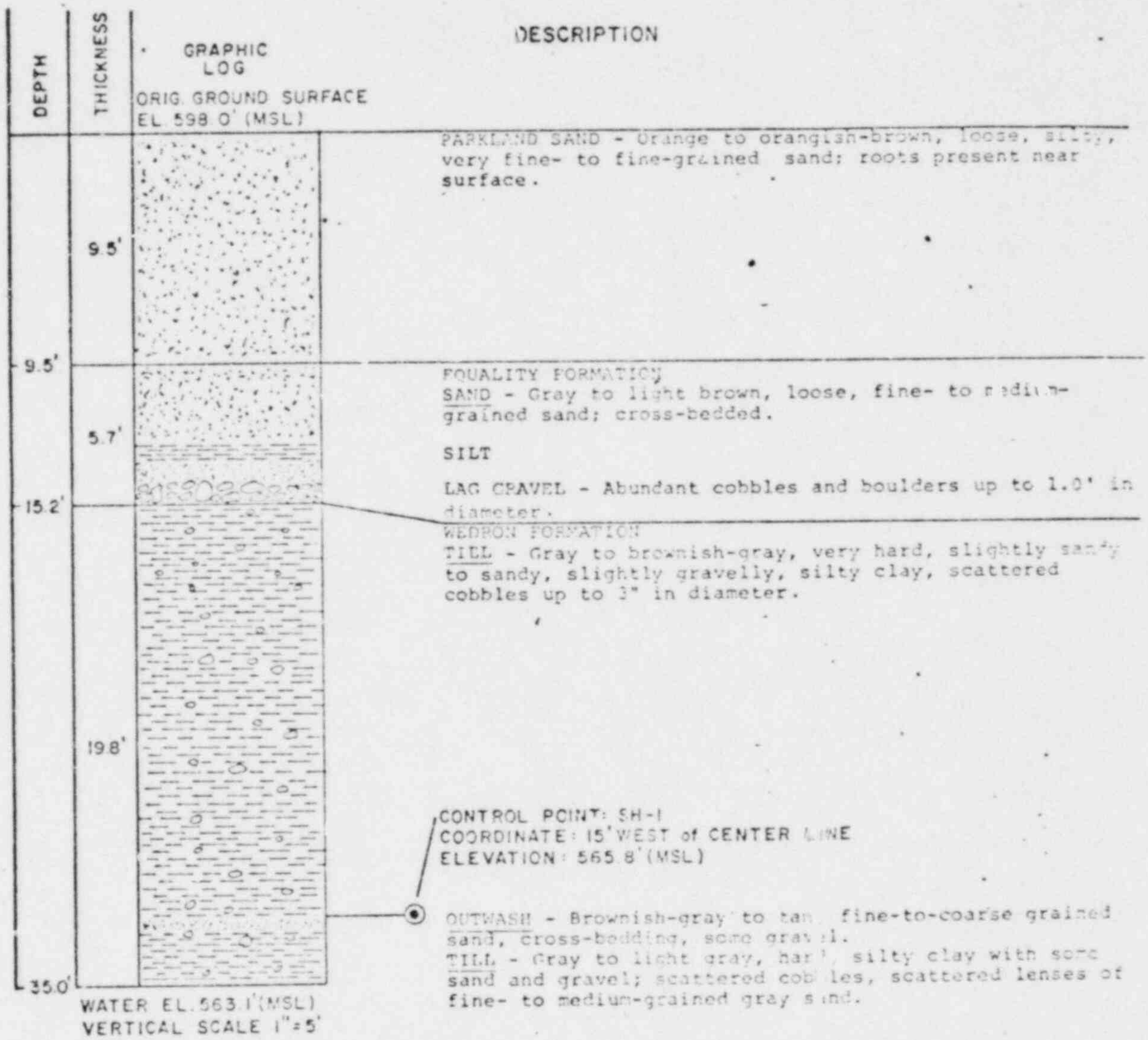
SKETCH MAP OF SCREENHOUSE
SHOWING SECTION LOCATIONS

Q362.3-27

Copy

SECTION SH-1

SCREENHOUSE - SOUTH WALL OF EXCAVATION



NOTE: LOCATION OF SECTION SH-1 IS SHOWN ON FIGURE Q362.3-3

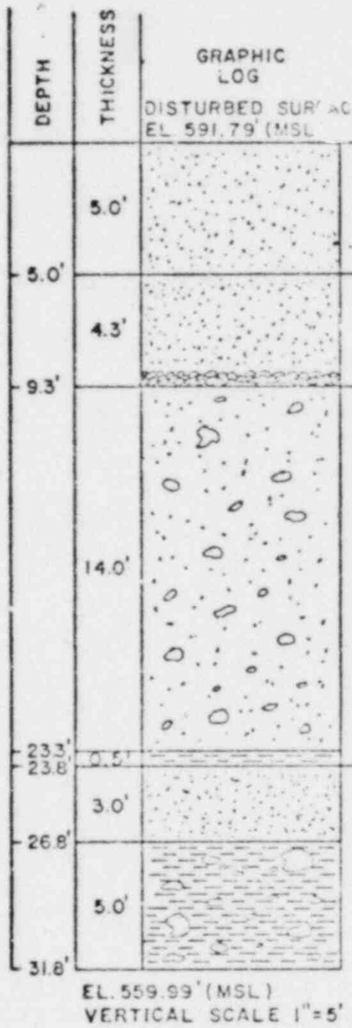
BRAIDWOOD STATION
FSAR

FIGURE Q362.3-4
SCREEN HOUSE
GEOLOGIC SECTIONS
(SHEET 1 OF 2)

Q362.3-30

SECTION SH-2

SCREENHOUSE - EAST WALL OF EXCAVATION



DESCRIPTION

PARKLAND SAND - Reddish-brown to orange, silty, very fine- to fine-grained sand.

EQUALITY FORMATION
SAND - Gray to tan, fine to coarse sand; cross-bedded, lenses of organics and coal fragments.

LAG GRAVEL - gravel to 6" diameter cobbles (EL. 582.49')

WEDRON FORMATION
TILL - Gray to brownish-gray, hard, silty, slightly sandy clay; scattered gravel and cobbles to 3" in diameter.

CONTROL POINT: SH-2
COORDINATE: 5'N COLUMN LINE CC
ELEVATION: 567.99' (MSL)

SILT - Gray, slightly sandy silt

OUTWASH - Brown to grayish-brown, fine- to medium-grained sand; coarse near base, scattered gravel to ½" diameter.

TILL - Gray to brownish-gray, hard, slightly sandy, silty clay with scattered gravel and cobbles.

NOTE: LOCATION OF SECTION SH-2 IS SHOWN ON FIGURE Q362.3-3

BRAIDWOOD STATION
FSAR

FIGURE Q362.3-4
SCREEN HOUSE
GEOLOGIC SECTIONS
(SHEET 2 OF 2)

Q362.3-31

QUESTION 362.4

"The coefficient of permeability, K, parameter of the fine sand stratum was determined by laboratory tests on reconstituted specimens. These tests do not simulate the in situ flow condition as in the case of a field pumping test. Why didn't you perform field permeability tests?

"Geologic Section 23 (Figure 2.5-50) and many borings drilled within the limits of the project (presented in FSAR section 2.5.4) reveal the presence of coarse grained material in the bottom 2 feet to 4 feet portion of the fine sand stratum. The gradation of the sand is from fine to coarse size; fine gravel and cobbles are also present. The cutface in the strip mining area, in the immediate vicinity of the project, revealed the presence of relatively more permeable material and showed marks to suspect that water is seeping in the bottom 2 to 4 feet zone of the fine sand stratum. It is feasible that the water from the Essential Service Cooling Pond (ESCP) may seep down to the more permeable zone beneath and travel horizontally. Does your seepage analysis cover this case? If so, submit the details of your analysis. If not, justify the rationality of your assumption that this seepage path does not exist and present your plan to confirm that this seepage path will not be operable."

RESPONSE

Laboratory permeability tests were performed on samples from the Equality Formation using the constant head permeability test in accordance with ASTM D-2434. The Equality Formation is composed primarily of fine- to medium-grained sands with some silt layers. In some borings and mapped sections, a 1 to 4 foot thick layer of coarse gravel, cobbles, and boulders in a fine- to medium-grained sand matrix occurs directly above the clay till of the Wedron Formation. Table 2.5-24 provides a summary of permeability tests performed on samples of SP and SM material from the Equality Formation. A total of 31 samples of the Equality Formation were tested using the constant head permeability test. The permeabilities for the SP and SM material ranged from 7.37×10^{-2} cm/sec to 3.658×10^{-4} cm/sec with an average permeability of approximately 6.7×10^{-3} cm/sec. Comparing these values with published data on coefficients of permeability for sands and gravels indicates that the coefficients of permeability determined through laboratory testing were within the range of the published data. Therefore, it was concluded that any further field testing could only provide a slight improvement of the laboratory values and that a field pumping test would not be required.

The fine-sand stratum is referred to as the Pleistocene-age Equality Formation. The upper few feet of the deposit is a fine sand, generally containing less than 15% silt, and having a consistency ranging from loose to medium dense. Below a depth of approximately 15 feet, the sand generally contains less than 5% silt, and has consistency ranging from medium dense to dense. Locally, coarse gravels, cobbles, and boulders in a fine sand matrix occur in a 1 to 4 foot thick layer overlying the Pleistocene-age Wedron Formation. A review of 354 boring logs and 31 mapped geologic sections indicates that the basal lag gravel is not continuous over the entire site area. The coefficient of permeability, K, of the Equality Formation, as determined by laboratory constant head permeability tests ranges from 3.658×10^{-4} cm/sec to 7.37×10^{-2} cm/sec with an average K value of approximately 6.7×10^{-3} cm/sec. Even though the basal 1 to 4 feet of the Equality Formation contains coarser material than the upper part of unit, the coefficient of permeability of the lower few feet will be near the average K value as the fine sand matrix is controlling the coefficient of permeability. In an unconsolidated deposit, the permeability will decrease for a given diameter as the standard deviation of particle size increases. The increase in standard deviation indicates a more poorly sorted sample, so that the finer material can fill the voids between the larger fragments.

The discussion of the Equality Formation exposures along a vertical strip-mining cut (Subsection 2.5.1.2.4.1.1.2) does not mention the presence of materials that is more permeable than the average Equality Formation sands. The groundwater found in the Equality Formation is underwater table conditions meaning that water infiltrates downward through a permeable unit due to gravity. This groundwater continues its downward movement until it encounters a less permeable unit. Once it encounters a less permeable unit, it will move downgradient to a discharge point. In the case of the seeps observed at the base of the Equality Formation, in the strip-mine cut, groundwater is infiltrating downward through the sands of Equality Formation until it reaches the aquiclude formed by the less permeable Wedron Formation clay till. The groundwater is then discharged as seeps along the cut face of the strip mine.

As stated above, the coefficient of permeability, K, of the Equality Formation averages approximately 6.7×10^{-3} cm/sec. Due to the sand matrix of the coarser basal material, the permeability of the lower part of the Equality Formation is near the average value. This, combined with the fact that the basal lag gravel is not a continuous deposit, indicates that the basal lag gravel does not exist as a seepage

path of higher permeability than the remainder of the Equality Formation. Therefore, the methods of seepage analyses and analysis conditions discussed in Subsections 2.5.6.6.1 and 2.5.6.6.2 are valid as this seepage path does not exist.

QUESTION 362.7

"What is the value of the coefficient of lateral earth pressure at-rest used in your design for the compacted backfill around the structures? Describe any conservatism involved in your earth pressure computations. Provide plots of earth pressure vs. depth used to design subsurface walls of various Category I structures."

RESPONSE

Figures Q362.7-1 and Q362.7-2 detail the lateral earth pressure plots versus depth for the auxiliary building and the lake screen house. Hydrostatic pressure on Category I structures are computed as detailed in Braidwood FSAR Subsection 2.5.4.10.1.3.1. The total static lateral pressure was obtained by combining soil and hydrostatic pressures. The response to FSAR Question 362.7 concerns only lateral earth pressure under static conditions.

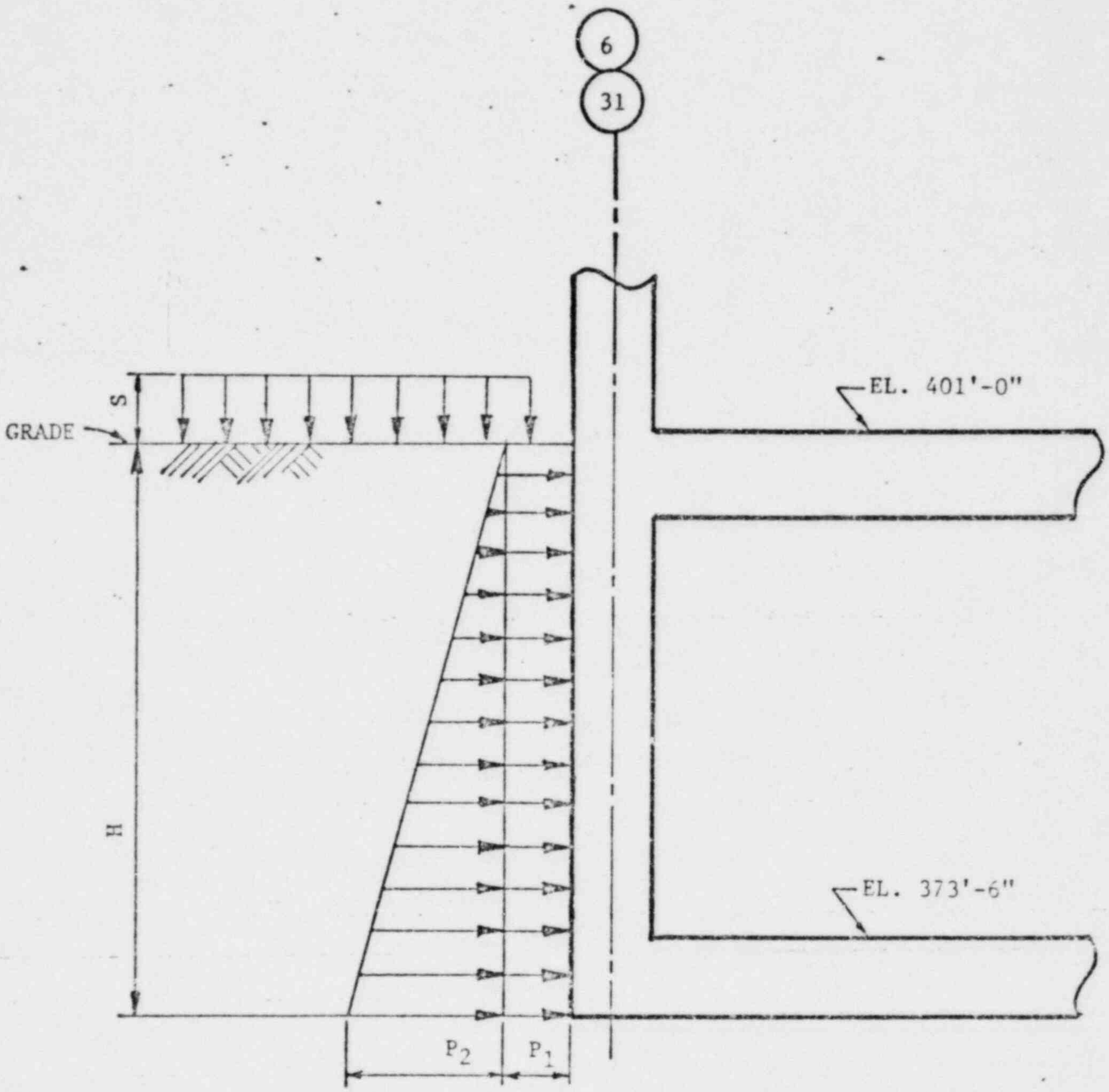
The coefficient of lateral earth pressure at rest is 0.88 for the compacted granular backfill around Category I structures.

Sources of conservatism in our earth pressure calculations are listed below:

1. All Category I subsurface walls and foundations are designed for a uniform construction surcharge load of 1,000 psf applied at grade for the normal loading conditions. Under normal plant operation conditions this surcharge load is not present.

2. The coefficient of lateral earth pressure at rest, K, for compacted granular backfill behind rigid walls is a function of the angle of internal friction, Ø. For the recompacted sands at Braidwood Station, the Ø angle is 34°. This is a conservative number for compacted sand. Therefore, the coefficient of lateral earth pressure at rest, K, for compacted granular backfill behind rigid walls is conservative.

3. The coefficient of lateral earth pressure at rest for a sand with in angle of internal friction (Ø) of 34° is 0.44. To increase the coefficient of lateral earth pressure at rest for the effect of compacting the soil in lifts the 0.44 coefficient was increased by a factor of two to give a coefficient of 0.88.

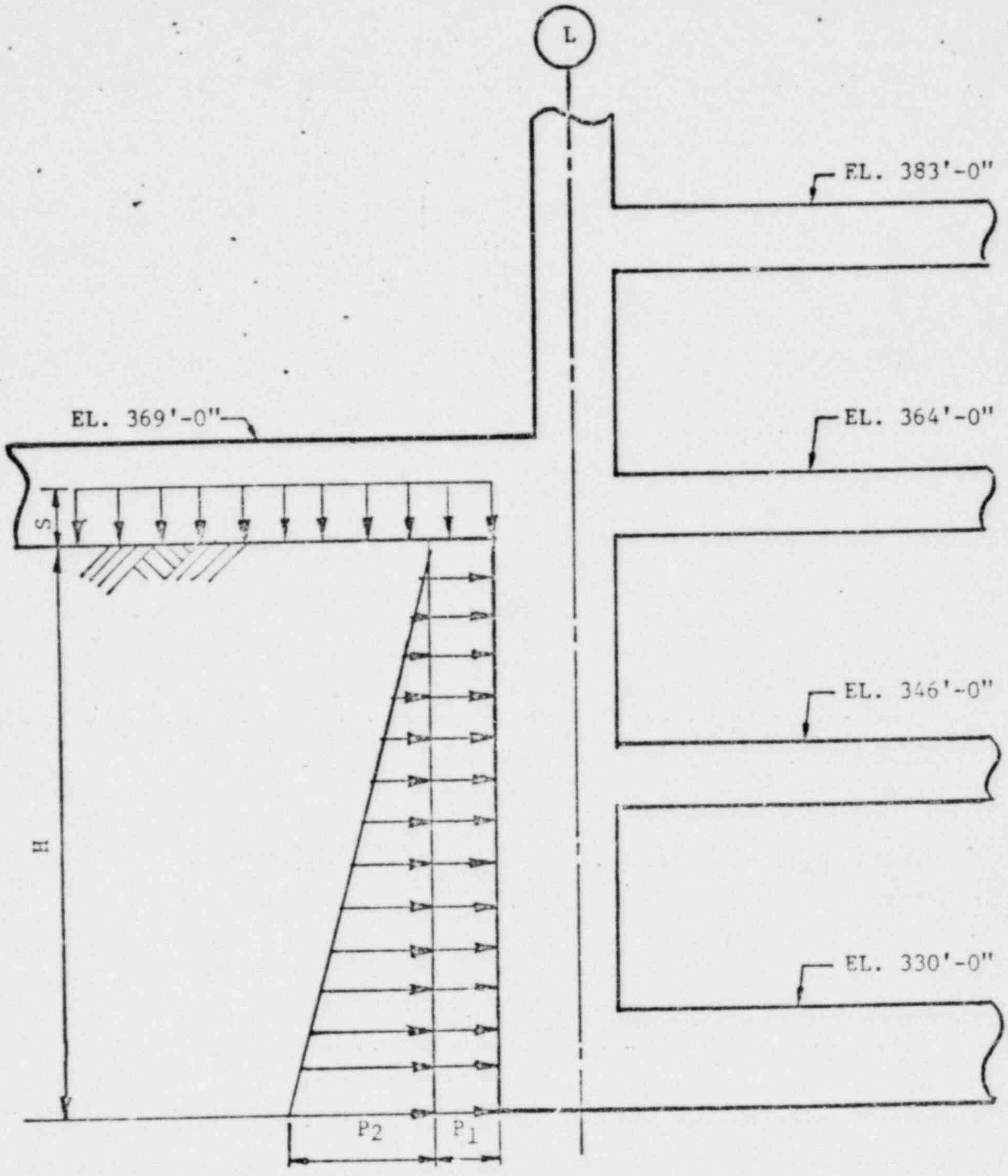


AUXILIARY BUILDING

P_1 = (ko) S
P_2 = (ko) γ_s (H)
ko = Coefficient of Lateral Earth Pressure (0.38)
S = Surcharge
H = Depth Below Grade
γ_s = Submerged Soil Density (68 pcf)
Water Table Assumed at Grade

BRAIDWOOD STATION
FSAR
Figure Q362.7-1
AUXILIARY BUILDING
LATERAL EARTH PRESSURE
(SHEET 1 OF 2)

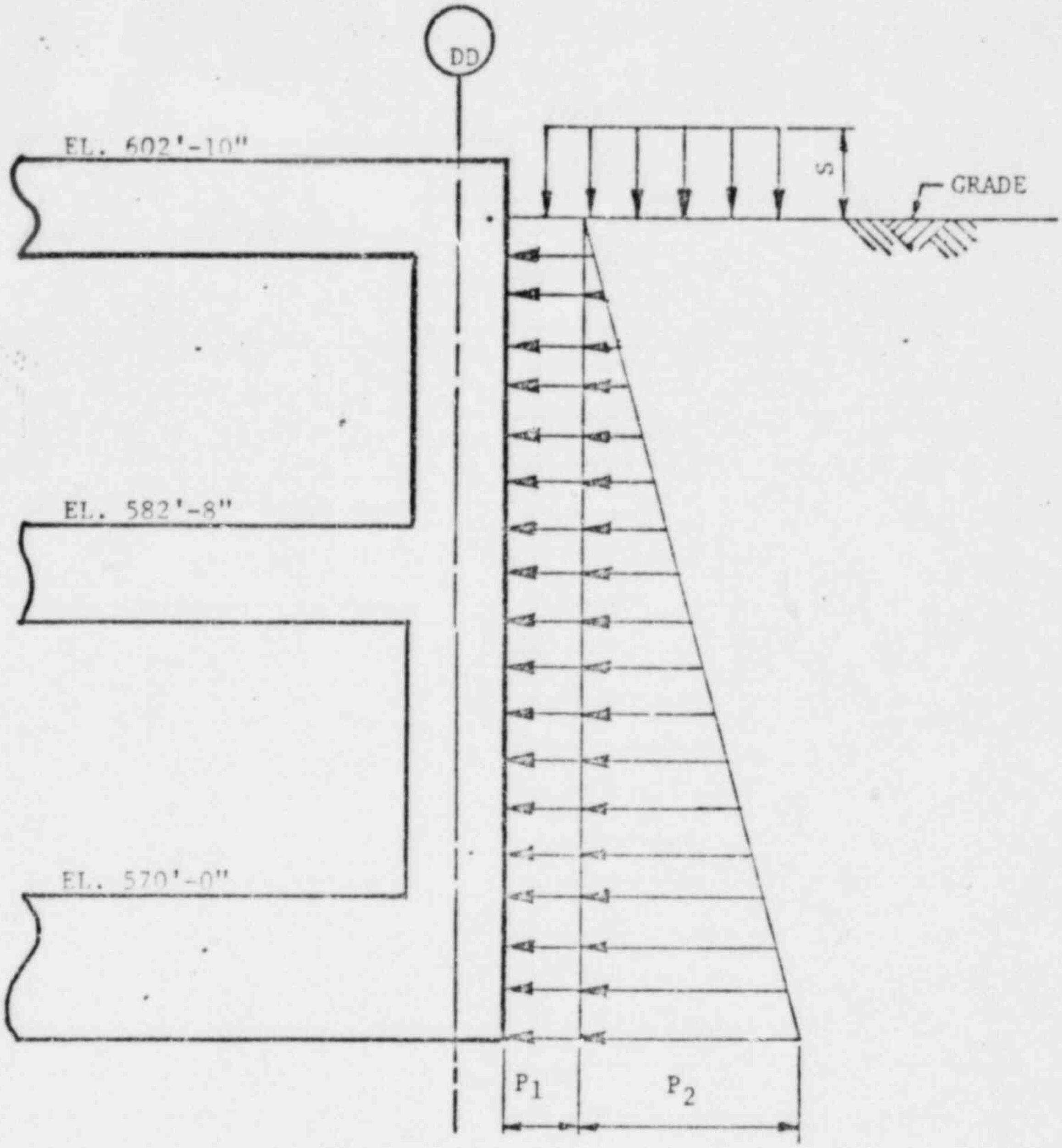
~~FIGURE 1~~



AUXILIARY BUILDING

P_1 = (ko) S
P_2 = (ko) γ_s (H)
ko = Coefficient of Lateral Earth Pressure (0.88)
S = Surcharge
H = Depth Below Grade
γ_s = Submerged Soil Density (68 pcf)
Water Table Assumed at Grade

BRAIDWOOD STATION
FSAR
FIGURE Q362.7-1
AUXILIARY BUILDING
LATERAL EARTH PRESSURE
(SHEET 2 OF 2)



LAKE SCREEN HOUSE

P_1 = (ko) S
P_2 = (ko) γ_S (H)
ko = Coefficient of Lateral Earth Pressure (0.88)
S = Surcharge
H = Depth Below Grade
γ_S = Submerged Soil Density (68 pcf)
Water Table Assumed at Grade

BRAIDWOOD STATION
FSAR
FIGURE Q362.7-2
LAKE SCREEN HOUSE
LATERAL EARTH PRESSURE

~~FIGURE 1~~

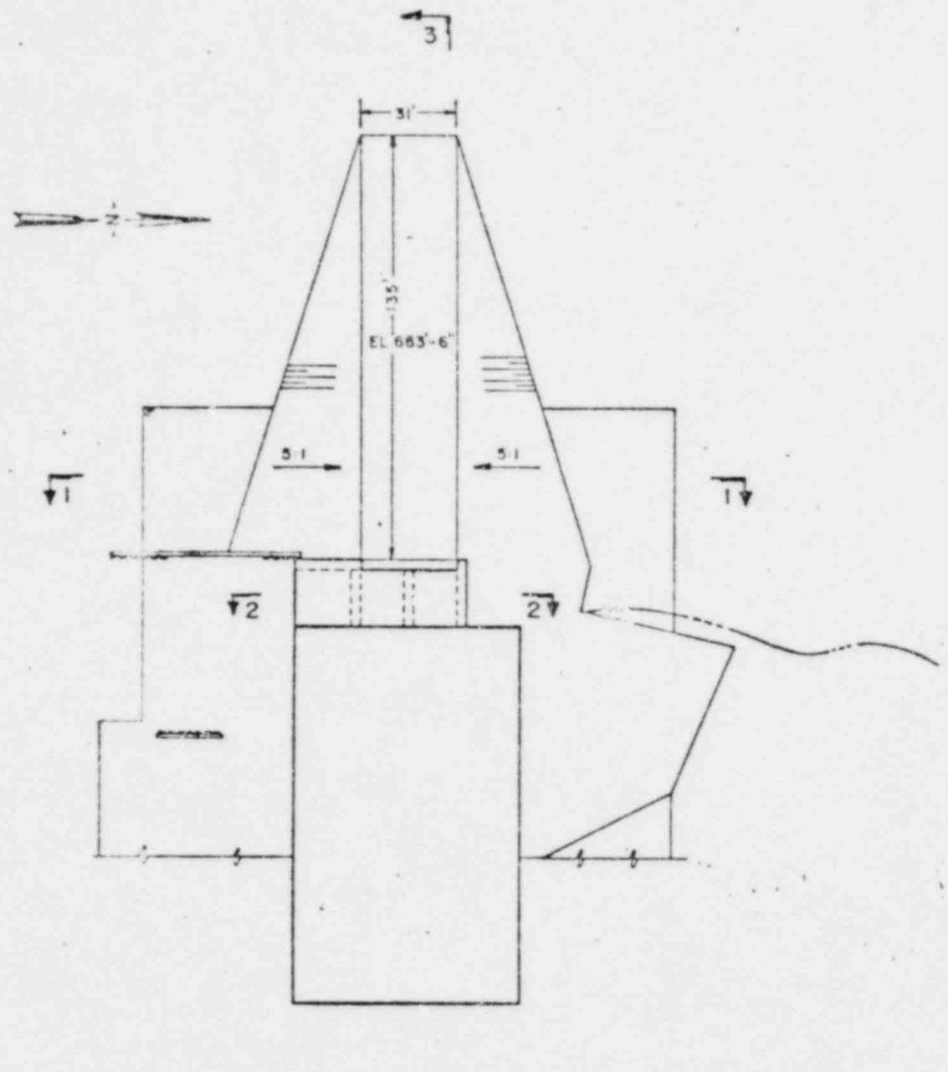
Q362.7-4

QUESTION 371.12

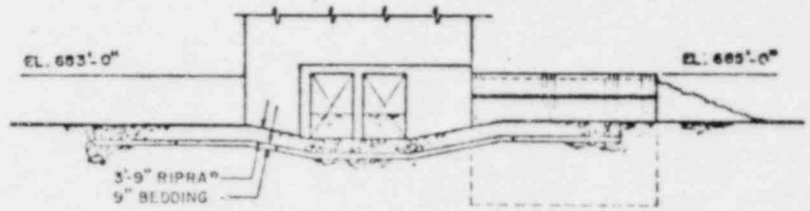
"Provide details of the canal or channel that connects the river screenhouse to the central river channel. Also, provide an expanded rating curve (at the intake) for flows from 0 to 2,000 cfs."

RESPONSE

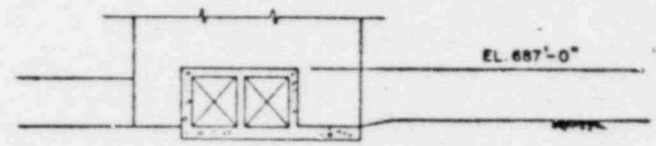
Figure Q371.12-1 shows the details of the channel that connects the river screen house to the central river channel. Figure Q371.12-2 shows the rating curve at the intake for low flows.



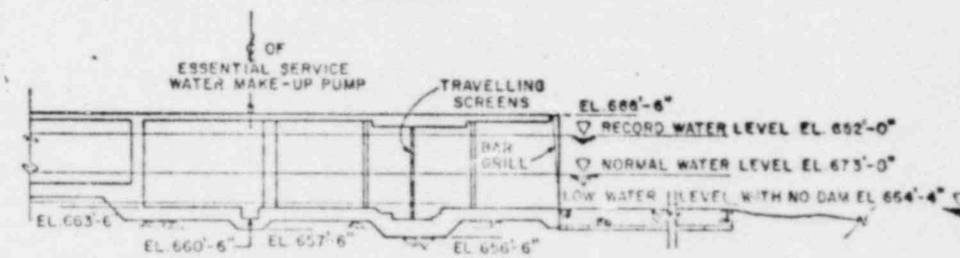
RIVER SCREEN HOUSE – PLAN



SECTION-1



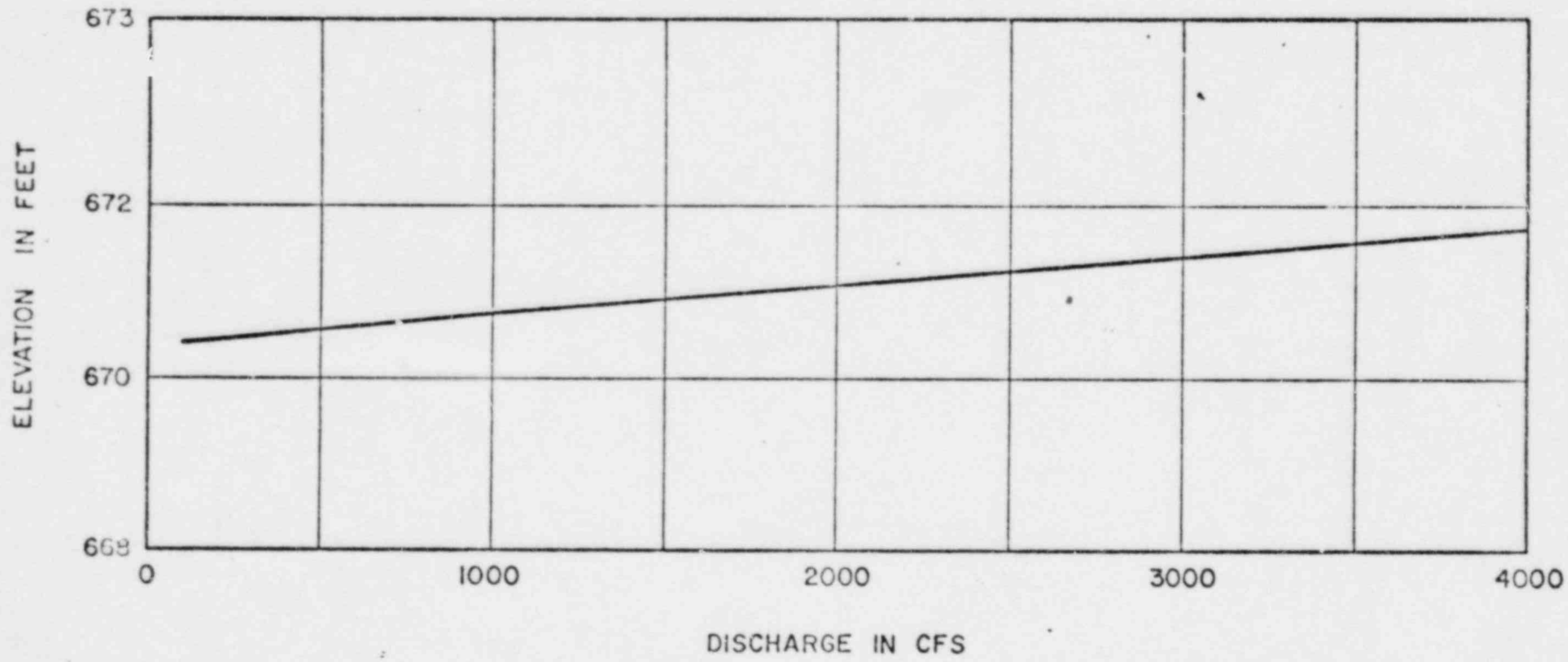
SECTION-2



SECTION-3

NOT TO SCALE

Q371.12-1


673
672
670
668
ELEVATION IN FEET
0
1000
2000
3000
4000
DISCHARGE IN CFS

Q371.12-2

QUESTION 371.14

"Provide detailed cross-sections of the river intake structure that clearly show the relationship between the invert elevation of the intake flow and/or sump flow, the invert of the dredged channel, the central river channel, minimum river levels and pump submergence requirements."

RESPONSE

The detailed cross-section of the river intake structure is provided with the response to Question 371.12.

Minimum pump submergence requirement is 22-1/2 inches. The pump intake is about 15-1/2 inches above the bottom of the sump which is at elevation 660 feet 6 inches.

QUESTION 371.15

"Provide the results of the 1978 pumping tests on well W-1 and a description of well modifications done as a result of the pumping tests."

RESPONSE

Refer to revised Subsections 2.4.13.1.2.3, 2.4.13.1.2.4, 2.4.13.1.2.5, 2.4.13.1.3, and 2.4.13.2.5.

QUESTION 371.16

"You state in Section 9.2.5 that the river screenhouse base mat elevation is 663.5 feet msl. However, in figure 1.2-16 the base mat elevation is shown at elevation 664.0 feet msl. A resolution of this inconsistency is necessary."

RESPONSE

Amendment 31 to the BYRON-FSAR reflects a top of base mat elevation equal to 663 feet 5 inches.

QUESTION 421.21

"The response to question 421.20 is unacceptable in that it does not address Regulatory Positions C.5, 6, 7, 8, and 10 of Revision 1 of Regulatory Guide 1.58. The response to question 421.20 refers to 'the revised compliance to Regulatory Guide 1.58 in Appendix A of the FSAR.' Appendix A of the FSAR (Amendment 28, October 1980) does not recognize the issuance or address Revision 1 of Regulatory Guide 1.58 (September 1980). Commit to meet Regulatory Positions C.5, 6, 7, 8, and 10 or indicate any specific positions which will not be met during the operations phase and describe your proposed alternatives."

RESPONSE

The compliance to Regulatory Guide 1.58 in Appendix A has been revised to address the items discussed in this question. Refer to the revised compliance for a response to this question.

groundwater movement radially outward. The piezometric surface generally reflects the ground surface, as expected in a water-table aquifer.

In the site area the Galena-Platteville dolomites are recharged by precipitation through the overlying glacial drift and discharge into the Rock River and its associated tributaries and in shallow domestic wells.

Water from the Galena-Platteville dolomites in the area is generally hard. Therefore, although it might be possible to obtain 10 to 50 gpm of hard water from the limestone with less than 200 feet of drilling, deeper drilling for softer water in the sandstone might be more economical. Relatively low yields, water hardness, and susceptibility of the aquifer to contamination because of thin drift, fractures, and solution channels do not favor development of the Galena-Platteville dolomites.

2.4.13.1.2.3 St. Peter Sandstone

Below the Galena-Platteville dolomites are the thin shales, sandstones, and limestones of the Glenwood Formation. This unit grades down into the thick sandstones of the St. Peter Sandstone. The Ordovician-age St. Peter Sandstone is permeable and has a relatively uniform lithology throughout the area. In the regional area, the St. Peter Sandstone is discharged primarily through wells for small municipalities, subdivisions, parks, and several industries that have water requirements generally less than 200 gpm (Table 2.4-22).

Numerous wells in the site area obtain water from the St. Peter Sandstone. A well in the southeastern quarter of Section 12, just north of the site, encountered the Glenwood Formation at 225 feet and the St. Peter Sandstone at 235 feet. The well was finished in 65 feet of the sandstone. A well in the western part of the site encountered the Glenwood Formation at 183 feet, penetrated the St. Peter Sandstone at 205 feet, and was finished in sandstone at 275 feet.

The uneroded thickness of the Glenwood Formation in the area ranges from about 18 to 32 feet. The full thickness of the St. Peter Sandstone in the area is about 420 to 450 feet. Artesian conditions prevail, and the static level is above the top of the sandstone unit. Based on available data, the average specific capacity of 17 wells within 2 miles of the site finished in the St. Peter Sandstone is about 2.7 gpm/ft of drawdown.

If the overlying Galena-Platteville dolomites were cased off and the casing extended through the Glenwood Formation into the sandstone, softer water than in any of the overlying units should be obtained.

2.4.13.1.2.4 Ironton and Galesville Sandstones

Deep-well logs in the area suggest that the Prairie du Chien Group is locally missing and that Cambrian-age dolomites of the Eminence Formation, Potosi Dolomite, and Franconia Formation underlie the St. Peter Sandstone. These units have a combined thickness of about 225 feet in the regional area. The onsite deep wells confirm that the Prairie du Chien Group and Eminence Formation are missing at the site. The combined thickness of the Potosi Dolomite and Franconia Formation at the site is about 125 to 140 feet.

Below the Franconia Formation are the Ironton and Galesville Sandstones comprising a portion of the aquifer which is about 150 feet thick in the regional area. In the site area, the Ironton and Galesville Sandstones are about 105 to 115 feet thick. The sandstones are discharged primarily through wells to industries and municipalities. Regionally, the Ironton and Galesville Sandstones are considered the best bedrock aquifer in northern Illinois because of their consistent permeability and thickness (Reference 18 and Table 2.4-22). Yields on the order of hundreds of gallons per minute may be obtained from the Ironton and Galesville Sandstones in wells less than 1000 feet deep. As reflected by the relatively high specific capacities of the Byron Station wells, the Ironton and Galesville Sandstones are a major water-producing zone in the Byron Station wells. Results of water quality analyses on samples collected from the onsite water supply wells are presented in Table 2.4-23.

2.4.13.1.2.5 Mt. Simon Sandstone

Below the Ironton and Galesville Sandstones is the Eau Claire Formation, about 405 feet thick. The basal part of the Eau Claire Formation and the underlying Mt. Simon Sandstone (which is about 1430 feet thick) form the basal Cambrian-age Mt. Simon Aquifer. Wells yielding many hundreds of gallons per minute have been finished in the Mt. Simon Sandstone, which contains fresh water to depths of about 2000 feet (Reference 19). It is recharged in outcrop areas in Wisconsin and from vertical leakage from overlying aquifers. The Mt. Simon Aquifer is a major water-producing zone in the Byron Station wells. Water quality analyses for the wells are presented in Table 2.4-23.

2.4.13.1.3 Onsite Use

Two groundwater wells have been installed, developed, and tested at the plant for potable water supply and for demineralizer water. The groundwater will be filtered and stored in

a 150,000-gallon storage tank prior to usage. The two plant wells will also serve as a backup system for makeup to the essential service water cooling towers.

Water for the demineralizer will be required at the rate of about 825 gpm for the first several months, and thereafter at an average rate of 450 gpm. On the basis of 50 gallons per capita per day and a design population for plant personnel of 300 people, the average annual groundwater potable supply required will be 15,000 gallons per day or about 10 gpm. Considering monthly, daily, and hourly above-average demand, the potable water supply developed will be on the order of 20 gpm. In the unlikely event that makeup to the essential service cooling tower is not available from the Rock River, the groundwater wells should provide a maximum of 1600 gpm for the duration of safe shutdown. The demineralizer and potable water use would not be additive to the 1600 gpm use as they would not be required simultaneously.

The two Byron Station deep wells (W1 and W2) were completed in the Ironton and Galesville Sandstones of the Cambrian-Ordovician Aquifer in 1974. The locations of these wells are shown in Figure 2.4-24. The wells were cased through the Galena-Platteville dolomites as indicated in Table 2.4-29. The water wells were open from the Ancell Group through the Ironton and Galesville Sandstones; as noted in Figure 2.4-23, the Ironton and Galesville Sandstone were the major producing zones. Grouting at the Byron Station did not extend into the formations open to the site water wells; therefore, the specific capacities are unaffected by onsite grouting.

The specific capacities obtained during well development pumping tests were 10.3 gpm/ft of drawdown at 620 gpm for 12 hours in Well 1 (east well) and 9.6 gpm/ft of drawdown at 1150 gpm for 24 hours in Well 2 (west well). The pumping rate for Well 1 was relatively constant for the initial 12 hours, then was varied between 433 and 930 gpm during the last 12 hours of the test. The pumping rate for Well 2 was relatively constant for the entire 24 hours.

In addition to the Byron Station water wells, a temporary water well (TW-1) was installed for construction supply. The location of this well is shown on Figure 2.4-24. The well construction was as follows: 16-inch diameter cased borehole to 30 feet, 15-inch diameter open borehole to 271 feet, and 12-inch diameter open borehole to 600 feet. This well was primarily open to the Galena-Platteville dolomites and the St. Peter Sandstone. The specific capacity of this well was 4.4 gpm/ft of drawdown at 840 gpm after 24 hours. The lower specific capacity may be attributed to the smaller borehole diameter and the lower productivity of the upper portions of the Cambrian-Ordovician Aquifer.

Two water wells were installed to provide water for the grouting operation. The wells are presently capped and are not in use. Well TW-2 is an 8-inch diameter well cased to 240 feet and open borehole to 505 feet. Well TW-3 is an 8-inch diameter well cased to 241.5 feet and open borehole to 500 feet. Pump settings are at 240 feet and 241.5 feet, respectively. Both wells produce water from the St. Peter Sandstone.

An aquifer pumping test was performed in November 1978 in order to demonstrate the ability of the station deep water wells to provide make-up to the essential service water system during the 30-day period for safe shutdown. The aquifer pumping test consisted of pumping water well W-1 at a continuous rate of 840 gpm while monitoring groundwater levels in water well W-2 (Ironton-Galesville Sandstones), the grouting supply well TW-2 (St. Peter Sandstone), and an observation well TW-4 installed in the Ironton-Galesville Sandstone approximately 300 feet from water well W-1 on the line connecting wells W-1, TW-2, and W-2. The aquifer pumping test consisted of a 22-hour pumping period followed by a recovery period of one and one-half hours. Locations of the wells are shown in Figure 2.4-24.

The aquifer pumping test results indicated that the aquifer transmissivity was 20,000 gpd/ft, the storage coefficient was 2.0×10^{-4}, and the specific capacity was 8.5 gpm/ft. The results also indicated that the pump setting would have to be deepened and that the specific capacity had decreased 15% since the well was first installed. On the basis of subsequent caliper logging and well depth measurements, the apparent decline in well productivity was attributed to caving from the St. Peter Sandstone with blockage of the productive aquifer in the lower portion of the well.

Well modifications performed in wells W-1 and W-2 after the 1978 pumping test consisted of reaming and casing off the caving St. Peter Sandstone, deepening the wells through the Ironton-Galesvillle Sandstones and into the upper portion of the Mt. Simon Sandstone, increasing the pumps' lift-capacity, and lowering the pump settings by 100 feet. Deep well construction details are summarized in Table 2.4-29 and presented schematically in Figure 2.4-29. The modified water wells are open from the Franconia Formation, through the Ironton and Galesville Sandstones, and into the Mt. Simon Sandstone. As noted in Figure 2.4-23, the Ironton and Galesville Sandstones and the Mt. Simon Aquifer are the major producing zones.

The specific capacities obtained during well development pumping tests in the modified wells were 12.3 gpm/ft of drawdown at 1330 gpm for 12 hours in Well 1 and 12.2 gpm/ft of drawdown at 1210 gpm for 9 hours in Well 2. Whereas the pumping rate for Well 1 was relatively constant for the entire 12 hour test, the pumping rate for Well 2 was relatively constant for only the initial 9 hours, then was varied between 1200 and 1600 gpm during the last 3 hours of the test. The available drawdown in the two wells is approximately 125 feet based on a static water level of 225 feet and a pumping level of 350 feet.

An aquifer pumping test was performed in July 1980 after the well modifications were completed in order to demonstrate the ability of the modified station water wells to provide the required make-up to the essential service water system. The test consisted of pumping Well W-1 at 790 gpm for 24 hours. Drawdown and recovery were again measured in Well W-2. The test results indicated that the aquifer transmissivity is 40,000 gpd/ft, the storage coefficient is 2.4×10^{-4}, and the specific capacity is 13.2 gpm/ft.

2.4.13.2 Sources

2.4.13.2.1 Present Regional Groundwater Use

Most of the water for domestic, municipal, and industrial use in the region is obtained from groundwater sources. The major unit is the St. Peter Sandstone within the Cambrian-Ordovician Aquifer, although minor supplies commonly are obtained from the shallower glacial drift and dolomite aquifers.

There are seven public water supply systems within 10 miles of the plant site. All use groundwater wells for water supply. All obtain their supplies from the Cambrian-Ordovician Aquifer or the Mt. Simon Aquifer, which are dependable and capable of high yields. Table 2.4-22 lists and gives details of each of the seven public water supply systems (Reference 20). The locations of the public pumping centers are shown on Figure 2.4-25.

The total groundwater pumpage in Ogle County ws 11.82 mgd in 1965, the highest during the period from 1960-1970. After 1965, pumpage declined at an average rate of 240,000 gpd per year until 1970. In 1970, total pumpage in the county was 10.62 mgd; of this total, 12% was from glacial drift wells, 20% was from shallow dolomite wells (Galena-Platteville dolomites), and 68% was from sandstones wells (Cambrian-Ordovician Aquifer). Pumpage for public supplies in 1970 accounted for 53% of the total groundwater withdrawal in Ogle County. All recorded public pumpage is from sandstone wells (Reference 21).

A piezometric surface map of the site vicinity, as measured in the Galena-Platteville dolomite, is shown in Figure 2.4-24. As shown on the map, the main plant structures are situated on a recharge area of the Galena-Platteville portion of the aquifer. The aquifer is recharged by direct infiltration of precipitation through the overlying, thin glacial drift. The Galena-Platteville dolomites have little primary permeability and precipitation moves downward from the overlying drift into solution enlarged joints that provide secondary permeability. Groundwater flows radially from the site, but the principal discharge boundaries are the Rock River to the west and northwest of the site and Black Walnut Creek to the east and southeast of the site.

Table 2.4-26 lists recorded active, domestic, or agricultural groundwater wells east of the Rock River with 2.25 miles of the site. These wells are primarily completed in the Galena-Platteville dolomites. Figure 2.4-27 illustrates the location of each well within 2.25 miles from the plant site. Domestic and agricultural wells west of the Rock River are not shown on Figure 2.4-27 because the river is a common discharge boundary for wells, east and west of the river, which are completed in the Galena-Platteville dolomites.

Pump tests were performed on June 20 and July 2, 1974 in two domestic water supply wells that are completed in the Galena-Platteville dolomites. These two wells are located at the western edge of the site along Razorville Road. From the pump tests, aquifer parameters were derived based on July 1, 1974, piezometeric levels. Based on estimated saturated thicknesses of 111 feet and 90 feet at these two wells, the hydraulic conductivity of this portion of the aquifer was 6.3 gpd/ft^2 and 22.2 gpd/ft^2 respectively. The effective porosity of this portion of the aquifer is estimated to range from 0.05 to 0.10.

2.4.13.2.4 Future Site Groundwater Use

There are no anticipated changes in the present pattern of groundwater use in the site area.

2.4.13.2.5 Effects of Plant Groundwater Use

The projected effects of plant groundwater withdrawals of approximately 470 gpm have been evaluated using the Theis equations (Reference 25) with assumed values of 17,000 gpd/ft and 3.5×10^{-4} for the coefficients of transmissivity and storage (Reference 18). The projected effects were reevaluated using the values of 40,000 gpd/ft and 2.5×10^{-4} determined from the 1980 aquifer pumping test. Theoretical distance-drawdown and time-drawdown curves were constructed in order to determine the anticipated shape of the cone of depression and the radius of influence of the Byron Station wells. These theoretical curves indicate that the Byron groundwater withdrawals should not impose measurable interference drawdowns on the nearest public water

supply wells completed in the Cambrian-Ordovician or Mt. Simon Aquifer. Indeed, the groundwater withdrawals at the Byron site will intercept groundwater that otherwise would naturally discharge from the aquifer into the Rock River.

The effects of pumping from Byron Station water wells will be minimal on domestic wells completed in the Galena-Platteville dolomites. As described in Subsection 2.4.13.2.3, the Galena-Platteville dolomites in the site vicinity are hydraulically separated from the lower portion of the Cambrian-Ordovician Aquifer by the Harmony Hill Shale Member of the Glenwood Formation. In addition, the Byron Station water wells are cased through the Galena-Platteville dolomites and the underlying Ancell Group (St. Peter Sandstone). Groundwater in the Galena-Platteville dolomites is perched on the Harmony Hill Shale Member and initially water levels in this aquifer will not be lowered by pumping for daily plant use from the lower portion of the Cambrian-Ordovician Aquifer and Mt. Simon Aquifer. As pumpage from the plant water wells continues with time, minor vertical leakage may occur through the Harmony Hill Shale Member. If recharge by rainfall infiltration is not considered, water levels in domestic and agricultural wells in the site vicinity may be lowered slightly as a result of long-term pumping of groundwater from the Byron Station water wells. Measurements made during the 1980 aquifer pumping test verified that the offsite drawdown effects in the Galena-Platteville dolomites and Ancell Group (St. Peter Sandstone) will be very minor.

2.4.13.3 Accident Effects

As described in Subsection 2.4.12, the largest tanks located outside the containment building and containing radioactive effluents are the boron recycle holdup tanks. These tanks are located in a portion of the Seismic Category I Auxiliary Building where the floor elevation is 815.0 feet. Each of the recycle holdup tanks has a capacity of 125,000 gallons. The design-basis radionuclide content of each tank is given in Table 2.4-20.

The plant grade elevation is 869.0 feet. The site area piezometric surface map, as measured on July 1, 1974, showed a groundwater elevation of approximately 840.0 feet. However, taking seasonal variations into account, it is conservatively assumed that the groundwater elevation at the time of the postulated accident would be 799.0 feet. The nearest offsite downgradient groundwater user is located at about 1960 feet from the auxiliary building (Well 55, Figure 2.4-27). The prevailing hydraulic gradient between the Auxiliary Building and the nearest well is 0.023. There is a spring located at a distance of about 3630 feet downgradient from the Auxiliary Building. The hydraulic gradient between the Auxiliary Building and this spring is 0.011 which is flatter than the gradient to the nearest offsite well. It is, therefore, concluded that the critical path

of movement of accidentally released effluents will be from the Auxiliary Building to the nearest downgradient offsite well. It is anticipated that there will be no future groundwater user within 1600 feet of the Auxiliary Building. Therefore, the results of the analysis of accident effects for the existing well are valid for any future wells also.

TABLE 2.4-21

GENERALIZED SITE HYDROGEOLOGIC COLUMN

UNIT	APPROX. DEPTH TO TOP (ft)	APPROX. DEPTH TO BOTTOM (ft)	APPROX. THICK-NESS (ft)	HYDROGEOLOGY
Glacial drift	0	16	16	Not an aquifer
Cambrian-Ordovician Aquifer*	16	915	900	Major aquifer
Galena and Platteville Groups	16	200	190	Minor unit
St. Peter Sandstone	225	675	450	Important unit
Ironton and Galesville Sandstones	805	915	110	Important unit
Eau Claire Formation	95	1320	405	Not an aquifer
Mt. Simon Sandstone	1320	2750	1430	Important unit, salty at depth

*Only the most important units are listed.

TABLE 2.4-23

WATER QUALITY DATA - BYRON STATION WATER WELLS

	December 1974 BYRON WATER WELL 1 (EAST)	July 1980 BYRON WATER WELL 1 (EAST)	December 1974 BYRON WATER WELL 2 (WEST)
Calcium; mg/l as $CaCO_3$	150.0	153	143.0
Magnesium; mg/l as $CaCO_3$	119.0	144	139.0
Sodium; mg/l as Na	4.6	2.6	3.6
Potassium; mg/l as K	--	3.4	--
Total Alkalinity; mg/l as $CaCO_3$	301.0	301	320.0
Sulfate; mg/l as SO_4	5.5	8	5.1
Chloride; mg/l as Cl	3.2	<1	2.7
Nitrates; mg/l as NO_3	7.8	<0.01	<0.1
Silica; mg/l as SiO_2	5.1	9.1	8.1
Total Dissolved Solids; mg/l	96.0	328	582.0
Conductivity; μ-mohs at 25° C	502.0	600	542.0
Iron; mg/l as Fe	0.30	0.46	0.28
Manganese; mg/l as Mn	0.01	0.02	<0.01
Turbidity; FTU	20.0	--	8.0
pH at 25° C	8.0	7.3	7.6
Total Organic Carbon; mg/l	5.0	--	--
Carbon Dioxide; mg/l as CO_2	8.0	38	30.0

1. Groundwater samples collected December 1974 were analyzed by Aqua Systems Corporation, Chicago, Illinois.

2. Groundwater samples collected July 1980 were analyzed by Aqualab Inc., Streamwood, Illinois; except conductivity, which was measured in the field.

New

BYRON-FSAR

TABLE 2.4-29
DEEP WELL CONSTRUCTION DETAILS

Well(1)	Location	Elevation, Ft, msl	Date Completed	Well Depth, Ft	Diameter, Inches	Pump Test Data				Static Water Level, Ft (Date)	Comments
						Date	Specific Capacity, gpm/Ft	Pumping Rate, gpm	Duration,(3) Hours		
Deep Well No. 1 (east well)	T24N, R10E, [illegible]'S, [illegible]10'W, NE NE NE NE Sec 24 31 + 03 N 45 + 20 E (Plant Coordinates)	875.5(2) Original Grade 876	December 1974 June 1980	0-20' 0-241' 241-462' 462-834' 0-700' 834-1500'	24" Cased 20" Cased 19" Open 15" Open 16" Cased 15" Open	12/74 11/78 06/80 07/80	10.3 8.5 12.3 13.2	620 840 1300 790	12 3 1/2 12 24	185 (12/74) 186 (11/78) 217 (06/80) 224 (07/80)	Principal producing units are St. Peter Sandstone, and upper portion of Ironton-Galesville Sandstones. Well modified in 1980.(4) Principal producing units are Ironton-Galesville Sandstones and Mt. Simon Aquifer.
Deep Well No. 2 (west well)	T24N, R10E, [illegible]'S, 520'E, NW SW SW SE Sec 24 2[illegible] + 16 N [illegible] + 90 E (Plant Coordinates)	870.4(2) Original Grade 870	October 1974 December 1979	0-20' 0-230' 230-549' 549-853' 0-700' 853-1500'	24" Cased 20" Cased 19" Open 15" Open 16" Cased 12" Open	11/74 12/79	9.6 12.2	1150 1210	24 9 1/2	187 (11/74) [illegible]1 (11/78) 206 (12/79) 218 (07/80)	Principal producing units are St. Peter Sandstone, and upper portion of Ironton-Galesville Sandstones. Well modified in 1979.(4) Principal producing units are Ironton-Galesville Sandstones and Mt. Simon Aquifer.

NOTES

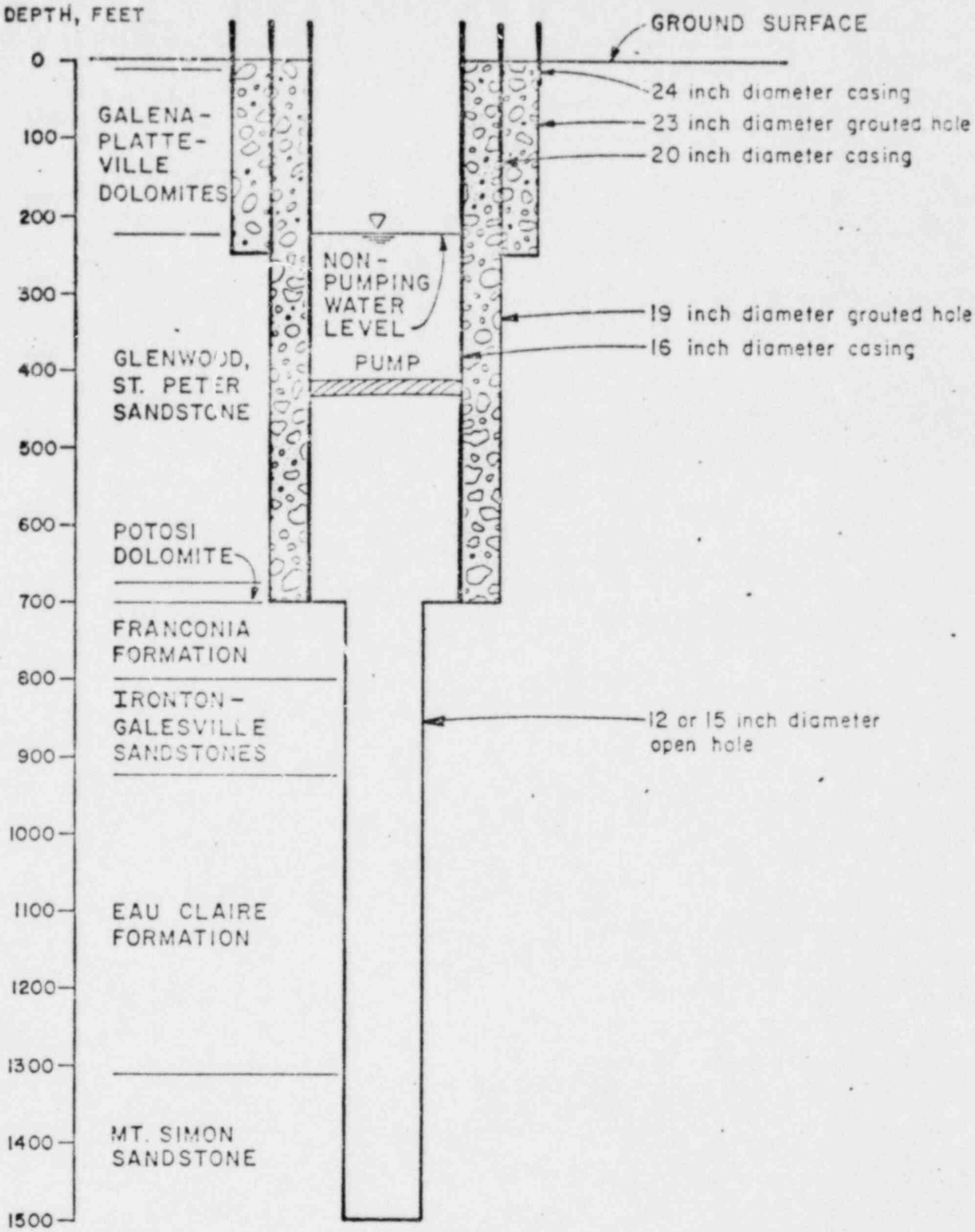
1. Well No. 2 was drilled before well No. 1; therefore, early drilling records on file with the Illinois State Geological Survey have reversed well numbers, i.e., well No. 2 is "well No. 1."

2. Measured at top of pitless adapter casing.

3. Duration refers to portion of test used to calculate specific capacity. Total test duration may have been longer than indicated.

4. Pumping equipment consists of 10-stage Byron Jackson submersible pumps installed 412 ft. deep. Maximum pumping level is 350 ft. Available drawdown is approximately 125 ft.

2.4-78



NOTES

1. NOT TO SCALE.
2. WELL CONSTRUCTION DETAILS ARE SUMMARIZED IN TABLE 2.4-29.
3. PITLESS ADAPTER NOT SHOWN.

BYRON STATION
FSAR
Figure 2.4-29

SKETCH OF DEEP WELL CONSTRUCTION

New

TABLE 2.5-35

RESULTS OF CYCLIC SHEAR STRENGTH

TESTS ON SAND DEPOSITS - PHASE 1

	TEST NO.	TYPE* OF TEST SPECIMEN	EL. (ft)	D_{50}* mm	FC* (%)	γ_{max} (lb/ft^3)	γ_{min} (lb/ft^3)	γ_d (lb/ft^3)	D_r (%)	σ_{3c} (lb/ft^2)	$\frac{\sigma_d}{2\sigma_{3c}}$	N (cycle) $\pm$ c: 1L	2.5%	5%	7.5%	10%
	BR-101a	R	*1	.25	1	104.8	91.6	101	74	1500	.426	13	16	19	24	44
Test Series 1	BR-101b	R	*1	.25	1	104.8	91.6	101	74	1500	.397	18	17	25	30	40
Test Series 1	BR-101c	R	*1	.25	1	104.8	91.6	101	74	1500	.497	6	8	10	12	18
Test Series 1	BR-101d	R	*1	.25	1	104.8	91.6	101	74	1500	.443	11	15	19	24	37
Test Series 1	BR-102a	R	*1	.25	1	104.8	91.6	103	88	1500	.457	15	20	28	62	68
Test Series 1	BR-102b	R	*1	.25	1	104.8	91.6	103	88	1500	.491	11	16	23	39	47
Test Series 1	BR-102c	R	*1	.25	1	104.8	91.6	103	88	1500	.404	30	37	44	64	81
Test Series 1	BR-103a	R	*1	.25	1	104.8	91.6	98	52	1500	.398	4	5	7	8	10
Test Series 1	BR-103b	R	*1	.25	1	104.8	91.6	98	52	1500	.378	12	14	16	17	[illegible]
Test Series 1	BR-103c	R	*1	.25	1	104.8	91.6	98	52	1500	.361	12	13	15	16	[illegible]
	BR-103d	R	*1	.25	1	104.8	91.6	98	52	1500	.388	7	8	10	11	14
Test Series 2	BR-104a	R	*2	.15	11	110.9	92.1	110	93	1500	.494	30	36	51	102	108
Test Series 2	BR-104b	R	*2	.15	11	110.9	92.1	110	93	1500	.470	30	39	58	105	110
Test Series 2	BR-104c	R	*2	.15	11	110.9	92.1	110	93	1500	.425	58	74	115	174	195
Test Series 2	BR-105a	R	*2	.15	11	110.9	92.1	107	80	1500	.494	17	21	26	38	66
Test Series 2	BR-105b	R	*2	.15	11	110.9	92.1	107	80	1500	.451	21	26	33	48	80
Test Series 2	BR-105c	R	*2	.15	11	110.9	92.1	107	80	1500	.426	31	36	43	70	[illegible]
Test Series 2	BR-106a	R	*2	.15	11	110.9	92.1	103.5	63	1500	.488	10	12	14	16	[illegible]
Test Series 2	BR-106b	R	*2	.15	11	110.9	92.1	103.5	63	1500	.447	11	13	15	19	[illegible]

*1 See Note 1

*2 See Note 2

New

New

TABLE 2.5-35 (Cont'd)

	TEST NO.	TYPE* OF TEST SPECIMEN	EL. (ft)	D_{50}* mm	FC* (%)	γ_{max} (lb/ft^3)	γ_{min} (lb/ft^3)	γ_d (lb/ft^3)	D_r (%)	σ_{3c} (lb/ft^2)	$\frac{\sigma_d}{2\sigma_{3c}}$	N (cycle) $\pm\epsilon_c$ 1%	2.5%	5%	7.5%	10%
Test Series 3	BR-107a	R	*3	.17	20	112.8	92.9	109.2	85	1500	.494	26	31	46	95	106
	BR-107b	R	*3	.17	20	112.8	92.9	109.2	85	1500	.447	35	41	50	106	128
	BR-107c	R	*3	.17	20	112.8	92.9	109.2	85	1500	.425	35	35	54	125	172
	BR-108a	R	*3	.17	20	112.8	92.9	106	70	1500	.489	14	17	22	29	68
	BR-108b	R	*3	.17	20	112.8	92.9	106	70	1500	.426	25	29	35	47	77

*Key:

R: test on reconstituted test specimens.

D_{50}: 50% of sample is smaller than this grain size.

FC: fines content - percent passing the 1200 mesh sieve (0.074 mm).

*3 See Note 3

New

Notes: 1. The test specimen for Test Series 1 was composed of material from Test Pit No. 3, block samples Nos. 37 and 38. FSAR Table 2.5-31 details the elevations of the block samples.

2. The test specimen for Test Series 2 was composed of material from Test Pit No. 3, block samples Nos. 32, 37, and 38. FSAR Table 2.5-31 details the elevations of the block samples.

3. The test specimen for Test Series 3 was composed of material from Test Pit No. 3, block samples Nos. 37 and 38 and Test Pit No. 5, block sample No. 21. FSAR Table 2.5-31 details the elevations of the block samples.

TABLE 2.5-36

SUMMARY OF SKEMPTON "B" VALUES FOR PHASE 1 TESTS

TEST NUMBER	"B" VALUE
BR-101a	0.95
BR-101b	0.96
BR-101c	0.95
BR-102a	0.98
BR-102b	1.0
BR-102c	0.98
BR-103a	0.95
BR-103b	0.98
BR-103c	0.95
BR-103d	0.95
BR-104a	0.98
BR-104b	0.98
BR-104c	1.0
BR-105a	0.95
BR-105b	0.98
BR-105c	0.98
BR-106a	0.95
BR-106b	0.98
BR-107a	0.95
BR-107b	0.96
BR-107c	0.95
BR-108a	0.95
BR-108b	1.0

NOTE: The source of the test specimens referenced above is detailed in Table 2.5-35.

and restraining devices are consistent with the dynamic and static analyses of the systems.

All Seismic Category I piping except for the reactor coolant loops is designed by Sargent & Lundy, including the location of supports and restraints. The field location of supports and restraints is done only for non-Seismic Category I piping, 4-inch nominal pipe size and smaller, and 200° F and colder.

3.7.3.4 Basis for Selection of Frequencies

The b sis for the selection of forcing frequencies is presented in the seismic qualification criteria. All frequencies in the range of 1 to 33 hertz are considered in the analysis and testing of the components and their supporting structures.

Three ranges of equipment/support behavior which affect the magnitude of the seismic acceleration are possible, as follows:

a. If the equipment is rigid relative to the structure, the maximum acceleration of the equipment mass approaches that of the structure at the point of equipment support. The equipment acceleration value in this case corresponds to the low period region of the floor response spectra.

b. If the equipment is very flexible relative to the structure, the internal distortion of the structure is unimportant, and the equipment behaves as though supported on the ground.

c. If the periods of the equipment and supporting structure are nearly equal, resonance occurs and must be taken into account.

Also, as noted in Subsection 3.7.3.2, rigid equipment/support systems have natural frequencies greater than 33 hertz.

3.7.3.5 Use of Equivalent Static Load Method of Analysis

Balance of Plant

No static load method is utilized in the seismic analyses of piping systems. However, in the seismic analyses of equipment, the equivalent static load method is used if the equipment is not rigid and a dynamic analysis is not performed.

If the FNP is known, the static seismic coefficient is equal to 1.5 times the g level corresponding to the equipment FNP in the applicable response spectrum curves (RSC). If the FNP is unknown, the static coefficient is equal to 1.5 times the peak g level in th applicable RSC.

equipment, and equipment supports, or because of rotations imposed upon the equipment by sources other than the piping". The effect of the differential motion is to impose a rotation on the component from the building. This motion, being a free end displacement and being similar to thermal expanion loads, will cause stresses which will be evaluated with ASME Code methods including the rules of NB-3227.5 used for stresses originating from restrained free end displacements.

The results of these two steps, the dynamic inertia analysis and the static differential motion analysis, are combined absolutely with due consideration for the ASME classification of the stresses.

3.7.3.10 Use of Constant Vertical Static Factors

In general, Seismic Category I subsystems are analyzed in the vertical direction using the methods specified in Subsection 3.7.3.1. No vertical static factors are used for subsystems.

3.7.3.11 Torsional Effects of Eccentric Masses

All concentrated loads in the piping system, such as valves and valve operators are modeled as massless members with the mass of the components lumped at its center of gravity. A rigid member is modeled connecting the center of gravity to the piping so that the torsional effects of the eccentric masses are considered.

3.7.3.12 Buried Seismic Category I Piping Systems and Tunnels

During an earthquake, buried structures such as piping and tunnels respond to various seismic waves propagating through the surrounding soil as well as to the dynamic differential movements of the buildings to which the structures are connected. The various waves associated with earthquake motion are P (compression) waves, S (shear) waves, and Rayleigh waves. The stresses in the buried structure are governed by the velocity and angle of incidence of these traveling waves. However, the wave types and their directions during earthquake are very complex. For design purposes, expressions for upper bound stresses as given in the published results of Newmark (Reference 10), Yeh (Reference 11), and Shah and Chu (Reference 12) were used.

Since all buried essential service water piping falls under subsection NC of ASME B&PV Code, Section III, the following stress limits are met:

Stresses due to sustained loads	$\leq 1.0\ S_h$
Stresses due to occasional loads (OBE)	$\leq 1.2\ S_h$
Stresses due to occasional loads (SSE)	$\leq 1.8\ S_h$

Stresses due to bending moments caused by soil settlement and/or overburden pressure $\leq 3.0\ S_c$

For all buried concrete electrical duct runs associated with the essential service water system, the design is in accordance with ACI-318-71 requirement.

For Byron, the essential service water pipeline was encased in concrete. This concrete encasement was designed to span the 50-foot-diameter design-basis sinkhole described in Subsection 2.5.4.10.4. The concrete encasement was designed such that the design-basis sinkhole could occur at any portion along the pipeline route.

radiation shielding and missile protection. The interior walls are of either concrete or concrete block construction. Fuel access is at grade level where a railroad track is provided to permit use of an overhead crane for handling the fuel.

3.8.4.1.3 Refueling Water Storage Tank and Tunnel

The refueling water storage tank is a reinforced concrete cylindrical structure supported on a mat foundation. The inside wall of the tank is lined with stainless steel liner. The tunnel which connects the refueling water storage tank with the auxiliary building is a reinforced concrete box section.

The tank and the tunnel are shown in Figure 1.2-11.

3.8.4.1.4 Main Steam and Auxiliary-Feedwater Tunnel

The main steam and auxiliary-feedwater tunnel is a bi-level reinforced concrete box section. It connects the containment with the turbine building through the auxiliary building. The top of the tunnel is 1 foot 0 inch below the grade level and it is shown in Figures 1.2-6 and 1.2-7.

The isolation valve room, known as the safety valve room, is a reinforced concrete structure which is an integral part of the main steam and auxiliary-feedwater tunnel at the containment building. It is designed using a two-way slab theory for all walls and slabs.

3.8.4.1.5 Electrical Duct Runs

Electrical Duct Runs are buried reinforced concrete conduits which carry Class 1E cables for safety-related equipment.

3.8.4.1.6 Essential Service Water Cooling Tower (Byron)

The essential service cooling tower consists of two four-cell concrete structures erected over one common reinforced concrete cold water basin. The mat foundation supporting structure rests on a grouted rock strata 9 feet 0 inch below grade level. The internal water distribution system and the fill are supported on concrete beam and column system with bracings to resist lateral loads.

The fan equipment including gear box are surrounded by 14 feet 0 inch high concrete recovery stack.

3.8.4.1.7 River Screen House (Byron)

The river screen house consists of reinforced concrete structure with main floor 3 feet 6 inches above grade, internal and external concrete walls, and concrete mat foundation. The roof and intermediate slab consists of steel framing with slab on metal deck. The superstructure consists of structural steel braced framework covered by insulated siding. Figure 1.2-16 shows the structural arrangement of the river screen house.

TABLE 3.8-7

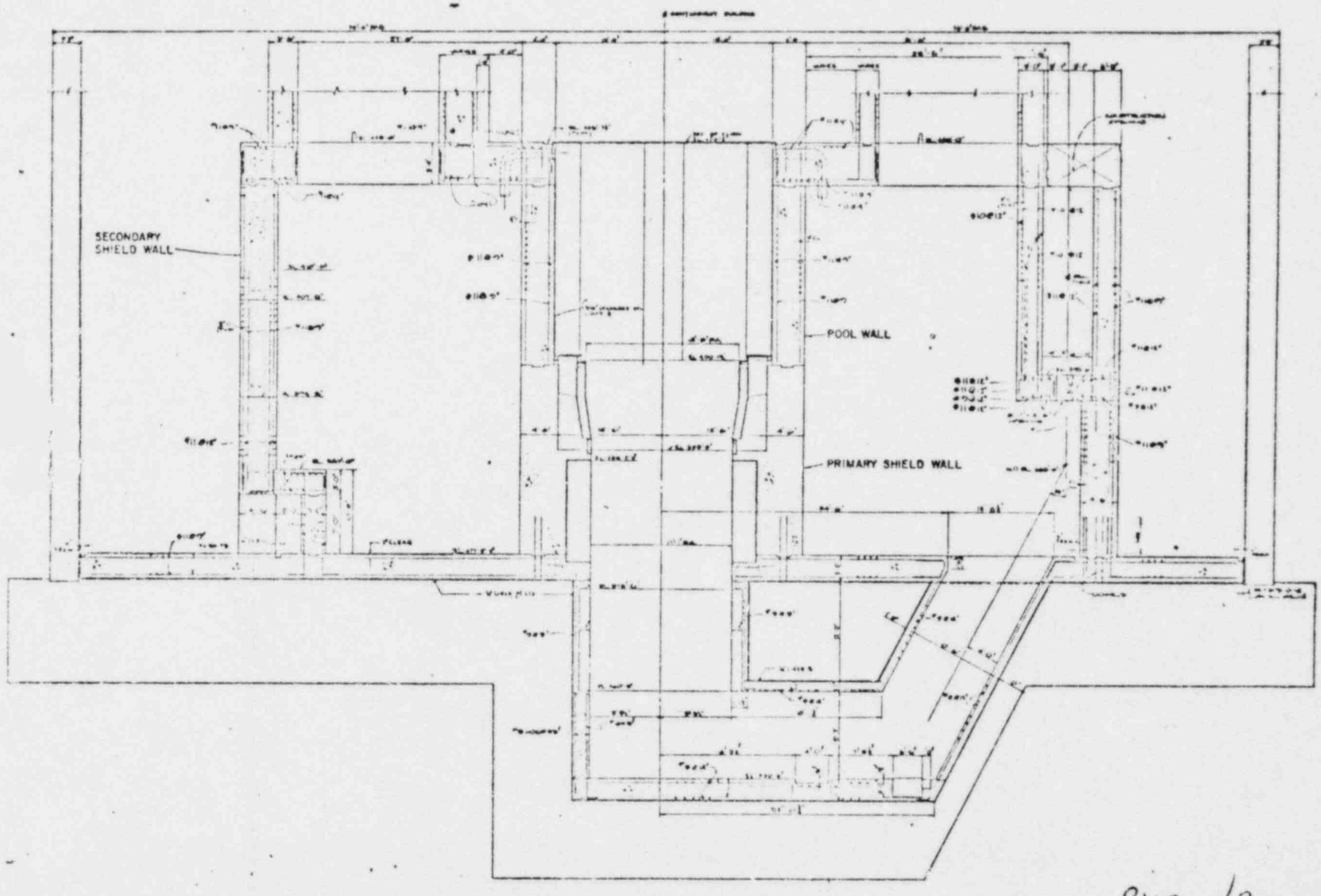
LOAD DEFINITIONS AND COMBINATIONS FOR CLASS MC CONTAINMENT COMPONENTS*

(For Definitions See Table 3.8-4)

LOADING CATEGORY	ITEM NUMBER	LOAD FACTORS													
									SEVERE ENVIRONMENTAL	ABNORMAL					EXTREME ENVIRONMENTAL
		D	L	R_o	T_o	P_o	P_e	P'	E	R_r	T_a	P_a	R_a	M	E'
Construction	1	1.0	1.0		1.0										
Test	2	1.0	1.0		1.0**			1.0							
Normal	3	1.0	1.0	1.0	1.0	1.0	1.0								
Severe Environmental	4	1.0	1.0	1.0	1.0	1.0	1.0		1.0						
Abnormal	5	1.0	1.0				1.0				1.0	1.0	1.0		
	6	1.0	1.0				1.0			1.0	1.0	1.0	1.0		
Extreme Environmental	7	1.0	1.0	1.0	1.0	1.0									1.0
Abnormal/Severe Environmental	8	1.0	1.0				1.0		1.0		1.0	1.0	1.0		
	9	1.0	1.0						1.0		1.0		1.0	1.0	
	10	1.0	1.0				1.0		1.0	1.0	1.0	1.0	1.0		
Abnormal/Extreme Environmental	11	1.0	1.0				1.0			1.0	1.0	1.0	1.0		1.0
	12	1.0	1.0				1.0			1.0	1.0	1.0	1.0		1.0

*Does not include process piping penetrations.

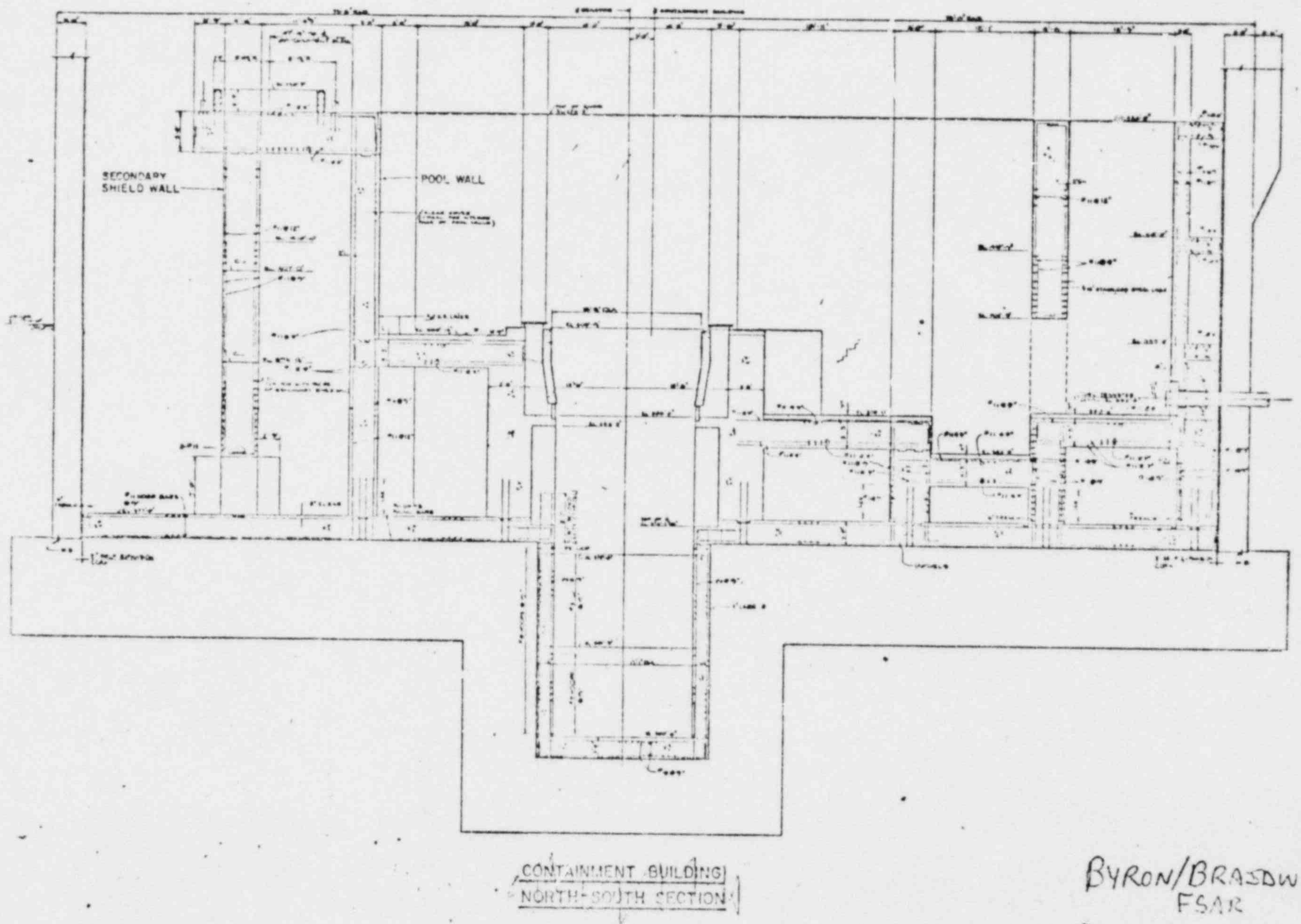
** Temperature at time of test.



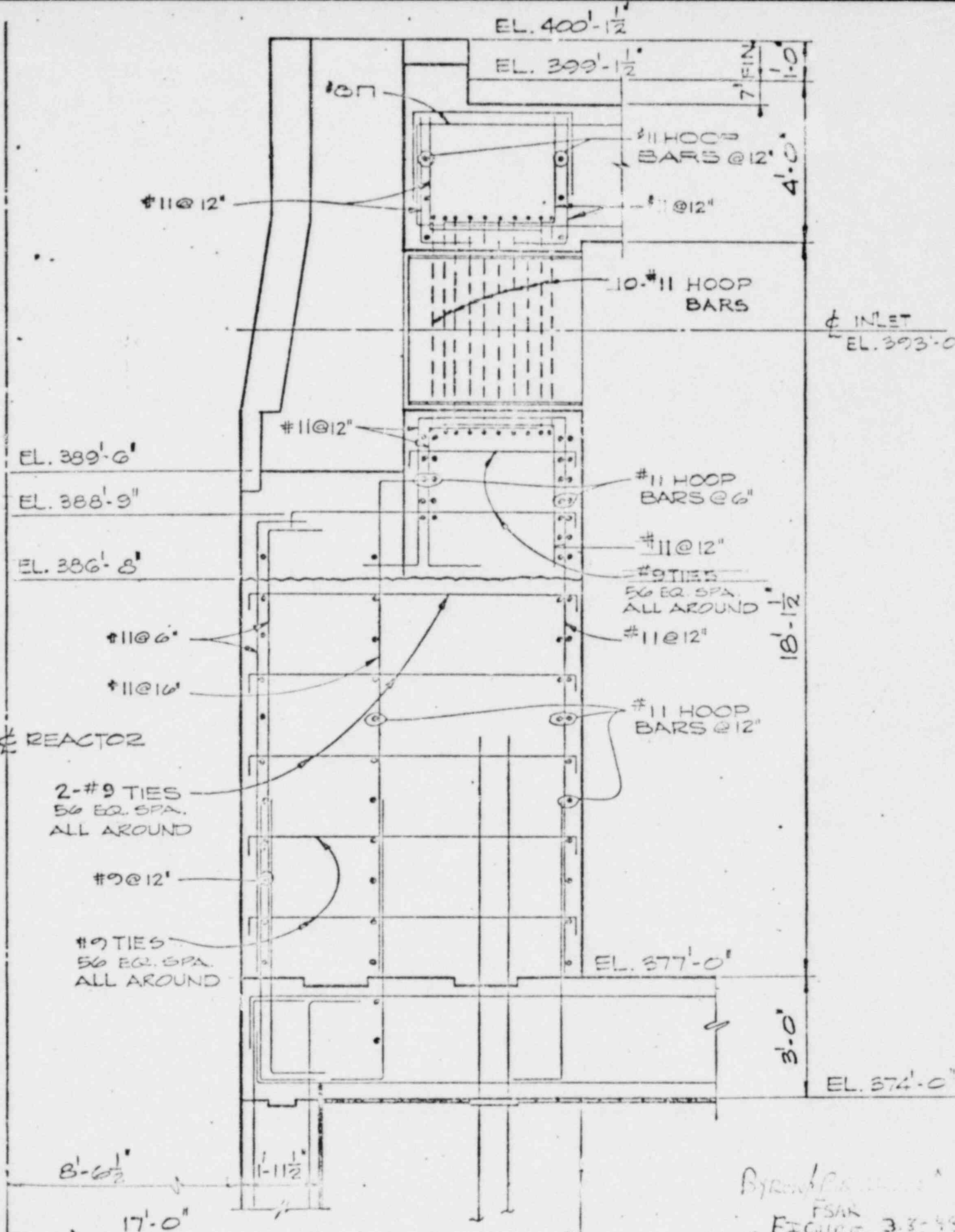
CONTAINMENT BUILDING
EAST-WEST SECTION

BYRON/BRAIDWOOD
FSAR

FIGURE 3.8-46
CONTAINMENT BUILDING
EAST-WEST SECTION

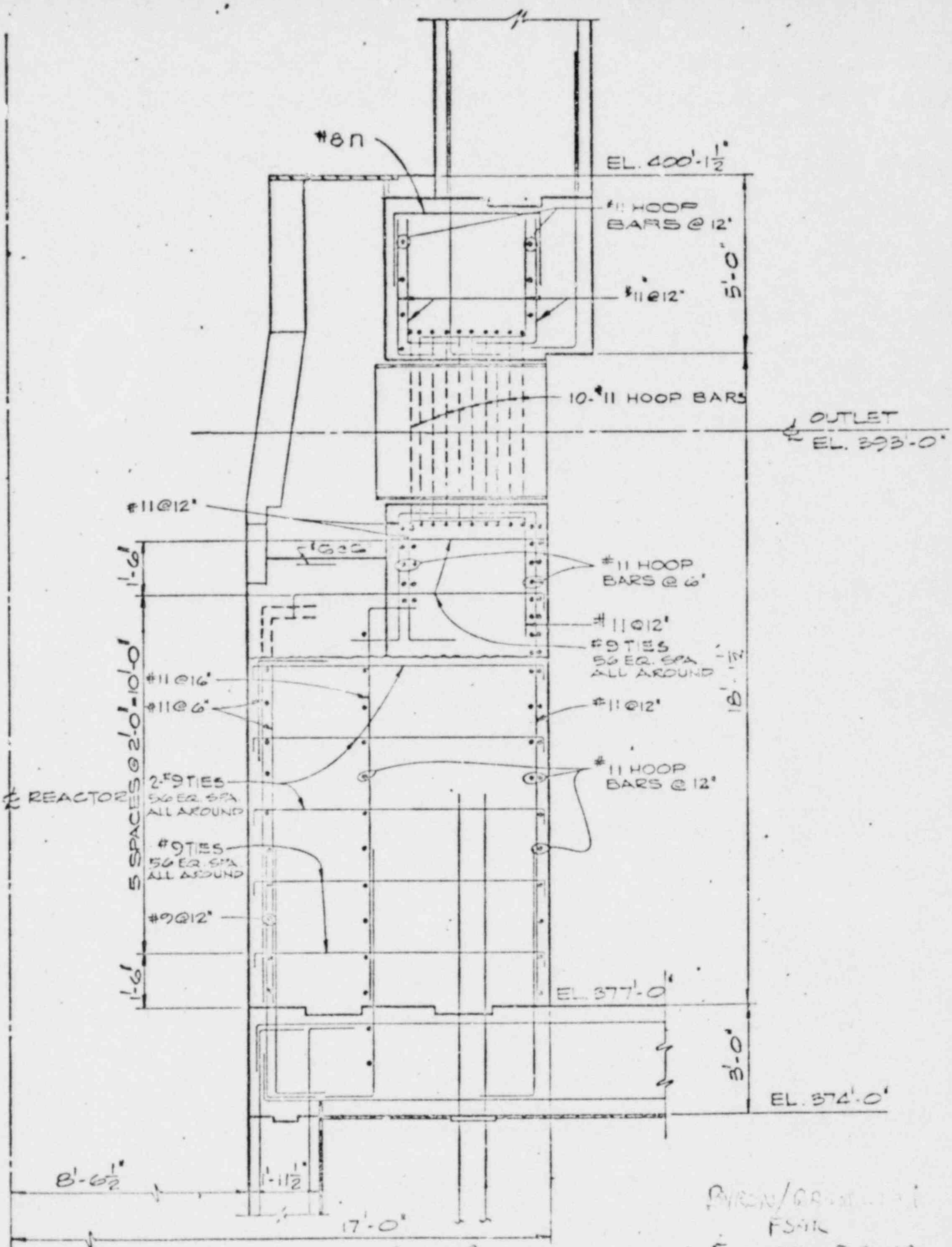


BYRON/BRAIDWOOD
FSAR
FIGURE 3.8-47
CONTAINMENT BUILDING
NORTH-SOUTH SECTION



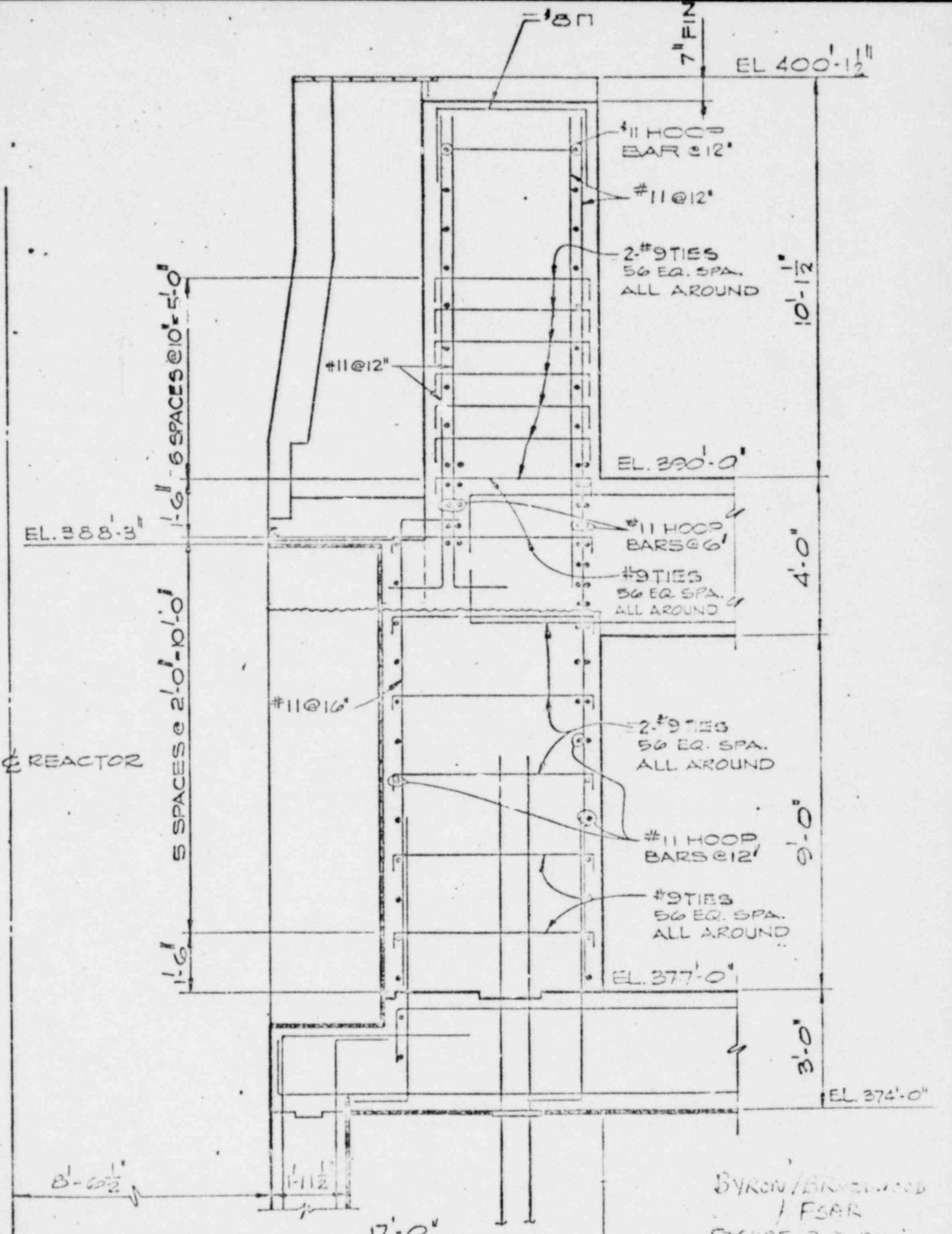
CONTAINMENT BUILDING
PRIMARY SHIELD WALL
AT INLET NOZZLE

Byron/
FSAR
FIGURE 3.8-48
CONTAINMENT BUILDING
PRIMARY SHIELD
WALL AT INLET
NOZZLE



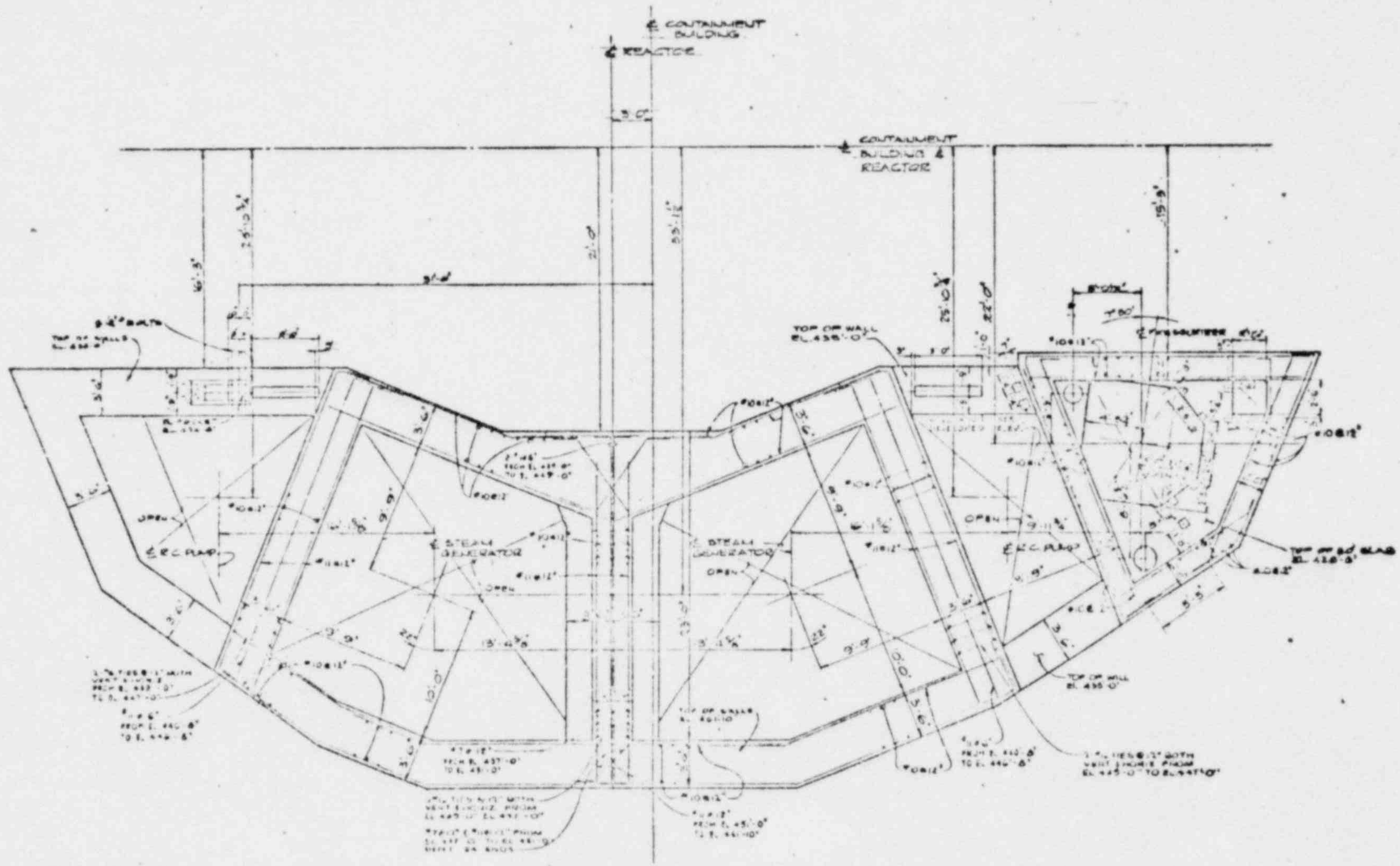
CONTAINMENT BUILDING
PRIMARY SHIELD WALL
AT OUTLET NOZZLE

BYRON/BRAIDWOOD
FSAR
FIGURE 3.8-49
CONTAINMENT BUILDING
PRIMARY SHIELD WALL
AT OUTLET NOZZLE



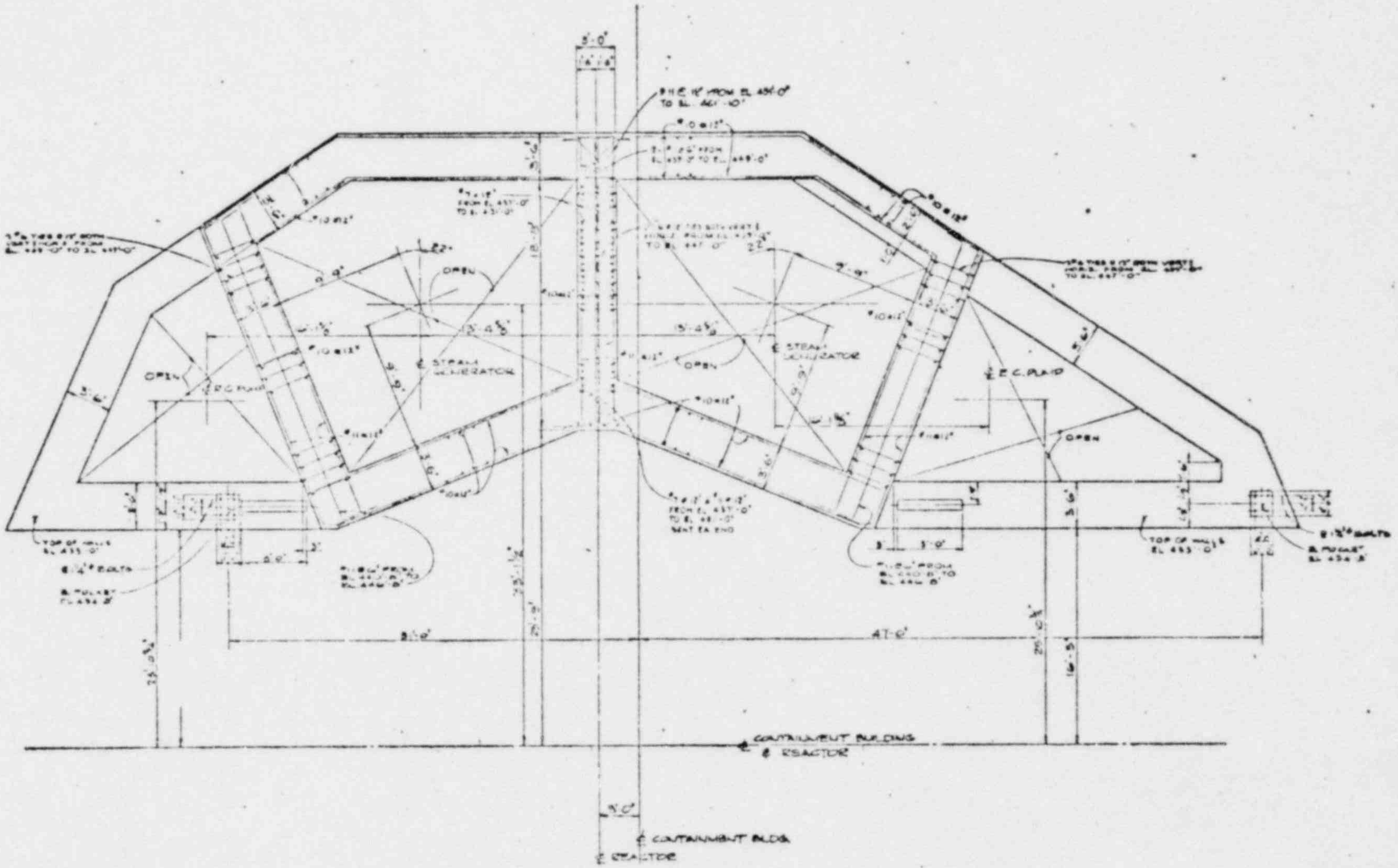
CONTAINMENT BUILDING
PRIMARY SHIELD WALL AT
NEUTRON DETECTOR CAVITY

BYRON/BRAIDWOOD
FSAR
FIGURE 3.8-50
CONTAINMENT BUILDING
PRIMARY SHIELD WALL
AT NEUTRON DETECTOR
CAVITY



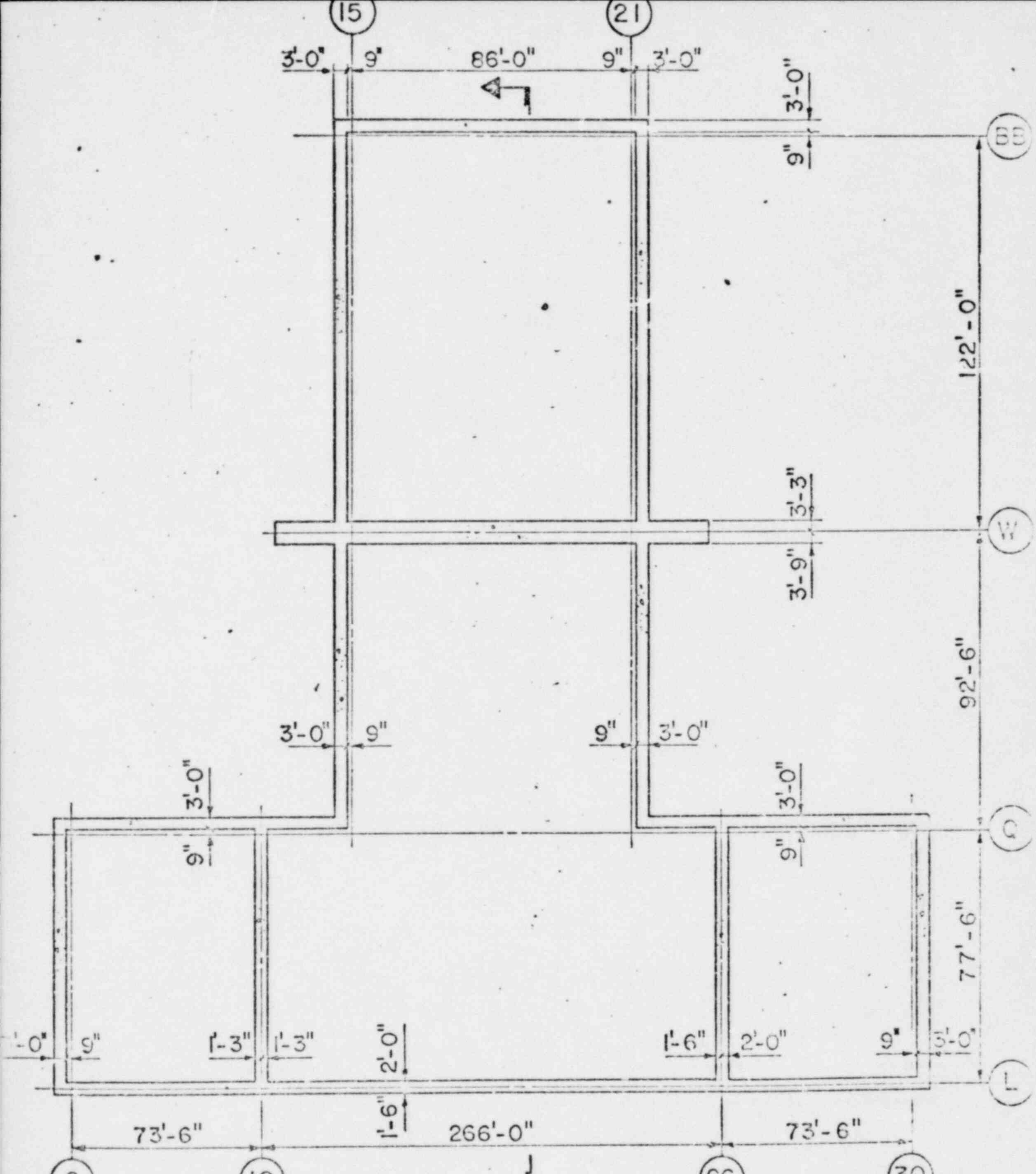
CONTAINMENT BUILDING
N.S.S.S. COMPONENT ENCLOSURES

BYRON / BRAIDWOOD
FSAR
FIGURE 3.8-51
CONTAINMENT BUILDING
N.S.S.S. COMPONENT
ENCLOSURES



CONTAINMENT BUILDING
N.S.S.S. COMPONENT ENCLOSURES

BYRON/ BRAIDWOOD
FSAR
FIGURE 3.8-51
CONTAINMENT BUILDING
NSSS COMPONENT ENCLOSURES
(SHEET 2 OF 2)



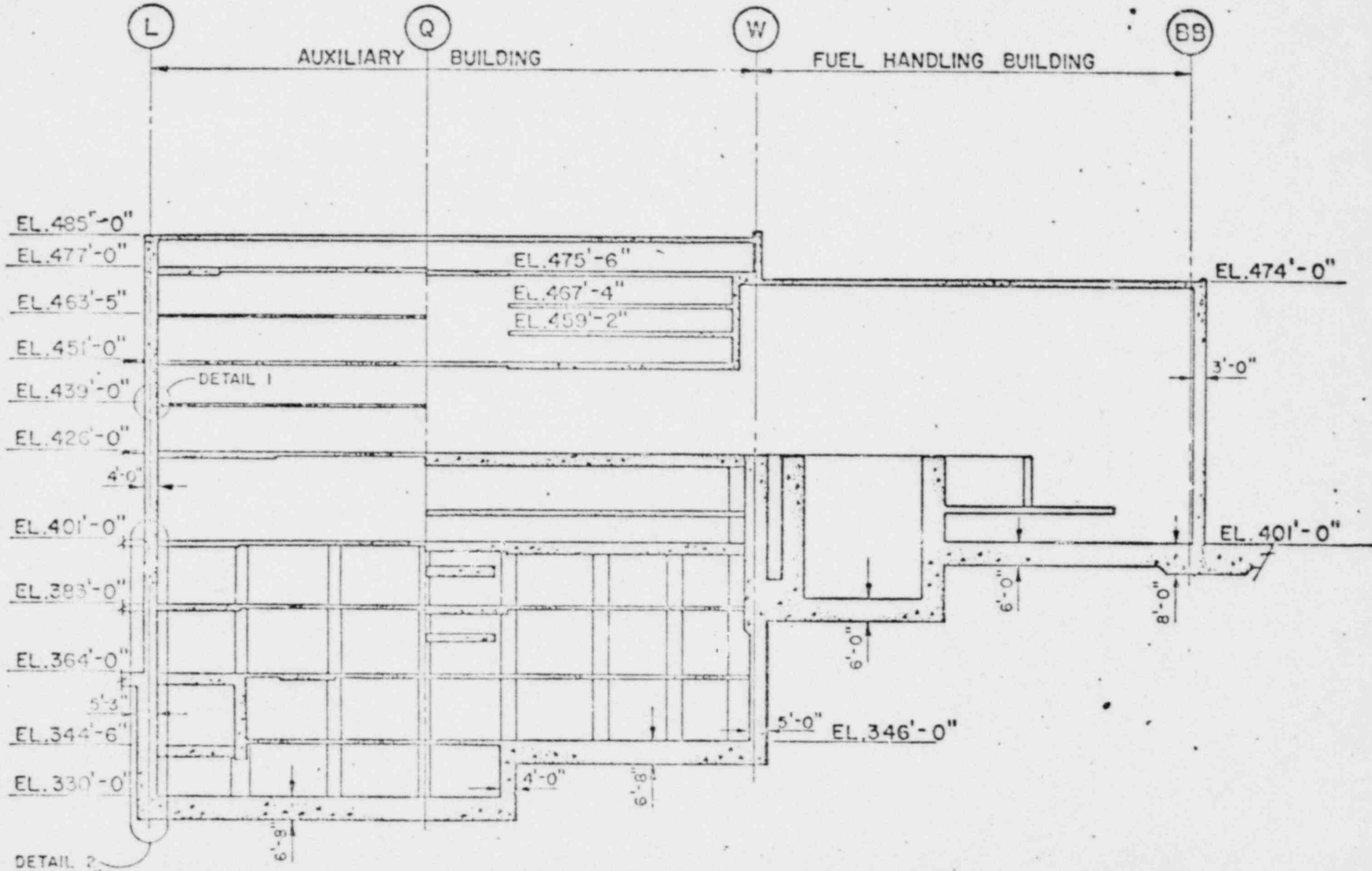
AUXILIARY - FUEL HANDLING BUILDING Complex

SHEAR WALLS AT ELEVATION 401'-0"

BYRON/BRAIDWOOD
FSAR
FIGURE 3.8-52

BYRON → N

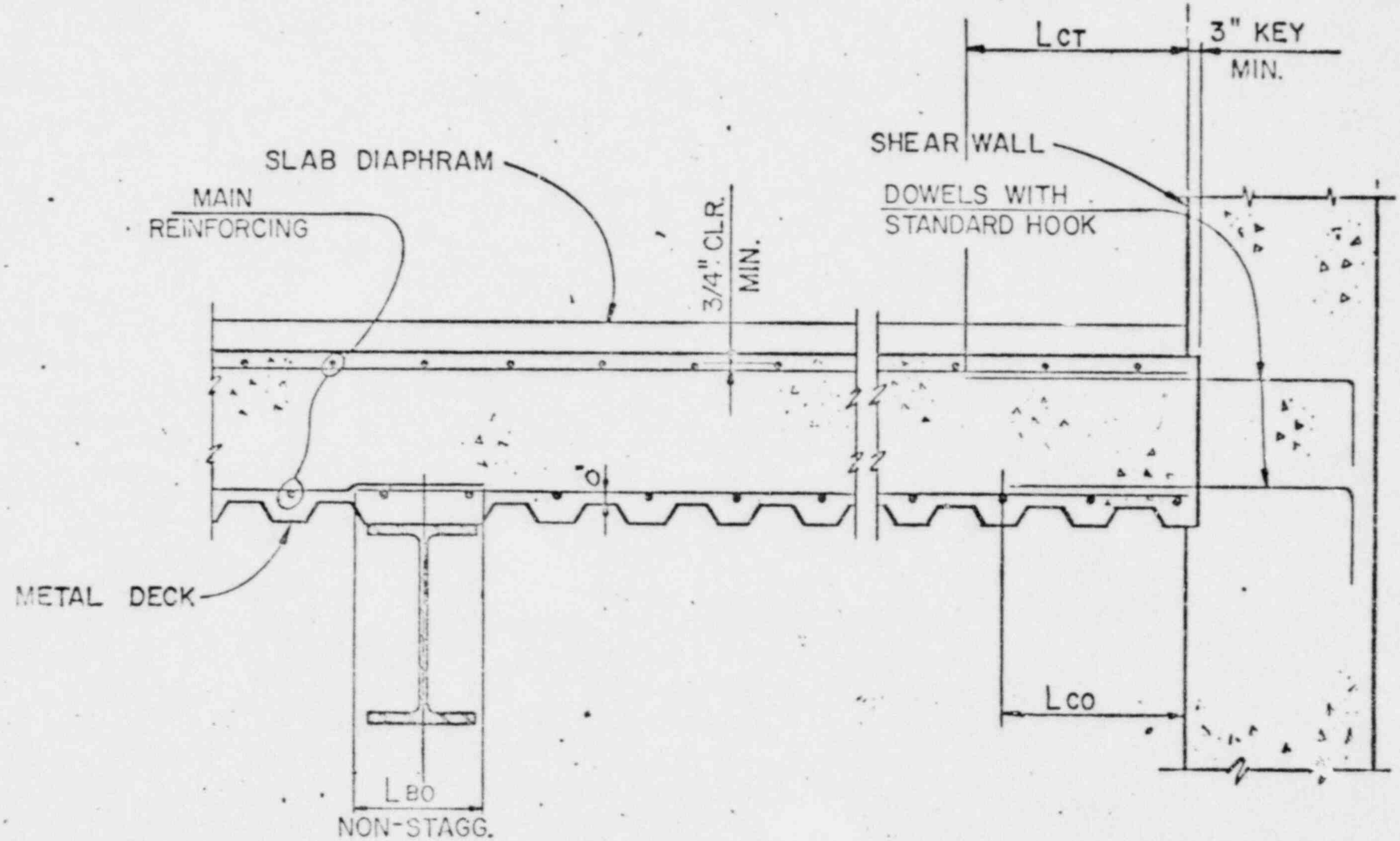
N ← BRAIDWOOD



AUXILIARY BUILDING SECTION

Byron/Braidwood
FSAR
Figure 3.8-53
Auxiliary Building
SECTION

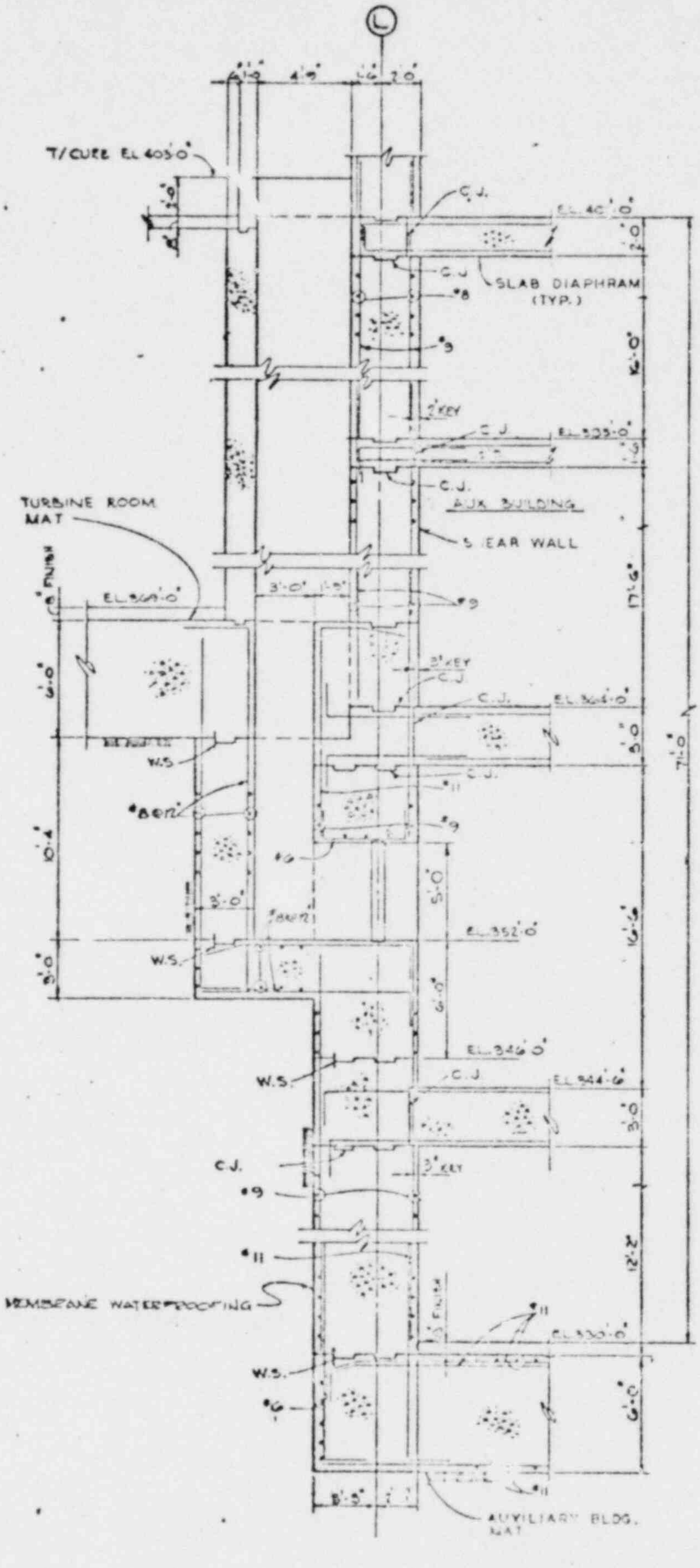
Figure 3.8-54



SHEAR WALL - SLAB DIAPHRAM ABOVE GR

DETAIL I

Byron/Braidwood
FSAR
FIGURE 3.8-57
SHEAR WALL - SLAB
DIAPHRAM ABOVE
GRADE DETAIL 1



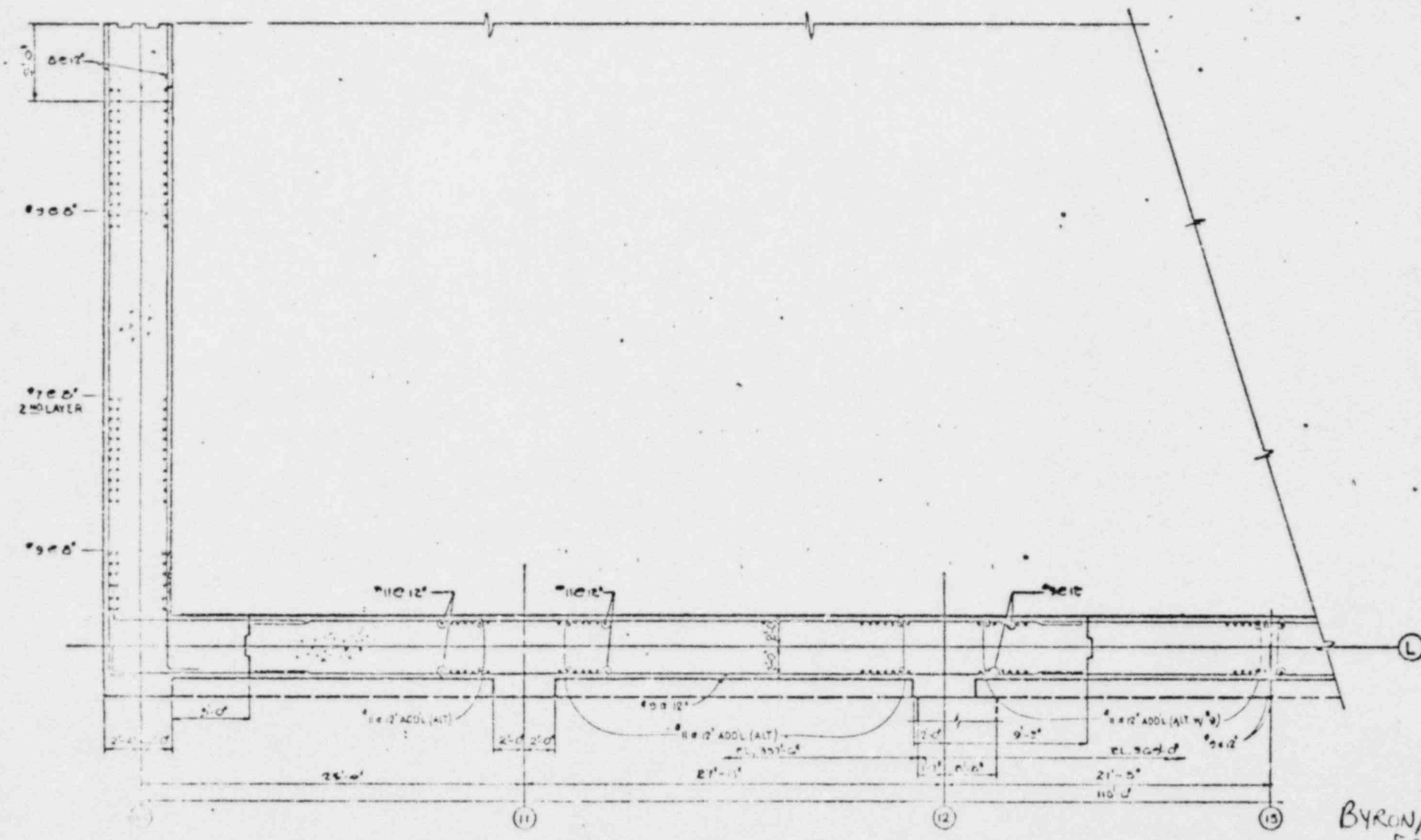
SHEAR WALL - SLAB DIAPHRAM BELOW

DETAIL 2

BYRON / BRAIDWOOD FSAR

Figure 3.8-55

SHEAR WALL - SLAB DIAPHRAM BELOW GRADE DETAIL 2



TYPICAL WALL CORNER REINFORCING PLAN

BYRON/BRAIDW
FSAR
FIGURE 3.8-5
TYPICAL WALL
CORNER REINFORC

d. Mass released to Containment during blowdown. (Figure 15.6-16)

e. Energy released to Containment during blowdown. (Figure 15.6-17)

f. Fluid quality in the hot assembly during blowdown. (Figure 15.6-18)

g. Mass velocity during blowdown. (Figure 15.6-20)

h. Accumulator water flow rate during blowdown. (Figures 15.6-19)

i. Pumped safety injection water flow rate during reflood. (Figures 15.6-21)

The maximum clad temperature calculated for a large break is 2102°F which is less than the Acceptance Criteria limit of 2200°F of 10CFR50.46. The maximum local metal-water reaction is 5.53% which is well below the embrittlement limit of 17% as required by 10CFR50.46. The total core metal-water reaction is less than 0.3% for all breaks, as compared with the 1% criterion of 10CFR50.46, and the clad temperature transient is terminated at a time when the core geometry is still amenable to cooling. As a result, the core temperature will continue to drop and the ability to remove decay heat generated in the fuel for an extended period of time will be provided.

Small Break Results

As noted previously, the calculated peak clad temperature resulting from a small break LOCA is less than that calculated for a large break. Based on the results of the LOCA sensitivity studies (Reference 21) the limiting small break was found to be less than a 10 inch diameter rupture of the RCS cold leg. Therefore, a range of small break analyses are presented which establishes the limiting break size. The results of these analyses are summarized in Tables 15.6-1 and 15.6-4.

Figures 15.6-34 through 15.6-47 present the principal parameters of interest for the small break ECCS analyses. For all cases analyzed the following transient parameters are presented:

a. RCS pressure. (Figure 15.6-34, 15.6-41, 15.6-42)

b. Core mixture height. (Figure 15.6-[illegible], 15.6-43, 15.6-44)

REGULATORY GUIDE 1.46

Revision 0, May 1973

PROTECTION AGAINST PIPE WHIP INSIDE CONTAINMENT

The applicant complies with Regulatory Guide 1.46. Further clarifications are provided in Subsections 3.6.2.3, 3.9.3.4.2, 3.8.3.2.1, 5.4.11.3, and 7.1.2.10 of the B/B-FSAR.

REGULATORY GUIDE 1.58

Revision 1, September 1980

QUALIFICATION OF NUCLEAR POWER PLANT INSPECTION, EXAMINATION, AND TESTING PERSONNEL

The applicant complies with the positions of this Regulatory Guide with the following exception:

> Regulatory Guide 1.58, Revision 1, Position 6 requires that a candidate for Level I, II, or III Inspector for inspection activities be a high school graduate or should have earned the General Education Development equivalence of a high school diploma. In the applicant's judgment, this is an unnecessary and unfair restriction. Personnel can be trained, evaluated, and tested to determine if they are qualified for a specific test or inspection function, regardless of whether or not they have a high school diploma or the equivalent. A person should be utilized based upon his capability to perform the job, and on his qualifications and abilities. In the applicant's view, continual on the job training with timely performance reviews is an acceptable method of complying with 10 CFR 50, regarding such qualification of plant inspection, examination, and testing personnel.

Also refer to the Commonwealth Edison Company Quality Assurance Program Topical Report CE-1-A.

REGULATORY GUIDE 1.67

Revision 0, October 1973

INSTALLATION OF OVERPRESSURE PROTECTION DEVICES

The Applicant complies with the regulatory position with the following comments and exceptions keyed to paragraph numbers in the Positions.

1. Safety/relief valve design loads and load characteristics are not included in the design specifications but are calculated as part of the analysis which is performed after issuance of the design specification. The design specification includes the requirement that such loads be included in analysis.

4. The guide states that either a dynamic analysis must be performed or a dynamic load factor (DLF) of 2.0 must be used. Code Case 1569 provides the analyst the option of performing a parametric study to determine the DLF. A discussion of the main steam relief valves determined that these valves do not utilize a DLF of 2.0. The design basis for utilizing a DLF other than 2.0 was based on a parametric study based on a dynamic analysis as allowed for in Code Case 1569. All other relief valves utilize a DLF of 2.0.

a. Further discussion of the reactor trip system and overpressurization is in Section 7.1 and Subsection 5.2.2, respectively.

(See Paragraph 5.4.11.3 for further information.)