DUANE ARNOLD ENERGY CENTER

RESPONSES TO NUREG 0578

As Clarified By NRC Letter
Dated October 30, 1979 To All
Operating Nuclear Power Plants

IOWA ELECTRIC LIGHT AND POWER COMPANY
JANUARY 1, 1980

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APPENDIXES

1. Emergency Power Supply Requirements for the Pressurizer Heaters, Power Operated Relief and Block Valves, and Pressurizer Level Indicators in PWRs - NUREG 0578, Section 2.1.1

NRC POSITION:

Consistent with satisfying the requirements of General Design Criteria 10, 14, 15, 17 and 20 of Appendix A to 10 CFR Part 50 for the event of loss of offsite power, the following positions shall be implemented:

Pressurizer Heater Power Supply

1. The pressurizer heater power supply design shall be capable of supplying, from either the offsite power source or the emergency power source (when offsite power is not available), a predetermined number of pressurizer heaters and associated controls necessary to establish and maintain natural circulation at hot standby conditions. The required heaters and their controls shall be connected to the emergency buses in a manner that will provide redundant power supply capability.

2. Procedures and training shall be established to make the operator aware of when and how the required pressurizer heaters shall be connected to the emergency buses. If required, the procedures shall identify under what conditions selected emergency loads can be shed from the emergency power source to provide sufficient capacity for the connection of the pressurizer heaters.

3. The time required to accomplish the connection of the preselected pressurizer heater to the emergency buses shall be consistent with the timely initiation and maintenance of natural circulation conditions.

4. Pressurizer heater motive and control power interfaces w'th the emergency buses shall be accomplished through devices that have been qualified in accordance with safety-grade requirements.

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Power Supply for Pressurizer Relief and Block Valves and Pressurizer Level Indicators

1. Motive and control components of the power-operated relief valves (PORVs) shall be capable of being supplied from either the offsite power source or the emergency power source when the offsite power is not available.

2. Motive and control components associated with the PORV block valves shall be capable of being supplied from either the offsite power source or the emergency power source when the offsite power is not available.

3. Motive and control power connections to the emergency buses for the PORVs and their associated block valves shall be through devices that have been qualified in accordance with safety-grade requirements.

4. The pressurizer level indication instrument channels shall be powered from the vital instrument buses. The buses shall have the capability of being supplied from either the offsite power source or the emergency power source when offsite power is not available.

RESPONSE:

Iowa Electric Light and Power Company concurs with the BWR Owner's Group position on this matter, Reference 3.

The Owner's Group has concluded that because natural circulation is a strong and inherent phenomenon in BWRs, independent of any powered system, there is no need for action in response to this matter.

This position was accepted provided that emergency power is available to provide a long-term source of air for air-operated relief valves, Reference 4.

At Duane Arnold Energy Center, the safety relief valves are nitrogen-operated. The solenoids controlling the nitrogen supply to the operating cylinders are powered from the 120-volt instrument ac bus which receives its power from the plant essential bus. On loss of power from one bus, the load can be manually transferred to the alternate essential bus. Sufficient nitrogen is available in the valve's individual accumulators to provide several safety relief valve cycles (only the four ADS valves are equipped with accumulators).

Modifications will need to be made to the nitrogen supply containment isolation valve controls to allow a long-term supply of nitrogen from the normal nitrogen system to be available for valve operation. Presently, the containment isolation valves close on an isolation signal and the valve logic does not allow reopening unless the isolation signal has cleared. Additionally, safety-grade power is not supplied to these valves.

As an interim measure, procedures have been developed to allow for modifying the isolation logic to the valves and connecting a safety-grade power source in order to allow opening of the valves. These procedures would only be implemented after an accident and only if continued operation of the valve was deemed necessary.

A permanent modification will be made to allow manual reopening of the nitrogen supply valves from the control room if considered necessary. This modification will be made after receipt of required material.

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2. Performance Testing for BWR and PWR Relief and Safety Valves - NUREG 0578 Section 2.1.2

NRC POSITION:

Pressurized Water Reactor and Boiling Water Reactor licensees and applicants shall conduct testing to qualify the reactor coolant system relief and safety valves under expected operating conditions for design basis transients and accidents.

RESPONSE:

Iowa Electric Light and Power Company concurs with the BWR Owners' Group response to Reference 4 concerning the performance testing of relief and safety valves.

The response (Ref 6) states that a testing program will be conducted for low-pressure single-phase liquid and two-phase flow conditions. A commitment on whether to conduct high-pressure tests or provide safety-grade high level trips to preclude ECCS or feedwater operation at high levels will be made by January 31, 1980.

3. Direct Indication of Power-Operated Relief Valve and Safety Valve Position for PWRs and BWRs NUREG 0578 Section 2.1.3.a

NRC POSITION:

Reactor System relief and safety valves shall be provided with a positive indication in the control room derived from a reliable valve position detection device or a reliable indication of flow in the discharge pipe.

RESPONSE:

Iowa Electric Light and Power Company will provide relief and safety valve indication as follows:

1. There will be three sensing devices for each safety and safety-relief valve.

2. All sensing devices will be inside the drywell.

3. All sensing devices will be environmentally qualified

4. The sensing devices will use Class 1E penetrations.

5. The backup method of determining valve position (thermocouples) is powered separately from the power for the sensing devices.

As described in Reference 5, until environmentally qualified pressure switches become available, Iowa Electric intends to install 18 pressure switches, Pressure Controls Inc. (PCI) Model #A17-N1, in the six safety-relief valve discharge lines.

For the safety valves, it is intended to install pitot-tube type pressure switches supplied by General Electric.

The set point of the pressure switches will be selected to provide the operator with an unambiguous indication of the position of the valves.

The valve positions for the safety and the safety-relief valves will be indicated and alarmed in the control room.

All permanent valve position indications will be safety grade and environmentally qualified. In addition, they will be powered from one of the station's 125-volt batteries. Backup indication is provided by currently installed thermocouples in the discharge of the safety and the safety-relief valves. Interpretation of high discharge temperatures is discussed in the plant's emergency procedures.

The initially installed SRV pressure switches will not be seismically qualified. Justification for operation during the interim is based upon the fact that the Duane Arnold Energy Center presently has temperature indicators installed which indicate relief valve opening and are alarmed in the control room. In addition, as a result of an evaluation of the effect of a stuck-open relief valve, procedures have been modified to require reactor cooldown if a stuck-open safety or safety-relief valve is experienced.

Installation of the SRV pressure switches, both the initial, nonqualified switches and the permanent, qualified switches, as well as the SRV switches, will be within 30 days of receipt of material. It is presently anticipated that the unqualified SRV switches will be installed during the refueling outage scheduled to commence February 9, 1980.

4. Instrumentation for Detection of Inadequate Core Cooling for PWRs and BWRs - NUREG 0578 Section 2.1.3b

NRC POSITION:

Licensees shall develop procedures to be used by the operator to recognize inadequate core cooling with currently available instrumentation. The licensee shall provide a description of the existing instrumentation for the operators to use to recognize these conditions. A detailed description of the analyses needed to form the basis for operator training and procedure development shall be provided pursuant to another short-term requirement, "Analysis of Off-Normal Conditions, Including Natural Circulation" (see Section 2.1.9 of NUREG-0578)

In addition, each PWR shall install a primary coolant saturation meter to provide on-line indication of coolant saturation condition. Operator instruction as to use of this meter shall include consideration that is not to be used exclusive of other related plant parameters.

Licensees shall provide a description of any additional instrumentation or controls (primary or backup) proposed for the plant to supplement those devices cited in the preceding section giving an unambiguous, easy-to-interpret indication of inadequate core cooling. A description of the functional design requirements for the system shall also be included. A description of the procedures to be used with the proposed equipment, the analysis used in developing these procedures, and a schedule for installing the equipment shall be provided.

RESPONSE:

Iowa Electric Light and Power Company concurs with the BWR Owner's Group position, Reference 3.

A generic report on the adequacy of existing instrumentation for inadequate core cooling is being prepared by General Electric and is expected to be submitted in January 1980.

Interim operating procedures have been developed for mitigation of accidents that have the potential for inadequate core cooling. General Electric has already issued operator guidelines for small break accidents, Reference 9. These guidelines have been implemented and incorporated in the operating procedures. As General Electric issues additional guidelines, these will be incorporated in the Iowa Electric Light and Power Company procedures.

The recommendations of General Electric SIL-299, regarding ECCS low level actuation setpoints, have been implemented at Duane Arnold Energy Center.

5. <u>Containment Isolation Provisions for PWRs and BWRs - NUREG 0578 Section 2.1.4</u>

GENERAL RESPONSE:

Iowa Electric Light and Power Company concurs with the BWR Owner's Group position on this matter, as provided in Reference 3, and with the NRC comments on this position provided in Reference 4.

NRC POSITION:

1) All containment isolation system designs shall comply with the recommendations of SRP 6.2.4; i.e., that there be diversity in the parameters sensed for the initiation of containment isolation.

RESPONSE:

1) The DAEC containment isolation system design complies with the recommendations of SRP 6.2.4, Section II.6. Diversity of the parameters sensed is shown in the following list of isolation signals.

SIGNAL CODE	SIGNAL TYPE
A	Reactor Vessel Low-Water Level
B	Reactor Vessel Low-Low Water Level
C	Main Steam Line High Radiation
F	Primary Containment High Pressure
G	Reactor Vessel Low-Low-Low Water Level
Z	Reactor Building Vent Exhaust High Radiation
E	RCIC/HPCI Steam Supply Valves Open Following Initiation of the System by Isolation Signal B

Table 2.1.4 provides a list of isolation signal codes received by each isolation valve.

NRC POSITION:

2a) All plants shall give careful reconsideration to the definition of essential and nonessential systems, shall identify each system determined to be essential, shall identify each system determined to be nonessential, and shall describe the basis for selection of each essential system.

RESPONSE:

2a) Reconsideration of the definition of essential and nonessential systems has resulted in the following criteria for lines penetrating containment:

If a fluid line does not have a post-accident function, the line is nonessential and requires isolation following an accident.

If a fluid line provides an engineering safety feature function or engineering safety feature-related system function, it is essential and the isolation valves in the lines may remain open or be opened following an accident.

Engineering judgment was used to apply this criterion to each line in light of the system requirements as interpreted from the FSAR and piping and instrumentation diagrams.

A list of isolation valves is provided in Table 2.1.4 which identifies those valves essential and nonessential lines. Each line can be identified by finding the valve number on the appropriate piping and instrument diagram. This diagram then shows the location of the subject valve in relation to the piping for the system.

NRC POSITION:

2b) (Based on identification of lines as essential and nonessential), all plants shall modify their containment isolation designs accordingly, and shall report the results of the reevaluation to the NRC.

RESPONSE:

2b) Modification of the DAEC containment isolation design is to be completed when materials are delivered. The design changes are included as part of the response to Positions 3,4 (Section 2.1.4). The results of the reevaluation are contained in Table 2.1.4 and in the design change discussion following.

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NRC POSITION:

3) All nonessential systems shall be automatically isolated by the containment isolation signal.

4) The design of control systems for automatic containment isolation valves shall be such that resetting the isolation signal will not result in the automatic reopening of containment isolation valves. Reopening of containment isolation valves shall require deliberate operator action.

RESPONSE:

3,4) Degree of compliance with the above positions is shown on Table 2.1.4.

Only nonessential valves were reviewed for compliance with Position 4.

Noncompliances with the criteria of Positions 3 and 4 are noted as follows:

a) Type A isolation valves as listed in the remarks column in Table 2.1.4 will return to the position they were in before introduction of the isolation signal following operator reset of the isolation trip system. These valves have control circuits which utilize maintained contact control switches in series with the isolation signal's contacts. This means that if the control switch is left in the "open" position and the operator resets the isolation signal, then the valve will automatically reopen without any additional operator action.

b) Several nonessential valves do not have diverse isolation signals.

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PROPOSED DESIGN CHANGES

Part 1

The present design which this solution addresses concerns Type A valves listed on Table 2.1.4. Each valve presently uses an isolation signal contact to override the normal control switch function. The control switches are maintained contact type. This means that when the isolation signal is reset, the valve will automatically move to the position designated by the control switch, i.e., the valve will open whenever the switch is left in the "open" position.

The required design goal for this change is to make the operator go to each valve which is to be opened and use the control switch to move the valve. This can be done by replacing the maintained contact control switch with a spring return to normal type and adding a seal-in coil. (See Figure 1 for the present design and the proposed design.) The seal-in relay will deenergize when the isolation signal closes the valve. This will open the seal-in contact and require operator action to reenergize the relay and thus open the valve.

Part 2

All valves in Table 2.1.4 which only list one isolation signal require an additional isolation signal added into the control circuit. This isolation signal must be different from the existing signal. The valves which require this are:

Valve No	Existing Isolation Signal	New Signal (Relay No)
1804A	B	F, Z (K23-3,4)
1804B	B	F, Z (K24-3,4)

The suggested additions are shown as spare contacts on design documents.

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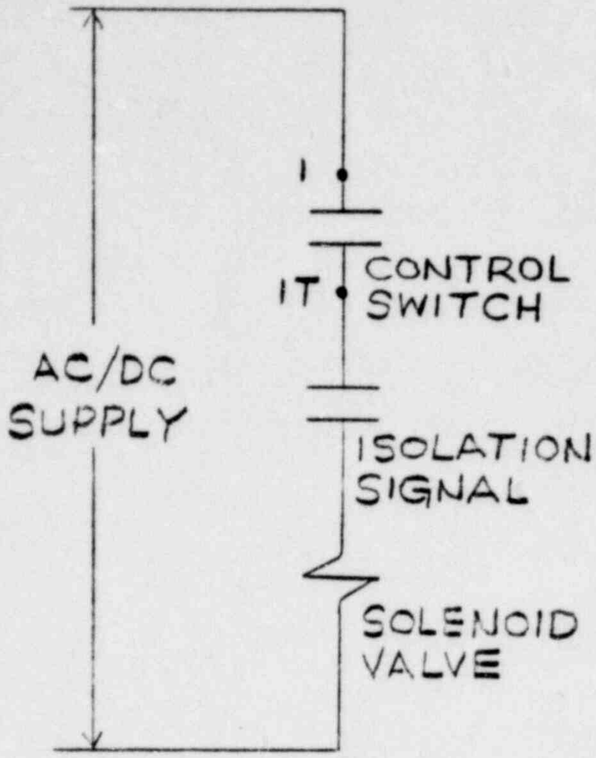
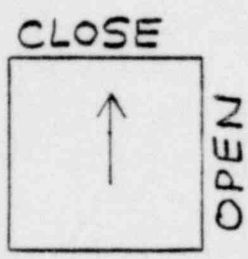
Part 3

Valves 5703A,B and 5719A,B have an open actuated position. The present design will be changed to show closed actuated positions for the subject valves. This will be done by removing power to these valves since control is not required during normal operation.

The above modifications will be implemented within 30 days of receipt of required material.

PRESENT DESIGN

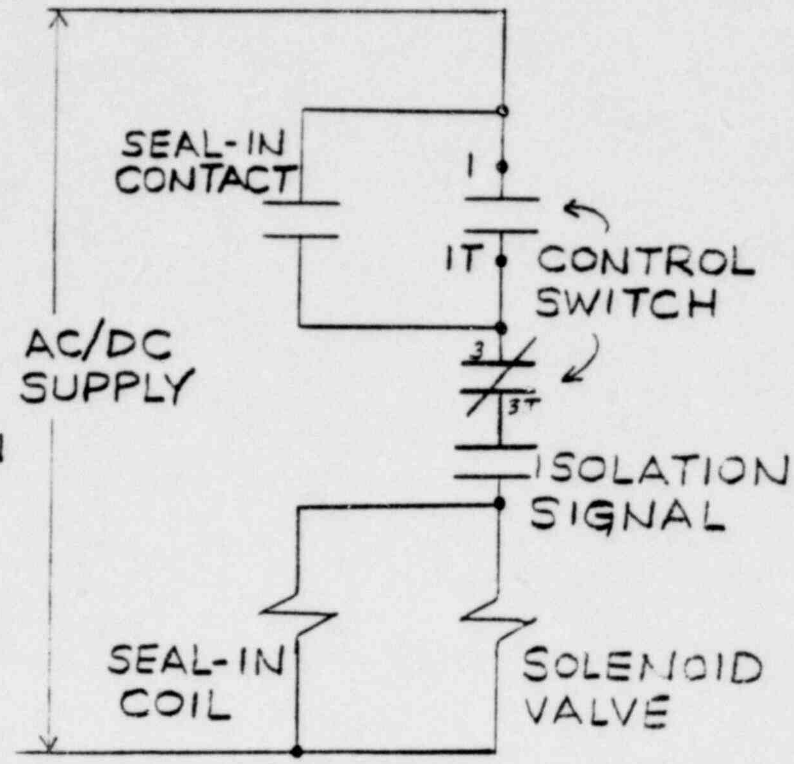


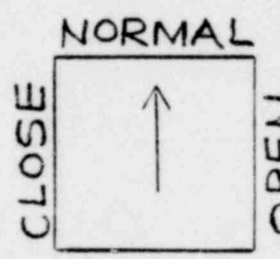


CONTACT	POSITION	
	OPEN	CLOSE
1T –||– 1	X	
2 –||– 2T	X	
3T –||– 3		X
4 –||– 4T		X

CONTROL SWITCH
WITH
MAINTAIN CONTACT

PROPOSED DESIGN





CONTACT	POSITION		
	OPEN	NORMAL	CLOSE
1T –||– 1	X		
2 –||– 2T	X		
3T –||– 3	X	X	
4 –||– 4T	X	X	

CONTROL SWITCH
WITH
SPRING RETURN TO NORMAL

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FIG. 1

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TABLE 2.1.4

CONTAINMENT ISOLATION

TABLE LEGEND

Essential/Nonessential

E - Essential
N - Nonessential

Isolation Signals

A - Reactor vessel low water level
B - Reactor vessel low-low water level
C - High radiation, main steam line
F - High drywell pressure
G - Reactor vessel low-low-low water level
Z - High radiation, reactor building and/or fuel pool ventilation exhaust
E - RCIC/HPCI steam supply valves open following initiation of the system by isolation signal 'B'

Valve Positions

O - Open
C - Closed
R - Returns to the valve's position before actuation

Remarks

A - Valve will open on disappearance of isolation signal and pressing of one of the reset pushbuttons (S32 or S33)
B - Valve will need individual operator action to reopen on disappearance of isolation signal

Item	Line Isolated	Valve No.	Essential/ Nonessential	Nonessential Valve Isolation Signals (see Note 1)	Nonessential Valve Positions: Actuated	Nonessential Valve Positions: Following Isolation Signal Reset	Electrical Schematic No.	APED Schematic No.	Exceptions and Remarks
1	Main steam line (A-D)	4412,15, 18,20	N	B,C	C	R	122/11	A71-3(10)	Type B
2	Main steam line (A-D)	4413,16, 19,21	N	B,C	C	R	122/11	A71-3(11)	Type B
3	Main steam line drain	4423	N	B,C	C	C	122/2	A71-3(8)	Type B
4	Main steam line drain	4424	N	B,C	C	C	122/5	A71-3(13)	Type B
5	Reactor water sample	4639	N	B,C	C	R	122/10	A71-3(9)	Type A
6	Reactor water sample	4640	N	B,C	C	R	122/10	A71-3(9)	Type A
7	Mini-purge	1804A,B	N	B	C	R	122/13	A71-3(14) A71-3(15)	Type A - Valve does not have diverse isolation signals.
8	Control rod drive (CRD) withdraw	1852	N	Note 7					This valve does not automatically isolate. See Note 7 for explanation.

TABLE [illegible]4 (continued)

Item	Line Isolated	Valve No.	Essential/ Nonessential	Nonessential Valve Isolation Signals (see Note 1)	Nonessential Valve Positions: Actuated	Nonessential Valve Positions: Following Isolation Signal Reset	Electrical Schematic No.	APED Schematic No.	Exceptions and Remarks
9	CRD withdraw	1854	N	Note 7					This valve does not automatically isolate. See Note 7 for explanation.
10	CRD insert	1851	N	Note 7					This valve does not automatically isolate. See Note 7 for explanation.
11	CRD insert	1853	N	Note 7					This valve does not automatically isolate. See Note 7 for explanation.
12	Scram inlet	1849	E						
13	Scram discharge	1850	E						
14	Residual heat removal (RHR) reactor shutdown cooling supply	1909	N	A,F	C	C	122/4	E11-7(2)	Type B
15	RHR reactor shutdown cooling supply	1908	N	A,F	C	C	122/2	E71-3(9)	Type B
16	RHR suppression pool suction	1989/2069	E						
17	RHR pump suction	1921,13 2012,15	E						
18	RHR discharge to suppression pool	1932/2005	E						
19	RHR to suppression spray	1933/2006	E						
20	RHR test line to suppression pool	1934/2007	N	A,F	C	C	121/59	E11-7(4,5)	Type B signals.
21	RHR containment spray	1902/2000	E						
22	RHR containment spray	1903/2001	E						
23	RHR reactor head spray	1900	N	A,F	C	C	122/2	E11-7(2)	Type B
24	RHR reactor head spray	1901	N	A,F	C	C	122/6	A71-3(13)	Type B

TABLE 2.1.[illegible] (continued)

Item	Line Isolated	Valve No.	Essential/ Nonessential	Nonessential Valve Isolation Signals (see Note 1)	Nonessential Valve Positions: Actuated	Nonessential Valve Positions: Following Isolation Signal Reset	Electrical Schematic No.	APED Schematic No.	Exceptions and Remarks
25	RHR low-pressure coolant injection (LPCI) to reactor	1905/2003	E						
26	RHR-LPCI to reactor	1904/2004	E						
27	RHR minimum pump flow	1935/2009	E						
28	RHR discharge to radwaste	1936	N	F,A	C	C	122/7		Type B
29	RHR discharge to radwaste	1937	N	F,A	C	C	122/15		Type B
30	RHR sample	1972/2051	N	F,A	C	R	122/13		Type A
31	RHR sample	1973/2052	N	F,A	C	R	122/13	A71-3(16)	Type A
32	Reactor water (RW) cleanup from reactor	2700	N	A	C	C	122/3	A71-3(13)	Type B - Valve does not have diverse isolation signals. See Note 8
33	RW cleanup from reactor	2701	N	A	C	C	122/5	A71-3(13)	Type B - Valve does not have diverse isolation signals. See Note 8
34	RW cleanup return	2740	N	A	C	C	122/14	A71-3(12)	Type B - Valve does not have diverse isolation signals. See Note 8
35	Reactor core isolation cooling (RCIC) to feedwater	2512	E						
36	RCIC turbine steam supply	2400	E						
37	RCIC turbine steam supply	2401	E						
38	RCIC pump suction (suppression pool)	2517	E						
39	RCIC pump suction (suppression pool)	2516	E						
40	RCIC minimum pump flow	2510	E						

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TABLE 2.1.4 (continued)

Item	Line Isolated	Valve No.	Essential/ Nonessential	Nonessential Valve Isolation Signals (see Note 1)	Nonessential Valve Positions: Actuated	Nonessential Valve Positions: Following Isolation Signal Reset	Electrical Schematic No.	APED Schematic No.	Exceptions and Remarks
41	Core spray to reactor	2115/2135	E						
42	Core spray to reactor	2117/2137	E						
43	Core spray test to suppression pool	2112	N	A,F	C	C	121/7	E21-6(2)	Type B
44	Core spray pump suction	2146/2147	E						
45	Core spray pump suction	2100/2120	E						
46	Core spray minimum pump flow	2104/2124	E						
47	Drywell equipment drain discharge	3728	N	A,F	C	R	122/9	A71-3(7)	Type A
48	Drywell equipment drain discharge	3729	N	A,F	C	R	122/9	A71-3(7)	Type A
49	Drywell floor drain discharge	3704	N	A,F	C	R	122/9	A71-3(7)	Type A
50	Drywell floor drain discharge	3705	N	A,F	C	R	122/9	A71-3(7)	Type A
51	High-pressure coolant injection (HPCI) to feed-water	2312	E						
52	HPCI turbine steam	2238	E						
53	HPCI turbine steam	2239	E						
54	HPCI pump suction (suppression pool)	2321	E						
55	HPCI pump suction (suppression pool)	2322	E						
56	HPCI/RCIC exhaust vacuum breaker	2290A,B	E						

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TABLE 2.1.4 (continued)

Item	Line Isolated	Valve No.	Essential/ Nonessential	Nonessential Valve Isolation Signals (see Note 1)	Nonessential Valve Positions: Actuated	Nonessential Valve Positions: Following Isolation Signal Reset	Electrical Schematic No.	APED Schematic No.	Exceptions and Remarks
57	HPCI minimum pump flow	2318	E						
58	Traveling incore probe (TIP) shear valves	--	N	Note 4					This valve does not automatically isola.-. See Note 4 for explanation.
59	TIP valves (see Note 3)	--	N	F,A	C	See remarks		A71-3(9)	Information was not available to evaluate the position of this valve following reset of the isolation signal. See Note 3
60	TIP purge valves	--	N	F,A	C	See remarks		A71-3(9)	Information was not available to evaluate valve position following reset of the isolation signal. See Note 3
61	Inst N_2 to drywell	4371A	N	A,F,Z	C	R	122/24	A71-3(14)	Type A
62	Inst N_2 to drywell	4371B	N	A,F,Z	C	R	122/24	A71-3(14)	Type A
63	Inst N_2 to drywell	4371C	N	A,F,Z	C	R	122/24	A71-3(14)	Type A
64	Containment N_2 component suction	4378A,B	N	A,F,Z	C	R	122/24	A71-3(14)	Type A
65	Reactor building cool water in	4841B	N	G	C	C	111/17		Type B - Valve does not have diverse isolation signals. See Note 2
66	Reactor building cool water out	4841A	N	G	C	C	111/18		Type B - Valve does not have diverse isolation signals. See Note 2
67	Well water in	5718A,B	N	F,G	C	R	113/94		Type A - This valve does not always isolate on signal "F". See Note 6.
68	Well water out	5704A,B	N	F,G	C	R	113/94		Type A - This valve does not always isolate on signal "F". See Note 6.
69	Well water back-flush inlet	5703A,B	N	See remarks	O	R	113/94		Type A - This valve opens when valves 5718A,B and 5704A,B close. No isolation provided.
70	Well water back-flush outlet	5719A,B	N	See remarks	O	R	113/94		Type A - This valve opens when valves 5718A,B and 5704A,B close. No isolation provided.

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TABLE 2.1.4 (continued)

Item	Line Isolated	Valve No.	Essential/ Nonessential	Nonessential Valve Isolation Signals (see Note 1)	Nonessential Valve Positions: Actuated	Nonessential Valve Positions: Following Isolation Signal Reset	Electrical Schematic No.	APED Schematic No.	Exceptions and Remarks
71	Vacuum breaker reactor building - Torus	4304,05	N	F,A,Z	C	R	122/23	A71-3(14)	Type A - This valve has an override signal which can open it when requested. See Note 5.
72	Purge inlet	4306	N	F,A,Z	C	R	122/13	A71-3(15)	Type A
73	Drywell purge inlet	4307	N	F,A,Z	C	R	122/12	A71-3(14)	Type A
74	Torus purge inlet	4308	N	F,A,Z	C	R	122/12	A71-3(14)	Type A
75	Drywell vent	4302	N	F,A,Z	C	R	122/12	A71-3(14)	Type A
76	Drywell vent valve bypass	4310	N	F,A,Z	C	R	122/12	A71-3(14)	Type A
77	Drywell vent	4303	N	F,A,Z	C	R	122/12	A71-3(15)	Type A
78	Torus vent	4300	N	F,A,Z	C	R	122/12	A71-3(14)	Type A
79	Torus vent valve bypass	4309	N	F,A,Z	C	R	122/12	A71-3(14)	Type A
80	Torus vent	4301	N	F,A,Z	C	R	122/13	A71-3(15)	Type A
81	Drywell atmospheric analyzer suction	8101A,B 8102A,B 8103A,B 8104A,B	E						
82	Makeup N_2	4311	N	F,A,Z	C	R	122/13	E71-3(15)	Type A
83	Makeup N_2 drywell	4312	N	F,A,Z	C	R	122/13	E71-3(14)	Type A
84	Makeup N_2 Torus	4313	N	F,A,Z	C	R	122/13	E71-3(14)	Type A
85	Drywell atmospheric analyzer return	8105A,B 8106A,B	E						
86	Torus atmospheric analyzer suction	8107A,B 8108A,B	E						
87	Torus atmospheric analyzer return	8109A,B 8110A,B	E						
88	CAD system isolation	4333A,B/ 4334A,B	N	Keylocked Closed			E122/33,35		

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TABLE 2.1.[illegible] (continued)

Item	Line Isolated	Valve No.	Essential/ Nonessential	Nonessential Valve Isolation Signals (see Note 1)	Nonessential Valve Positions: Actuated	Nonessential Valve Positions: Following Isolation Signal Reset	Electrical Schematic No.	APED Schematic No.	Exceptions and Remarks
89	CAD system isolation	4331A,B/ 4332A,B	N	Keylocked	Closed		E122/34,35		

NOTES:

1) These columns have been completed for nonessential valves only. For additional information, see FSAR Table 7.3-1.

2) These lines are not open to either the reactor or the drywell. In the event of a LOCA, failure of the RB cooling system would have to occur in addition to failure of the isolation signal "G" to allow any communication between the reactor or the drywell and the outside of the containment. Therefore, no additional automatic isolation is required.

3) Signal "A" or "F" causes automatic withdrawal of TIP probe. When probe is withdrawn, the ball valve automatically closes when the detector is housed in the changer shield. Isolation reset position was identified. However, the TIP is a closed system with no direct interface with reactor containment atmosphere. No additional isolation is required.

4) Remote-manual control switch is located in the control room. Should the probe fail to automatically withdraw, and a failure of the TIP tubing, the operator can isolate this line using the shear valve. Therefore, no additional automatic isolation is required.

5) Torus differential pressure high can open this valve with the isolation signals present. However, isolation is still achieved by upstream check valves which have position indication in the control room.

6) Signal "F" will affect the operation of these valves when the control switch is in the 'BW' position. However, automatic isolation requirements are adequate because these lines are not open to either the reactor or the drywell.

7) These lines can be closed from remote-manual controls located in the control room. In addition, these lines are part of a closed system which itself would have to be breached for a breach of containment to occur. No automatic isolation required.

8) The RWCU system intentionally remains active to keep cleansing the vessel water during the situation where high drywell pressure exists or high RB/fuel pool vent radiation exists. It is desirable to keep the RWCU operating under these conditions as long as the reactor vessel level is not low.

6. Dedicated H_2 Control Penetrations - NUREG 0578 Section 2.1.5.a

NRC POSITION:

Plants using external recombiners or purge systems for post-accident combustible gas control of the containment atmosphere should provide containment isolation systems for external recombiner or purge systems that are dedicated to that service only, that the redundancy and single failure requirements of General Design Criterion 54 and 56 of Appendix A to 10 CFR 50, and that are sized to satisfy the flow requirements of the recombiner or purge system.

RESPONSE:

Duane Arnold Energy Center utilizes a containment air dilution (CAD) system for control of combustible gases post-LOCA, as described in the DAEC FSAR, Appendix G, Section G.7. The CAD system is safety-grade, redundant, and designed so that failure of a single active component will not prevent the system from performing its safety function.

A purge capability also exists, but only as a backup to the CAD system. For this reason, the provisions of the NUREG position are not considered to apply to the Duane Arnold Energy Center design.

(Section 2.1.5.b, Inerting BWR Containments, is not applicable to DAEC. Implementation for applicable BWRs has been deferred.)

7. Capability to Install Hydrogen Recombiner at Each Light Water Nuclear Power Plant - NUREG 0578 Section 2.1.5c

NRC POSITION:

The procedures and bases upon which the recombiners would be used on all plants should be the subject of a review by the licensees in considering shielding requirements and personnel exposure limitations as demonstrated to be necessary in the case of TMI-2.

RESPONSE:

According to the clarification provided (Ref 2) this item does not apply to the Duane Arnold Energy Center.

8. Integrity of Systems Outside Containment Likely to Contain Radioactive Materials for PWRs and BWRs - NUREG 0578 Section 2.1.6.a

NRC POSITION:

Applicants and licensees shall immediately implement a program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as-low-as-practical levels. This program shall include the following:

a. Immediate Leak Reduction

1) Implement all practical leak reduction measures for all systems that could carry radioactive fluid outside of containment.

2) Measure actual leakage rates with system in operation and report them to the NRC.

b. Continuing Leak Reduction

Establish and implement a program of preventive maintenance to reduce leakage to as-low-as-practical levels. This program shall include periodic integrated leak tests at intervals not to exceed each refueling cycle.

RESPONSE:

Iowa Electric Light and Power Company concurs with the BWR Owner's Group position on this matter, as provided in Reference 7.

Iowa Electric Light and Power Company has developed an immediate leak reduction program. This program includes the following:

1. PURPOSE

The program is designed to provide an assessment of the leakage rates for the systems outside the containment at Duane Arnold Energy Center which could contain highly radioactive fluids during a serious transient. This includes a walkdown for visual inspection and identification of leakage sources for these systems, and where possible, immediate correction of these leakages.

2. PROGRAM DESCRIPTION

The immediate leakage reduction program is being accomplished as follows:

Systems subject to post-accident leakage have been selected (Section 3).

Leakage test procedures have been developed (Section 6).

Testing hardware and personnel are being provided.

Testing is being conducted and leakage test data are being collected.

Collected leakage test data is being evaluated and reviewed.

Corrective actions required for leakage reduction are being provided.

Corrective actions and retest as appropriate are being provided.

The preventive maintenance program for the plant to reduce the leakage to as-low-as-practical levels is being provided.

3. SYSTEM SELECTION

Systems outside the primary containment that could contain reactor water, reactor steam, containment atmosphere, or suppression pool water after a postulated accident resulting in gross fuel failure have been selected for this program.

The following systems have been selected:

- RHR System
- Core Spray System
- RCIC System
- HPIC System
- RWCU System (only to second isolation valve)
- CRD System (scram discharge headers only)
- Liquid Radwaste System (up to radwaste and floor drain collection tank)
- Containment Atmospheric Monitoring System
- Post-Accident Sampling System

4. BASIS FOR SELECTION OF SYSTEMS

The RHR system may be used in all its operating modes (including steam condensing) following a postulated major accident. It will carry reactor coolant, reactor steam and suppression pool water. The core spray system, carrying suppression pool water, is also likely to be required to operate.

Although it is likely that the reactor will be rapidly depressurized following a serious accident, it is possible that the RCIC and HPCI systems will operate early in the accident. The RCIC system is required if the RHR system is operated in the steam condensing mode.

The RWCU system will be automatically isolated as soon as the accident is detected. This system has been included only to the second isolation valves. Radiation levels dictate that this system be inspected during an outage in order to minimize personnel exposure to levels as low as reasonably achievable.

The scram discharge headers will not contain "post-accident" reactor water following the postulated accident. However, should the reactor remain pressurized for an extended period, the scram discharge headers will collect reactor water leaking into the CRD system.

The liquid radwaste system is not expected to process waste. It is likely, however, that some time during the accident recovery it will be necessary to pump highly contaminated fluids to the radwaste and floor drain collector tanks in the HPCI, RCIC building. The liquid radwaste system has been included only to these tanks.

The containment atmospheric monitoring system will circulate post-accident containment atmosphere at about the containment pressure through a 1-inch piping system while monitoring the hydrogen and oxygen content of the drywell and the suppression pool. Therefore, this system has been included.

The post-accident reactor coolant sampling system will be utilized to obtain samples of the potentially highly radioactive coolant following an accident. Upon installation of the sampling system with this capability, it will be included in the leak reduction program.

5. SYSTEMS EXCLUDED FROM THE LEAK REDUCTION PROGRAM

Systems which are isolated at the containment following postulated accidents are excluded. All containment isolation valves are being tested in accordance with the requirements of 10 CFR 50, Appendix J.

It is unlikely that the containment will be purged via the Standby Gas Treatment System (SGTS) following a postulated accident. Therefore, the SGTS is excluded since it is located inside the reactor building and contains reactor building atmosphere.

The Gaseous Radwaste System (offgas) is excluded since the steam jet air ejectors, steam packing exhausters, and mechanical vacuum pumps will not operate post-accident.

The MSIV leakage control system is excluded because it is maintained at a negative pressure while it is operating and any input to it comes from the MSIVs which are included in the above Appendix J test program.

The RWCU system beyond the second isolation valve will be isolated by the accident. It is unlikely that the filter demineralizer units will be useful for the accident recovery.

6. TEST PROCEDURE FOR THE IMMEDIATE LEAKAGE REDUCTION PROGRAM

6.1 GENERAL

As a result of USNRC's NUREG 0578, it has become necessary to implement a leakage measurement program at Duane Arnold Energy Center.

The intent of the leakage measurement program is to provide assurance of integrity for systems outside the primary containment that could contain highly radioactive fluids after an accident.

6.2 PURPOSE

The purpose of the leakage measurement program is to obtain a quantitative measure of leakage within selected systems by identifying and measuring the leakage from each component or potential source of leakage in the system. After leaking components have been repaired and reinspected, total system leakage can be determined. Total leakage from all systems will allow the estimation of airborne radioactivity. Leakage measurements from individual components will provide an input to the preventive maintenance program.

6.3 SCOPE

This program shall include the systems and portions of systems listed in Section 3. The inspection and subsequent actions are concentrated on areas where leakage is likely to occur. This includes valve stems, vents and drains, pump seals, pump seal leak-offs, pump case joints, valve body-to-bonnet joints, flanged pipe joints, body drain plugs, and relief valve discharges.

Only visually detectable leakage will be reported. Insulation will not be removed for this inspection.

6.4 TEST METHODS

The water and steam containing systems shall be visually inspected for leakage while the system operating conditions are duplicated.

The preferred method of testing water containing systems is in the test mode with recirculation to the source of water. Pump seal leakage shall be measured with the pump operating. In most cases, the system lineup provides post-accident flows and pressures.

In some cases, it is not practical to inspect systems operating in the test mode. In such cases, the system will be inspected under static conditions. Portions of the system which operate at high pressure shall be pressurized, with a hydrostatic pressure test pump, to the operating pressure. (Note that piping between pump discharges and the first block valve may not be pressurized.)

The steam side of the RCIC and HPCI system will be tested with nuclear steam, unless this is precluded by high radiation levels. In that case, auxiliary steam shall be used during shutdown.

The containment atmospheric monitoring system will be tested with helium gas and a helium sniffer.

Detailed test procedures for water pressurized systems and for steam pressurized systems have been implemented.

6.5 CORRECTIVE ACTION

The results of this inspection shall be noted for evaluation and corrective action where required. It should be noted that in some cases the reported leakage will be as low as practical (e.g. pump seal leak-offs). In cases where corrective action is performed, the leakage will be measured and recorded after the corrective action is complete.

Cases where leakage is observed, but is too low to be quantified or corrected, will be monitored during the preventive maintenance program.

6.6 DOCUMENTATION

Attached to each inspection procedure is a form listing each component to be inspected for leakage. The inspection of each system component shall be documented on this form.

The inspector shall note the system name, the operating mode of the system (e.g. test mode, standby mode, hydrostatic pressure test), and the area where the inspection was made (e.g. torus room or pump room).

In all cases where leakage is detected, the inspector shall note the source of the leakage (e.g. body drain valve). He shall note the operating mode or condition of valves or pumps (i.e. open or closed for valves, or running or off for pumps). Where a leak rate is measurable, it shall be noted in terms of cubic centimeters per minute or drops per minute. Where unquantified leakage is detected (e.g. damp packing or rust or water stains), the leak rate shall be shown as less than 5 drops per minute and the condition shall be described in the remarks column of the form. Where leakage is measured at a remote location (e.g. piped-up leak-offs), this location shall also be noted in the remarks column. The inspector shall sign and date each form.

When all forms for a particular system have been completed, they shall be sequentially numbered in the space provided.

System inspections completed prior to receipt of the attached forms may be documented by retention of the marked-up P&IDs and forms used to conduct that inspection. This should be noted on the attached forms.

The report forms shall be retained for future use and documentation of the work.

If the inspection reveals additional components not listed on the forms, such components shall be added to the form by the inspector.

7. PREVENTIVE MAINTENANCE PROGRAM

The data collected during this inspection shall be evaluated for acceptability and capability for further leak reduction.

Additional walkdown inspections (without the component check lists attached to the system procedures) shall be performed quarterly. This may be done in conjunction with the system operation test if the area is accessible. A general inspection for visible leakage shall be performed.

Once per year, or no less frequently than once per refueling cycle, a detailed walkdown inspection with the component check lists shall be made. It is recommended that the inspection be performed in conjunction with the system pressure test required by Section XI of the ASME B&PV Code (Article IWA5000).

In both of the above cases, special attention shall be given to leakages reported during previous inspections.

Leaks detected shall be evaluated and repaired as required. The effectiveness of repairs shall be verified after completion.

Iowa Electric Light and Power Company is taking prompt action to reduce all leakages detected to as-low-as-practical levels. Corrective action is being documented and retests are being made. The preventive maintenance program requires regular reinspection of these systems and prompt corrective action where required.

This program is presently in process. The inspection of the RHR, Core Spray, HPCI, RCIC, and liquid radwaste systems has been accomplished. These are the systems most likely to carry significant quantities of radioactive fluid following an accident, and therefore, have been inspected first. The inspection of the RWCU and CRD systems must be deferred until plant shutdown. High radiation levels during plant operation limit access to the RWCU system. The scram discharge header is not available for this type of testing during reactor operation.

It is planned that the inspection of the containment atmospheric monitoring system will be completed during January 1980. This inspection will be performed by a contractor having expertise in helium testing.

The initial leak-testing and reduction program will be completed by the end of the outage scheduled to commence February 9, 1980. The initial leak rate test results of the systems already inspected are provided in Appendix D. Corrective action is being accomplished. Upon completion of the testing, the initial and corrected leak rate values will be reported to the NRC.

9. Design Review of Plant Shielding and Environmental Qualification of Equipment for Spaces/Systems Which May Be Used in Post-Accident Operations - NUREG 0578 Section 2.1.6.b

NRC POSITION:

With the assumption of a post-accident release of radioactivity equivalent to that described in Regulatory guides 1.3 and 1.4 (i.e., the equivalent of 50% of the core radioiodine, 100% of the core noble gas inventory, and 1% of the core solids, are contained in the primary coolant), each licensee shall perform a radiation and shielding design review of the spaces around systems that may, as a result of an accident, contain highly radioactive materials. The design review should identify the location of vital areas and equipment, such as the control room, radwaste control stations, emergency power supplies, motor control centers, and instrument areas, in which personnel occupancy may be unduly limited or safety equipment may be unduly degraded by the radiation fields during post-accident operations of these systems.

Each licensee shall provide for adequate access to vital areas and protection of safety equipment by design changes, increased permanent or temporary shielding, or post-accident procedural controls. The design review shall determine which types of corrective actions are needed for vital areas throughout the facility.

RESPONSE:

Iowa Electric Light and Power Company concurs with the BWR Owner's Group position on this matter (Reference 7).

A program for design review of plant shielding has been implemented for DAEC and is in progress. The criteria and objectives of this program are described in Appendix B. The radiation dose levels due to containment shine and equipment operation in the reactor building have been calculated and are discussed below.

Source terms used in the analyses are based upon and consistent with those provided in Reference 2.

The systems assumed to contain radioactive fluid are listed in Appendix B, and are indicated on attached piping and instrumentation drawings. General Electric is preparing a generic report on this topic and, when it is made available, the DAEC assumptions will be checked against it and modified as necessary.

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The only areas/equipment in the reactor building that may require access post-accident that have been identified to date are the reactor coolant and containment atmosphere sample stations and the stack effluent radiation monitors. General Electric is also preparing a generic report on access to equipment post-accident. If this report identifies any additional applicable access requirements, they will be included in the DAEC evaluation.

The radiation dose levels due to containment shine and equipment operation in the reactor building have been calculated and are shown in the drawings attached to Appendix B. The dose values shown are order-of-magnitude only, and do not include normal operating dose rates. Doses due to airborne sources and to filter loading in the standby gas treatment system will be determined when the results of system integrity testing (see response to Section 2.1.6.a) are available. Assumptions concerning airborne dose determination are discussed in Appendix B.

The calculations indicate that dose rates in the control room, turbine building access to north and south vent stack monitors, temporary technical support center, and operational support center are all less than 1 mR/hr at all times after the accident.

The dose levels at the containment atmosphere sample stations (el 757') are excessive. At one hour after the accident they are on the order of 134,000 R/hr, and at 30 days, 250 R/hr. The primary contributors to these levels are the CRD discharge headers which were assumed in the analysis to contain reactor coolant at the maximum assumed activity level. Indications are that this is not likely to be the case. Upon better definition of the source term within the headers, the dose levels within these areas will be reevaluated. As discussed in the response to NUREG 0578 Section 2.1.8.a, shielding to permit access to these sample stations would not be a minor modification. Since, as also discussed in the response to NUREG Section 2.1.8.a, the long-term design changes for post-accident sampling will include removal of the sample points from the reactor building, no long-term shielding changes are planned.

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Dose levels at the reactor coolant sample station (el 786') are also high. At one hour after the accident, dose level at the station is approximately 250 R/hr, and while drawing the sample, the dose rate increases to about 3,130 R/hr, at the design activity levels. As discussed in the response to NUREG 0578 Section 2.1.8.a, dose reduction and limitation techniques will be employed while sampling; however, it is apparent that taking a sample with maximum activity levels present will not be possible. Since the post-accident sample point will eventually be removed from the reactor building, no long-term shielding modifications are planned for the present sample station.

Upon completion of the final dose rate calculations, shielding modifications will be developed, and will be effected by January 1, 1981. The final results and modification plans will be submitted to the NRC for review prior to implementation.

An equipment environmental qualification evaluation in response to IE Bulletin 79-01 was provided in Reference 10. Upon completion of the dose rate calculations, including the airborne contribution, a reevaluation of environmental qualifications with respect to total integrated dose will be made.

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10. Automatic Initiation of the Auxiliary Feedwater System - NUREG 0578 Section 2.1.7a

NRC POSITION:

Consistent with satisfying the requirements of General Design Criterion 20 of Appendix A to 10 CFR Part 50 with respect to the timely initiation of the auxiliary feedwater system, the following requirements shall be implemented in the short term:

1. The design shall provide for the automatic initiation of the auxiliary feedwater system.

2. The automatic initiation signals and circuits shall be designed so that a single failure will not result in the loss of auxiliary feedwater system function.

3. Testability of the initiating signals and circuits shall be a feature of the design.

4. The initiating signals and circuits shall be powered from the emergency buses.

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

6. The a-c motor-driven pumps and valves in the auxiliary feedwater system shall be included in the automatic actuation (simultaneous and/or sequential) of the loads onto the emergency buses.

7. The automatic initiating signals and circuits shall be designed so that their failure will not result in the loss of manual capability to initiate the AFWS from the control room.

In the long term, the automatic initiation signals and circuits shall be upgraded in accordance with safety-grade requirements.

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RESPONSE:

Iowa Electric Light and Power Company concurs with the BWR Owners' Group position (Ref 3). This position concludes that this item is not applicable to BWRs.

11. Auxiliary Feedwater Flow Indication to Steam Generators - NUREG 0578, Section 2.1.7b

NRC POSITION:

Consistent with satisfying the requirements set forth in GDC 13 to provide the capability in the control room to ascertain the actual performance of the AFWS when it is called to perform its intended function, the following requirements shall be implemented:

1. Safety-grade indication of auxiliary feedwater flow to each steam generator shall be provided in the control room.

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

RESPONSE:

Iowa Electric Light and Power Company concurs with the BWR Owners' Group position (Ref 3). This position concludes that this item is not applicable to BWRs.

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12. Improved Post-Accident Sampling Capability NUREG 0578 Section 2.1.8.a

NRC POSITION:

A design and operational review of the reactor coolant and containment atmosphere sampling systems shall be performed to determine the capability of personnel to promptly obtain (less than 1 hour) a sample under accident conditions without incurring a radiation exposure to any individual in excess of 3 and 18-3/4 Rems to the whole body or extremities, respectively. Accident conditions should assume a Regulatory Guide 1.3 or 1.4 release of fission products. If the review indicates that personnel could not promptly and safely obtain the samples, additional design features or shielding should be provided to meet the criteria.

A design and operational review of the radiological spectrum analysis facilities shall be performed to determine the capability to promptly quantify (less than 2 hours) certain radioisotopes that are indicators of the degree of core damage. Such radionuclides are noble gases (which indicate cladding failure), iodines and cesiums (which indicate high fuel temperatures), and non-volatile isotopes (which indicate fuel melting). The initial reactor coolant spectrum should correspond to a Regulatory Guide 1.3 or 1.4 release. The review should also consider the effects of direct radiation from piping and components in the auxiliary building and possible contamination and direct radiation from piping and components in the auxiliary building and possible contamination and direct radiation from airborne effluents. If the review indicates that the analyses required cannot be performed in a prompt manner with existing equipment, then design modifications or equipment procurement shall be undertaken to meet the criteria.

In addition to the radiological analyses, certain chemical analyses are necessary for monitoring reactor conditions. Procedures shall be provided to perform boron and chloride chemical analyses assuming a highly radioactive initial sample (Regulatory Guide 1.3 or 1.4 source term). Both analyses shall be capable of being completed promptly; i.e., the boron sample analysis within an hour and the chloride sample analysis within a shift.

RESPONSE:

A design and operational review of the reactor coolant and containment atmosphere sampling systems has been performed to determine the capability of personnel to promptly obtain (less than 1 hour) a sample under accident conditions without incurring a radiation exposure to any individual in excess of 3 and 18-3/4 Rems to the whole body or extremities, respectively.

The DAEC Reactor Coolant Sample System was not designed for post-accident sampling because the sample line from the reactor recirculation loop contains containment isolation valves which can not be overridden when a main steam isolation signal is present. Procedures have been implemented to install temporary jumpers in the sample valve control logic (in the control panel) which would permit opening of these valves after an accident. The jumper would be installed after the accident occurred and would not affect the control logic of any other equipment. The sample hood has been provided with a short piece of tygon tubing to temporarily connect the sample point with the sample sink drain in order to minimize airborne contamination while the sample line is being purged. A shielded bottle will be used to collect the sample.

The DAEC Containment Atmosphere Monitoring System has provisions for post-accident containment atmosphere monitoring. The system contains redundant hydrogen, oxygen and particulate analyzers which are located on opposite sides of the containment in the reactor building. Each analyzer is provided with redundant pumps which permit containment atmosphere monitoring when the containment is at negative or positive pressure. The hydrogen and oxygen analyzers are designed to operate under post-LOCA conditions. The readout for all of the analyzers is in the main control room. Although the lines from the containment to the analyzers isolate with a containment isolation signal, the isolation valve switches on the control panel are provided with a keylocked override feature, which will permit opening of the valves with an isolation signal present. The particulate analyzer is not designed for post-accident radioactivity levels, and therefore, cannot be used for monitoring containment atmosphere radioactivity. A grab sample can be obtained from a valved sample point at one of the analyzers, so that the containment atmosphere radioactivity can be analyzed.

The reactor coolant and containment atmosphere radioactivity sampling capabilities may be limited by the severity of the accident and the resulting dose rates in the reactor building. In order to obtain reactor coolant samples with radioactivity levels corresponding to the source terms used in the response to NUREG Position 2.1.6.b, a sample return line to the containment would need to be added, and several other piping modifications, including the addition of remote operated valves and extensive shielding modifications, would be necessary. These are not considered to be minor modifications. Also, the reactor building may not be accessible, (see response to Section 2.1.6.b) thereby potentially precluding the ability to use such a system.

In order to ensure safety of the personnel taking the samples: 1) procedures have been written for taking and handling the samples (Refer to Appendix A); 2) dose rates along the access path to the sample stations have been analyzed (see response to Section 2.1.6); and 3) equipment for monitoring at the sample station has been provided.

A Post-Accident Sampling System is being designed to promptly take pressurized and unpressurized reactor coolant samples and a containment atmosphere sample for radioactivity levels corresponding to the source terms given in the clarifications to NUREG 0578 Item 2.1.6.b. Reference 2.

The system will have provisions for the following:

1. Containment atmosphere sampling under both negative and positive containment pressure

2. Purging of sample lines

3. Containment isolation valves meeting appropriate GDC

Post-accident sampling equipment will include an analysis facility having the capability to:

1. Identify and quantify -

    a. Certain isotopes that are indicators of the degree of core damage (i.e., noble gases, iodines and cesiums, and nonvolatile isotopes)-reactor coolant and containment atmosphere

    b. Hydrogen and oxygen levels in the containment atmosphere

    c. Dissolved hydrogen and cxygen in the reactor coolant

2. Dilute samples

3. Restrict backround levels of radiation

Installation of the post-accident sampling equipment will require the following plant modifications:

1. Construction of a sample station outside the reactor building

2. Installation of sample line piping, flush piping, drain piping and associated valves

3. Installation of shielding, as necessary, along pathways and areas used for obtaining, transporting, and analyzing samples

4. Construction of a sample analysis facility

All plant modifications and procedures necessary for post-accident sampling will be completed by January 1, 1981. The detailed system design will be submitted to the NRC prior to implementation of the modifications.

13. <u>Increased Range of Radiation Monitors - NUREG 0578 Section 2.1.8.b</u>

NRC POSITION:

The requirements associated with this recommendation should be considered as advanced implementation of certain requirements to be included in a revision to Regulatory Guide 1.97, "Instrumentation to Follow the Course of an Accident", which has already been initiated, and in other Regulatory Guides, which will be promulgated in the near-term.

1. Noble gas effluent monitors shall be installed with an extended range designed to function during accident conditions as well as during normal operating conditions; multiple monitors are considered to be necessary to cover the ranges of interest.

   a. Noble gas effluent monitors with an upper range capacity of 10^5 μCi/cc (Xe-133) are considered to be practical and should be installed in all operating plants.

   b. Noble gas effluent monitoring shall be provided for the total range of concentration extending from normal condition (ALARA) concentrations to a maximum of 10^5 μCi/cc (Xe-133). Multiple monitors are considered to be necessary to cover the ranges of interest. The range capacity of individual monitors should overlap by a factor of ten.

2. Since iodine gaseous effluent monitors for the accident condition are not considered to be practical at this time, capability for effluent monitoring of radioiodines for the accident condition shall be provided with sampling conducted by adsorption on charcoal or other media, followed by onsite laboratory analysis.

3. In-containment radiation level monitors with a maximum range of 10^8 rad/hr shall be installed. A minimum of two such monitors that are physically separated shall be provided. Monitors shall be designed and qualified to function in an accident environment.

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RESPONSE:

A) The January 1, 1980 requirements were clarified in Reference 2 as follows:

The licensee shall provide the following information on his methods to quantify gaseous releases of radioactivity from the plant during an accident.

1. Noble Gas Effluents

a. System/Method description including:

i. Instrumentation to be used including range or sensitivity, energy dependence, and calibration frequency and technique.

ii. Monitoring/sampling locations, including methods to assure representative measurements and backgound radiation correction.

iii. A description of method to be employed to facilitate access to radiation readings. For January 1, 1980. Control Room Readout is preferred: however, if impratical, in-situ readings by an individual with verbal communication with the Control Room is acceptable based on (iv) below.

iv. Capability to obtain radiation readings at least every 15 minutes during an accident.

v. Source of power to be used. If normal AC power is used, an alternate back-up power supply should be provided. If DC power is used, the source should be capable of providing continuous readout for 7 consecutive days.

b. Procedures for conducting all aspects of the measurement/analysis including:

i. Procedures for minimizing occupational exposures

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ii. Calculational methods for converting instrument readings to release rates based on exhaust air flow and taking into consideration radionuclide spectrum distribution as function of time after shutdown.

iii. Procedures for dissemination of information.

iv. Procedures for calibration.

2. Radioiodine and Particulate Effluents

A. For January 1, 1980 the licensee should provide the following:

1. System/Method description including:

a. Instrumentation to be used for analysis or the sampling media with discussion on methods used to correct for potentially interfering background levels of radioactivity.

b. Monitoring/sampling location.

c. Method to be used for retrieval and havdling of sampling media to minimize occupational exposure.

d. Method to be used for data analysis of individual radionuclides in the presence of high levels of radioactive noble gases.

e. If normal AC power is used for sample collection and analysis equipment, an alternate back-up power supply should be provided. If DC power is used, the source should be capable of prociding continuous readout for 7 consecutive days.

2. Procedures for conducting all aspects of the measurement analysis including:

a. Minimizing occupational exposure

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b. Calculational methods for determining release rates

c. Procedures for dissemination of information

d. Calibration frequency and technique

B) Iowa Electric Light and Power Company concurs with the BWR Owner's Group position on this matter as provided in Reference 3.

In response to NRC's request for information regarding the short-term criteria (January 1, 1980), the proposed methods to quantify gaseous releases of radiocativity from the plant during an accident are presented below. The radioactive gaseous effluents of interest are categorized as noble gases, iodines, and particulates. At the Duane Arnold Energy Center, the important gaseous effluent release points for accident conditions of operation include: off gas (OG), vent stack (one), reactor building vent stacks (three), and turbine building roof vents (eight). The control building, administration building, security building, MG set room (within reactor building), standby diesel generator room (within turbine building) and heater/boiler room (within radwaste building) have either an independent HVAC system or have air flows totally isolated from main plant HVAC systems. Airflows from the OG retention building, machine shop, and remainder of radwaste building are part of the reactor building HVAC system. In summary, the three effluent pathways: OG vent stack, reactor building vent stacks and turbine building roof vents constitute the common release points for which measures must be taken in order to comply with the January 1, 1980 criteria.

NOBLE GAS EFFLUENTS

Interim monitoring capabilities for high levels of radioactive noble gas effluents at DAEC have been accomplished using existing and currently available instrumentation. Placement of appropriate detectors inside lead shield stacks collimated to monitor small lengths of existing sampling lines and use of standard volume source will readily permit instrument readings to be converted to effluent stream conversion concentrations.

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Reactor Building Vent Stacks - Monitoring of the three exhaust ducts on this stream is provided by a small volume area radiation monitor (Eberline HP-200 or equivalent) placed within the collimator as described above. This device is stated to reliably detect gamma energies from 60 KeV to almost 2 MeV and to have a nominal response of 5 mR/hr (depending on area background) to about 50 R/hr. The shield serves to dramatically reduce local detector background and provides collimation of the sampling line to be monitored for high levels of noble gases. A remote readout device is driven by a local ratemeter system at the vent stack monitor. One such system monitors each of the exhaust duct sampling lines and is planned for continuous, unattended operation during accident conditions. This equipment is operated by normal plant ac power off the vital instrument bus which is served by the plant emergency standby power system in the event of offsite power loss. Built-in battery supply is also provided in each instrument.

Off Gas Stack Exhaust - The vent pipe to this system is monitored by a single system configured in essentially the same manner as described above for the reactor building exhaust ducts. This monitoring system is located in the off gas vent stack.

Turbine Building Roof Vents - Monitoring of releases via the turbine building roof vents is accomplished manually (as needed) by use of portable survey instruments. These instruments are used to measure the radioactive content of the effluent samplers in the same basic arrangement with a shield/collimator assembly as issued for both the reactor and off gas stack buildings. Conversion factors (volume source type) are then employed to correlate instrument readings to concentration values. Energy response of the portable instrumentation is stated to be 80 KeV to over 2 MeV (nominal) with an operating range from less than 0.5 mR/hr to more than 100 R/hr. There are eight roof vents in the turbine building, with sampling lines feeding existing equipment staged on platforms in pairs (serving two vents per platform). Both lines on each platform are exhausted to a mixing chamber and routed away via a common line. The high range noble gas monitoring locations are on the discharge line from each paired system. Thus four monitoring points serve the eight roof vents. Verbal communications

1695 340

between the individual(s) performing the monitoring and control room personnel may be maintained as needed using plant phones and/or portable transceiver units. The portable instruments are battery powered; hence, loss of onsite power does not affect their operational reliability.

Special procedures have been developed to gain access to monitoring points under adverse conditions (placing emphasis on radiological considerations), determine release rates through calculative methods, disseminate information, and calibrate equipment. (Refer to Appendix A for procedure identification.)

RADIOIODINE AND PARTICULATE EFFLUENTS

The existing plant sampling and monitoring equipment includes the capability of sampling for radioiodine and particulate effluents at all the release points discussed above in regard to noble gases. For purposes of implementing the interim criteria, existing sampling capabilities are employed in conjunction with special procedures for sample retrieval and analysis.

Quantification of sample radioiodine levels is accomplished using either existing counting room equipment for which standard counting procedures may be applicable or the portable sampler/analyzer system described in response to NRC Item 2.1.8.c which is specifically designed for use under accident conditions. Quantification of selected gamma-emitting radionuclides present in particulate samples is possible using these systems while other portable scaler systems may be used to determine gross beta-gamma levels of retrieved samples. Existing plant equipment available for this use is capable of operating on normal ac power or self contained battery systems. Procedures have been developed to address hot sample retrieval, handling and analysis (considering unusual radiological conditions), reporting results, and calibration of specialized equipment. (Refer to Appendix A for procedure identificaiton.)

CONTAINMENT HIGH RANGE MONITORS

By January 1, 1981 Iowa Electric Light and Power Company will have installed high range containment radiation monitors.

14. Improved In-Plant Iodine Instrumentation Under Accident Conditions - NUREG 0578 Section 2.1.8.c

NRC POSITION:

Each licensee shall provide equipment and associated training and procedures for accurately determining the airborne iodine concentration in areas within the facility where plant personnel may be present during an accident.

RESPONSE:

For purposes of implementing the short-term criteria no later than January 1, 1980, Iowa Electric Light and Power Company has provided a portable sampler/ analyzer system to measure I-131 concentrations which may be present in various region(s) of interest following an accident. This system is based on the use of an electronic scaler/detector having discrimination capabilities (baseline and window width). Primary detection is accomplished by "windowing" the 364.5 KeV (81.2%) photopeak of I-131 which the system is calibrated to measure. This system along with training of appropriate personnel in its calibration and use under special procedures, provides reasonable assurance that DAEC has the capability to accurately detect and thereby obtain an initial estimate of the presence of I-131 to determine if the use of respiratory protection equipment by plant personnel is warranted or required.

For the long-term (not later than January 1, 1981), Iowa Electric Light and Power Company is pursuing alternative methods of providing additional capability for further sample analysis to be performed in an area having relatively low background and contamination levels. Methods to effectively reduce interferences arising from accompanying radioactive noble gases in the sample medium (e.g., purging, special sampling media, etc.) will also be investigated.

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15. Transient and Accident Analysis - NUREG 0578 Section 2.1.9

NRC POSITION:

Analyses, procedures, and training addressing the following are required:

1. Small break loss-of-coolant accidents;

2. Inadequate core cooling; and

3. Transients and accidents.

Some analysis requirements for small breaks have already been specified by the Bulletins and Orders Task Force. These should be completed. In addition, pretest calculations of some of the Loss of Fluid Test (LOFT) small break tests (scheduled to start in September 1979) shall be performed as means to verify the analyses performed in support of the small break emergency procedures and in support of an eventual long term verification of compliance with Appendix K of 10 CFR Part 50.

In the analysis of inadequate core cooling, the following conditions shall be analyzed using realistic (best-estimate) methods:

1. Low reactor coolant system inventory (two examples will be required - LOCA with forced flow, LOCA without forced flow).

2. Loss of natural circulation (due to loss of heat sink).

These calculations shall include the period of time during which inadequate core cooling is approached as well as the period of time during which inadequate core cooling exists. The calculations shall be carried out in real time far enough that all important phenomena and instrument indications are included. Each case should then be repeated taking credit for correct operator action. These additional cases will provide the basis for developing appropriate emergency procedures. These calculations should also provide the analytical basis for the design of any additional instrumentation needed to provide operators with an unambiguous indication of vessel water level and core cooling adequacy (see Section 2.1.3.b in Appendix A to NUREG 0578).

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The analyses of transients and accidents shall include the design basis events specified in Section 15 of each FSAR. The analyses shall include a single active failure for each system called upon to function for a particular event. Consequential failures shall also be considered. Failures of the operators to perform required control manipulations shall be given consideration for permutations of the analyses. Operator actions that could cause the complete loss of function of a safety system shall also be considered. At present, these analyses need not address passive failures or multiple system failures in the short term. In the recent analysis of small break LOCAs, complete loss of auxiliary feedwater was considered. The complete loss of auxiliary feedwater may be added to the failures being considered in the analysis of transients and accidents if it is concluded that more is needed in operator training beyond the short-term actions to upgrade auxiliary feedwater system reliability. Similarly, in the long term, multiple failures and passive failures may be considered depending in part on staff review of the results of the short-term analyses.

The transient and accident analyses shall include event tree analyses, which are supplemented by computer calculations for those cases in which the system response to operator actions is unclear or these calculations could be used to provide important quantitative information not available from an event tree. For example, failure to initiate high-pressure injection could lead to core uncovery for some transients, and a computer calculation could provide information on the amount of time available for corrective action. Reactor simulators may provide some information in defining the event trees and would be useful in studying the information available to the operators. The transient and accident analyses are to be performed for the purpose of identifying appropriate and inappropriate operator actions relating to important safety considerations such as natural circulation, prevention of core uncovery, and prevention of more serious accidents.

The information derived from the preceding analyses shall be included in the plant emergency procedures and operator training. It is expected that analyses performed by the NSSS vendors will be put in the form of emergency procedure guidelines and that the changes in the procedures will be implemented by each licensee or applicant.

In addition to the analyses performed by the reactor vendors, analyses of selected transients should be performed by the NRC Office of Research, using the best available computer codes, to provide the basis for comparisons with the analytical methods being used by the reactor vendors. These comparisons together with comparisons to data, including LOFT small break test data, will constitute the short-term verification effort to assure the adequacy of the analytical methods being used to generate emergency procedures.

RESPONSE:

Iowa Electric Light and Power Company concurs with the BWR Owner's Group position on this matter as provided in Reference 3.

General Electric, under the direction of the BWR Owner's Group, is presently engaged in performing transient and accident analyses in response to NUREG 0578 Section 2.1.3.B (Inadequate Core Cooling), Section 2.1.9 (Transients and Accidents), and the small break loss-of-coolant accident requirements specified by the Bulletins and Orders Task Force. As discussed in the response to NUREG 0578 Section 2.1.3.b, the small break analyses have been completed.

The analysis of other transients and accidents for the purpose of upgrading emergency procedures is in process and the detailed scope and schedule of this analysis is the subject of continuing discussions between the BWR Owner's Group and the B&O task force.

As the analyses are completed and approved, Iowa Electric will modify the emergency procedures as appropriate.

16. Improve Containment Pressure, Water Level, and H2 Concentration Instrumentation - Enclosure 3 to Reference 1

NRC POSITION:

1. A continuous indication of containment pressure should be provided in the control room. Measurement and indication capability shall include three times the design pressure of the containment for concrete, four times the design pressure for steel, and minus five psig for all containments.

2. A continuous indication of containment water level shall be provided in the control room for all plants. A narrow range instrument shall be provided for PWRs and cover the range from the bottom to the top of the containment sump. A wide range instrument shall also be provided for PWRs and shall cover the range from the bottom of the containment to the elevation equivalent to a 600,000 gallon capacity. For BWRs, a wide range instrument shall be provided and cover the range from the bottom to 5 feet above the normal water level of the suppression pool.

3. A continuous indication of hydrogen concentration in the containment atmosphere shall be provided in the control room. Measurement capability shall be provided over the range of 0 to 10% hydrogen concentration under both positive and negative ambient pressure.

RESPONSE:

Iowa Electric Light and Power Company concurs with the BWR Owner's Group position on this matter as provided in Reference 3, and is participating in a qualification program being conducted by General Electric for this instrumentation.

The containment pressure instrumentation will provide continuous indication in the control room. The range will be -5 psig to three times design pressure. This instrumentation will meet the design provisions of Regulatory Guide 1.97, including qualification, redundancy, and testability, per the Reference 2 clarifications.

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The containment water level instrumentation will provide continuous indication in the control room. The range will be from, at, or below the lowest ECCS pump suction to 5 feet above the normal water level of the torus, per the Owner's Group position. The instrumentation will meet the design provisions of the proposed revision to Regulatory Guide 1.97 per the Reference 2 clarifications.

The hydrogen instrumentation will provide continuous indication in the control room. The range will be 0 to 10% H_2 under both positive and negative ambient pressure. The instrumentation will meet the design provisions of Regulatory Guide 1.97 including qualification, redundancy, and testability per the Refernece 2 clarifications.

The above instrumentation will be installed by January 1, 1981.

1695 347

17. Reactor Coolant System Venting - Enclosure 4 to Reference 1

NRC POSITION:

Each applicant and licensee shall install reactor coolant system and reactor vessel head high point vents remotely operated from the control room. Since these vents form a part of the reactor coolant pressure boundary, the design of the vents shall conform to the requirements of Appendix A to 10 CFR Part 50, General Design Criteria. In particular, these vents shall be safety grade, and shall satisfy the single failure criterion and the requirements of IEEE-279 in order to ensure a low probability of inadvertent actuation.

Each applicant and licensee shall provide the following information concerning the design and operation of these high point vents:

a) A description of the construction, location, size, and power supply for the vents along with results of analyses of loss-of-coolant accidents initiated by a break in the vent pipe. The results of the analyses should be demonstrated to be acceptable in accordance with the acceptance criteria of 10 CFR Part 50.46.

b) Analyses demonstrating that the direct venting of noncondensable gases with perhaps high hydrogen concentrations does not result in violation of combustible gas concentration limits in containment as described in 10 CFR Part 50.44, Regulatory Guide 1.7 (Rev 1), and Standard Review Plant Section 6.2.5.

c) Procedural guidelines for the operators' use of the vents. The information available to the operator for initiating or terminating vent usage shall be discussed.

RESPONSE:

Iowa Electric Light and Power Company concurs with the BWR Owners' Group position on this matter (Ref 3).

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The Owners' Group has concluded that adequate reactor coolant system venting is provided by existing plant designs.

Duane Arnold Energy Center is provided with four power-operated safety grade relief valves (ADS valves) remotely operable from the main control room. These valves are capable of venting the reactor coolant system of noncondensible gases and they discharge to the suppression pool.

Procedures have been provided to govern the operator's use of the relief valves for venting the reactor pressure vessel. These procedures are identified in Appendix A.

The DAEC FSAR provides a description of the construction, location, size, and power supply for these valves in Section 4.4.5, Figure 4.4-1, Table 4.4-14, and Section 7.4.3.3.2, respectively. An analysis of a more severe loss-of-coolant accident is contained in Section 14.6.3.

An analysis of post-accident combustible gas concentrations and methods to ensure uniform mixing are included in Section G.7-1 of the FSAR and in the response to Question G.7-4.

As a backup to ADS valve operation, the reactor vessel can also be vented by the reactor pressure vessel head vent line which contains two nitrogen-operated valves in series which are remotely operable from the control room. These valves are normally closed with solenoids that are normally deenergized. These valves, while not environmentally qualified, are powered from an emergency power source. The reactor pressure vessel head vent line discharges to the dry well equipment drain sump. In addition, the reactor vessel is vented to a main steam line through a normally open, manually operated valve. These vent lines are shown in FSAR Figure 4.4-1.

Interim and permanent measures to provide long-term nitrogen supply to the ADS and head vent valve operators are discussed in the response to NUREG Section 2.1.1.

Venting is also provided by the main steam driven turbines of the High Pressure Coolant Injection (HPCI) and Reactor Core Isolation Cooling (RCIC) systems which exhaust to the suppression pool.

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The Residual Heat Removal system (RHR) heat exchangers can be vented of noncondensible gases by motor-operated valves remotely operable from the main control room. These vents discharge to the suppression pool.

The HPCI, RCIC, and RHR systems are shown in FSAR Figures 7.4-2, 4.7-1, and 4.8-2, respectively.

General Electric, in conjunction with Terry Turbine, is investigating the effects of noncondensibles in the operation of the HPCI and RCIC turbines. The results of this investigation will be transmitted when received.

1695 350

18. <u>Shift Supervisor Responsibilities - NUREG 0578 Section 2.2.1.a</u>

<u>NRC POSITION:</u>

1. The highest level of corporate management of each licensee shall issue and periodically reissue a management directive that emphasizes the primary management responsibility of the shift supervisor for safe operation of the plant under all conditions on his shift and that clearly establishes his command duties.

2. Plant procedures shall be reviewed to assure that the duties, responsibilities, and authority of the shift supervisor and control room operators are properly defined to effect the establishment of a definite line of command and clear delineation of the command decision authority of the shift supervisor in the control room relative to other plant management personnel. Particular emphasis shall be placed on the following:

    a. The responsibility and authority of the shift supervisor shall be to maintain the broadest perspective of operational conditions affecting the safety of the plant as a matter of highest priority at all times when on duty in the control room. The idea shall be reinforced that the shift supervisor should not become totally involved in any single operation in times of emergency when multiple operations are required in the control room.

    b. The shift supervisor, until properly relieved, shall remain in the control room at all times during accident situations to direct the activities of control room operators. Persons authorized to relieve the shift supervisor shall be specified.

    c. If the shift supervisor is temporarily absent from the control room during routine operations, a lead control room operator shall be designated to assume the control room command function. These temporary duties, responsibilities, and authority shall be clearly specified.

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3. Training programs for shift supervisors shall emphasize and reinforce the responsibility for safe operation and the management function the shift supervisor is to provide for assuring safety.

4. The administrative duties of the shift supervisor shall be reviewed by the senior officer of each utility responsible for plant operations. Administrative functions that detract from or are subordinate to the management responsibility for assuring the safe operation of the plant shall be delegated to other operations personnel not on duty in the control room.

<u>RESPONSE</u>:

Iowa Electric Light and Power Company concurs with the BWR Owner's Group position on this matter, as provided in Reference 3.

A management directive has been issued emphasizing the shift supervisor's primary responsibility for the safe operation of the plant. Procedures have been implemented reflecting the NUREG position and as clarified in Reference 2. These procedures are identified in Appendix A.

19. Shift Technical Advisor - NUREG 0578 Section 2.2.1.b

NRC POSITION:

Each licensee shall provide an on-shift technical advisor to the shift supervisor. The shift technical advisor may serve more than one unit at a multi-unit site if qualified to perform the advisor function for the various units.

The Shift Technical Advisor shall have a bachelor's degree or equivalent in a scientific or engineering discipline and have received specific training in the response and analysis of the plant for transients and accidents. The Shift Technical Advisor shall also receive training in plant design and layout, including the capabilities of instrumentation and controls in the control room. The licensee shall assign normal duties to the shift technical advisors that pertain to the engineering aspects of assuring safe operations of the plant, including the review and evaluation of operating experience.

RESPONSE:

A shift technical advisor (STA) will be provided on a continuous basis during power operation commencing January 1, 1980. The duties, responsibilities, and qualifications of STAs will be in accordance with the NUREG position and as clarified in References 2 and 4.

The STAs will all be graduates of a 4-year college engineering or scientific degree program or equivalent and will be experienced in the design of commercial nuclear power plants. Additional personnel qualifications will be from among the following: advanced technical degree, naval nuclear propulsion program training and experience, plant startup experience, and experience with public utilities. These personnel will familiarize themselves with the specific Duane Arnold Energy Center design as described in the FSAR prior to start of work. Either prior to or during their initial work periods, they will become familiar with DAEC operating and emergency procedures. The initial crew of STAs will receive indoctrination in shift operations by plant personnel. Follow-on crews will receive this indoctrination from off-going crews during overlapping turnover watches.

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STAs will work under the direction of Iowa Electric Light and Power Company's Nuclear Generation Division Manager, Design Engineering.

Iowa Electric Light and Power Company is currently conducting a recruiting program, both within and outside of the company to enhance the number of STAs available.

The requirements and responsibilities of the STAs will include the following:

a. The STA will be stationed onsite and will be present in the control room within 10 minutes of being summoned during plant power operation.

b. The STA will perform assessment functions during off-normal transients and accidents.

c. The STAs will provide operating experience assessment functions as related to Duane Arnold design, procedures, and practice, and in support of their transient/accident assessment function.

d. In the performance of these duties, the STAs will be independent from duties associated with the commercial operation of the plant and will report directly to the Manager, Design Engineering.

During the year of 1980, a formal program will be developed concerning the STA position. This program will encompass the job description, qualifications, training and training programs, organization, operating experience function, and other aspects of the position. This program will be submitted to the NRC for review prior to its implementation, per Reference 2.

The formal program will be implemented, with STAs trained and qualified in accordance with its provisions, by January 1, 1981.

20. Shift and Relief Turnover Procedures - NUREG 0578 Section 2.2.1.c

NRC POSITION:

The licensees shall review and revise as necessary the plant procedure for shift and relief turnover to assure the following:

1. A checklist shall be provided for the oncoming and offgoing control room operators and the oncoming shift supervisor to complete and sign. The following items, as a minimum, shall be included in the checklist.

   a. Assurance that critical plant parameters are within allowable limits (parameters and allowable limits shall be listed on the checklist).

   b. Assurance of the availability and proper alignment of all systems essential to the prevention and mitigation of operational transients and accidents by a check of the control console.

      (what to check and criteria for acceptable status shall be included on the checklist);

   c. Identification of systems and components that are in a degraded mode of operation permitted by the Technical Specifications. For such systems and components, the length of time in the degraded mode shall be compared with the Technical Specifications action statement (this shall be recorded as a separate entry on the checklist).

2. Checklists or logs shall be provided for completion by the offgoing and ongoing auxiliary operators and technicians. Such checklists or logs shall include any equipment under maintenance or test that by themselves could degrade a system critical to the prevention and mitigation of operational transients and accidents or initiate an operational transient (what to check and criteria for acceptable status shall be included on the checklist); and

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3. A system shall be established to evalute the effectiveness of the shift and relief turnover procedure (for example, periodic independent verification of system alignments).

RESPONSE:

Iowa Electric Light and Power Company concurs with the BWR Owner's Group position on this matter, as provided in Reference 3.

A control room panel shift checklist has been developed and shift turnover procedures have been revised in accordance with the NUREG position. This procedure is identified in Appendix A.

21. Control Room Access - NUREG 0578 Section 2.2.2.a

NRC POSITION:

The licensee shall make provisions for limiting access to the control room to those individuals responsible for the direct operation of the nuclear power plant (e.g., operations supervisor, shift supervisor, and control room operators), to technical advisors who may be requested or required to support the operation, and to predesignated NRC personnel. Provisions shall include the following:

1. Develop and implement an administrative procedure that establishes the authority and responsibility of the person in charge of the control room to limit access, and

2. Develop and implement procedures that establish a clear line of authority and responsibility in the control room in the event of an emergency. The line of succession for the person in charge of the control room shall be established and limited to persons possessing a current senior reactor operator's license. The plan shall clearly define the lines of communication and authority for plant management personnel not in direct command of operations, including those who report to stations outside of the control room.

RESPONSE:

Iowa Electric Light and Power Company concurs with the BWR Owner's Group position on this matter, as provided in Reference 3.

Procedures have been implemented to effect the provisions of this NUREG position. The procedure is identified in Appendix A.

22. <u>Onsite Technical Support Center (TSC) - NUREG 0578 Section 2.2.2.b</u>

NRC POSITION:

Each operating nuclear power plant shall maintain an onsite technical support center separate from and in close proximity to the control room that has the capability to display and transmit plant status to those individuals who are knowledgeable of and responsible for engineering and management support of reactor operations in the event of an accident. The center shall be habitable to the same degree as the control room for postulated accident conditions. The licensee shall revise his emergency plans as necessary to incorporate the role and location of the technical support center. Records that pertain to the as-built conditions and layout of structures, systems and components shall be readily available to personnel in the TSC.

RESPONSE:

Iowa Electric Light and Power Company concurs with the BWR Owner's Group position on this matter, Reference 3, and as commented upon by the NRC in Reference 4.

A temporary technical support center will be provided by January 1, 1980, and a permanent support center will be available by January 1, 1981. The temporary support center will satisfy the clarification criteria (Items 1.A-1.G) of Reference 2, and the permanent support center will be in accordance with the criteria provided in the NUREG position and Reference 2 clarifications.

Details concerning the Temporary Technical Support Center (TTSC) are as follows:

LOCATION

The TTSC is located in Room A-203 (presently designated Conference Room) on the second floor (el 772'-6") of the Administration Building. The TTSC is easily accessible from the control room which is located on the third floor (el 786'-0") of the control building and is within a walking distance of 2 minutes or less. The relative location of TTSC is shown in Figures 1 and 2.

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RADIATION MONITORING

a. A selfcontained portable area monitor with visual and audible warning of hazardous radiation levels has been provided for monitoring direct radiation.

b. A portable continuous air monitoring system with visual and audible alarms has been provided to monitor airborne radioactive contaminants.

Direct shine radiation levels in the TTSC have been evaluated as acceptable as discussed in the response to Section 2.1.6.b.

Evacuation from the TTSC will occur at approximately 300 millirems/day equivalent dose rate. It may be necessary to revise this figure depending on length of accident duration; however, it will not exceed the levels specified in GDC 19 and SRP 6.4.

INSTRUMENTATION

A closed circuit TV (CCTV) system has been established to have direct display of plant parameters. This is accomplished by installation of two high performance low-light level cameras located in front of the control panels. Each of these two cameras will monitor a separate section of the control panel. The cameras have pan, tilt, and zoom capability and are capable of displaying any instrument or device on the main control panel including the CRTs. TV monitors and control units for these cameras are located in the TTSC.

The use of the CCTV system will provide accurate and reliable indication of plant parameters to TSC personnel, without possibility of degradation of the control room instrumentation. The CCTV system is powered from Class 1E power feeding the security CCTV system in the control room.

The floor area of the TTSC is 300 square feet, which will accommodate approximately 10 people, after accounting for desks, tables, chairs, etc.

The TTSC lighting and ventilation systems receive power from nonessential local lighting panels in the administration building which in turn receive power from station nonessential distribution. Ventilation is provided by the HVAC unit in the administration building.

STAFFING

Staffing of the TTSC is dictated by the Emergency Plan and summarized here:

The Assistant Chief Engineer who is responsible for all operations and maintenance activities at DAEC will be designated as the TSC supervisor and will exercise supervision and direction of personnel assigned to the TSC. In his absence, his responsibilities will be assumed by the Chief Engineer, or if he too is not present, by the Assistant Chief Engineer responsible for technical and administrative activities. Personnel who will assemble in that location include: selected members of the plant staff who are knowledgeable in one or more specific disciplines or functional areas at DAEC; selected IELP corporate engineering and licensing personnel who can assist in providing engineering evaluations, coordinating support engineering groups who may be called upon for assistance, or provide assistance with respect to licensing requirements and interface with the NRC; representatives from the NRC; and vendor representatives, including GE and Bechtel.

COMMUNICATIONS

Communication between the TTSC and the control room will be by a sound powered telephone and by a dial telephone that is part of the plant telephone system.

Communication to the NRC will be by an extension to the existing dedicated "hot line" to the NRC.

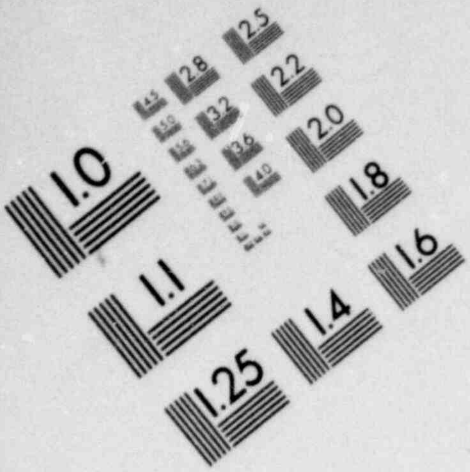
1695 360

IMAGE EVALUATION
TEST TARGET (MT-3)

6″

IMAGE EVALUATION
TEST TARGET (MT-3)

6″



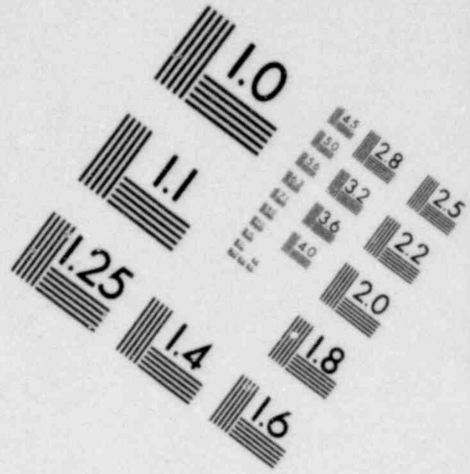
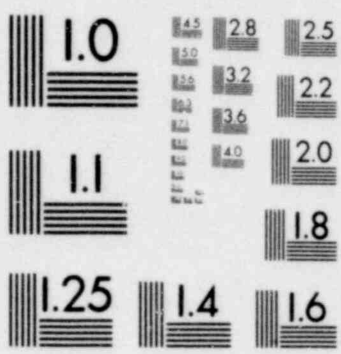


IMAGE EVALUATION
TEST TARGET (MT-3)



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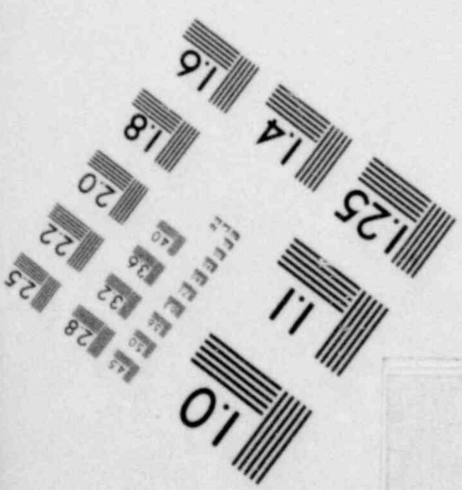


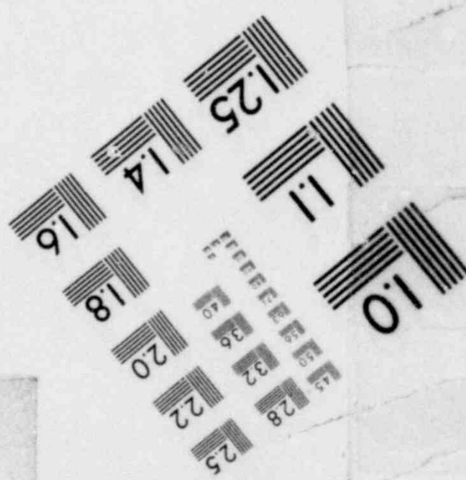


IMAGE EVALUATION TEST TARGET (MT-3)

6″

A dial telephone, which is part of the plant telephone system, will be used for all offsite communications, including near site emergency operations center. The main telephone switchboard which is within 30 feet of the TTSC will also be used for this purpose. (See Figure 1 for location of the telephone switchboard.)

TECHNICAL DOCUMENTS

Technical documents are available in the technical document storage area which is located within 40 feet of the TTSC. (See Figure 1 for location of the document area.)

Documents available include:

1. Drawing control report
2. Safety analysis reports
3. Plant operating procedures
4. Piping and instrument diagrams
5. Flow diagrams
6. Logic diagrams
7. Single line diagrams
8. Piping area drawings
9. Equipment location drawings

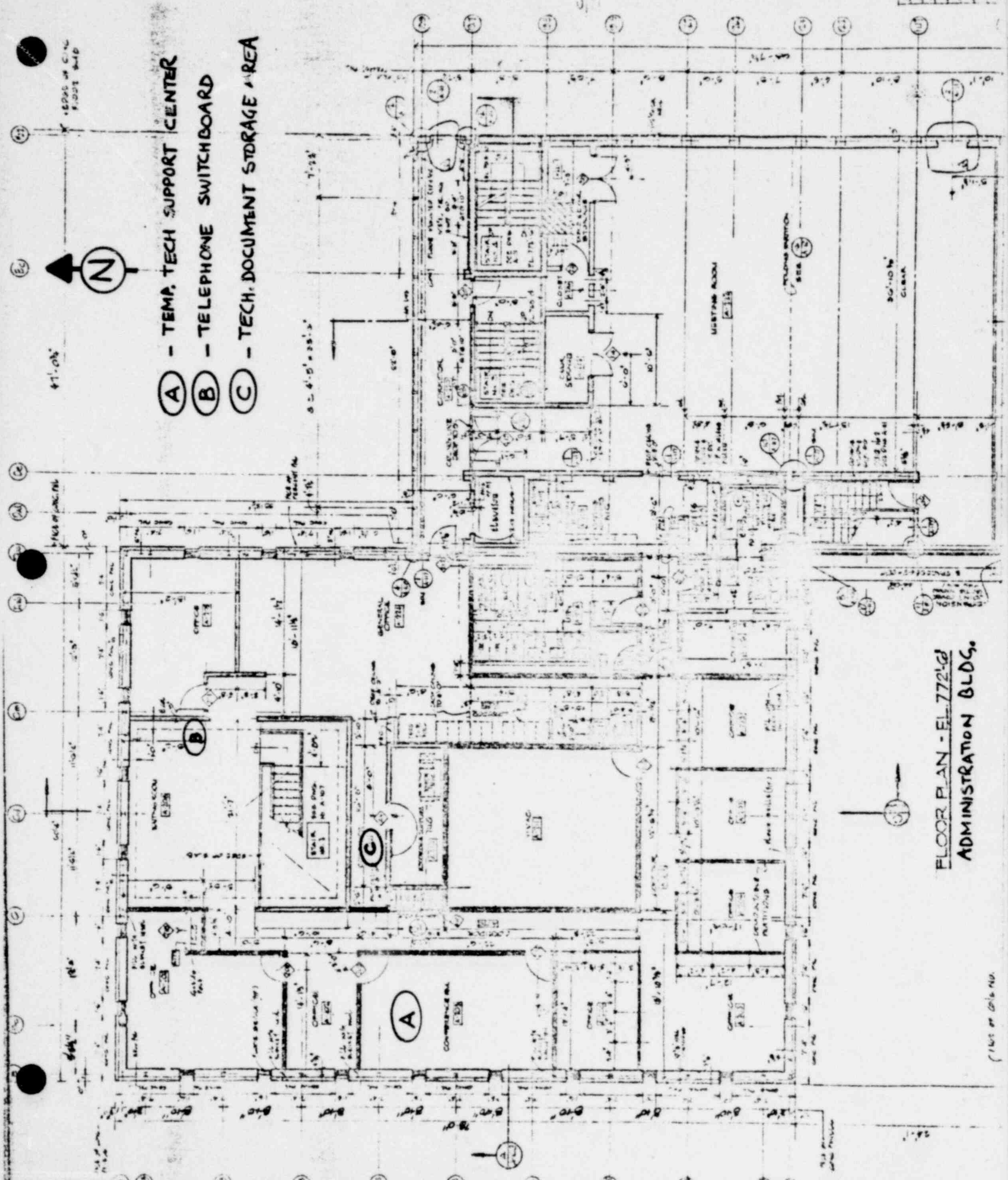
ALTERNATIVE TTSC

In the event the TTSC becomes unavailable, the Emergency Plan designates the Control Room as the alternative TSC. The emergency plan contains procedures for relocation to the control room.

PERMANENT TSC

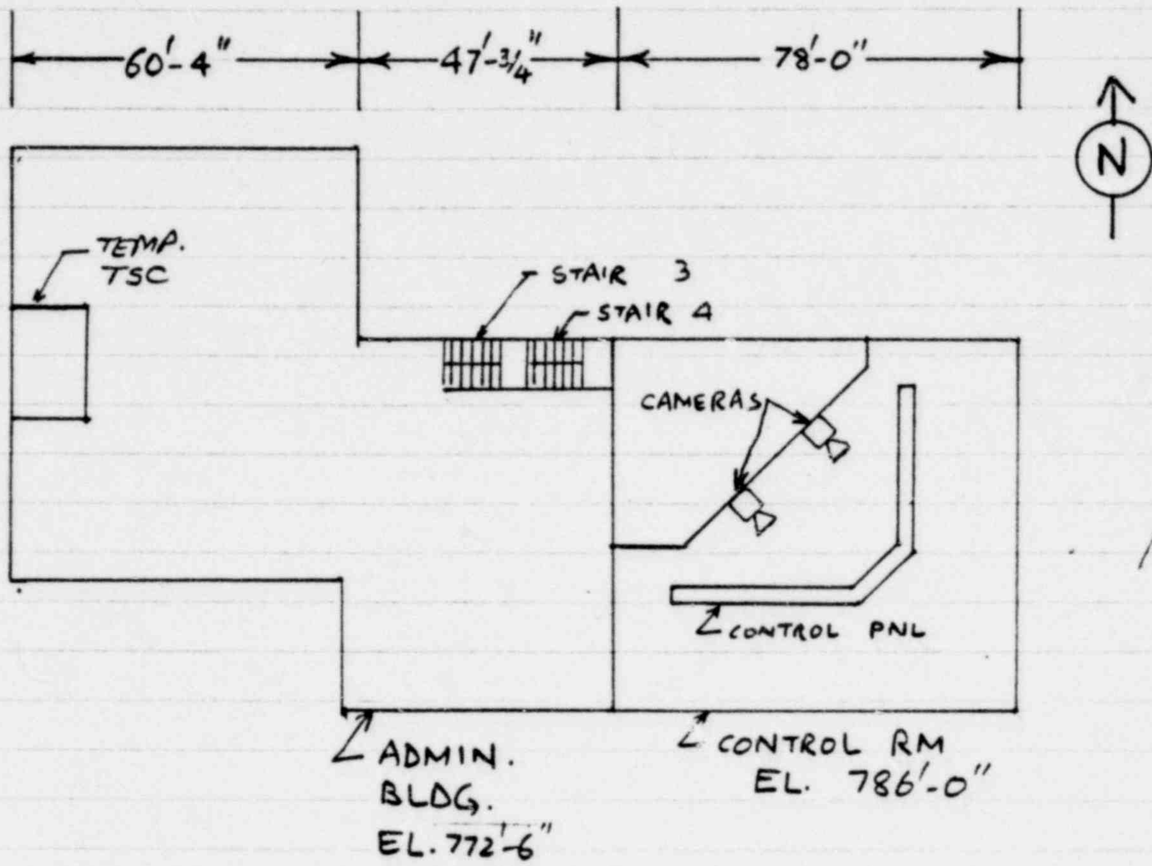
Plans for a permanent TSC are being developed. The permanent TSC will be housed in a new structure. The preliminary design criteria and conceptual design description of the permanent facility are provided in Appendix C.

FIGURE 1



1696 002

FIGURE 2



TEMP. TECH. SUPPORT CENTER

1696 003

23. Onsite Operational Support Center - NUREG 0578 Section 2.2.2.c

NRC POSITION:

An area to be designated as the onsite operational support center shall be established. It shall be separate from the control room and shall be the place to which the operations support personnel will report in an emergency situation. Communications with the control room shall be provided. The emergency plan shall be revised to reflect the existence of the center and to establish the methods and lines of communication and management.

RESPONSE:

Iowa Electric Light and Power Company concurs with the BWR Owner's Group position on this matter, as provided in Reference 3.

An onsite operational support center has been established and specified in the emergency plan. The center is located in the administration building and is composed of the security control point and adjacent locker room. Communications within the control room are provided by existing security control point facilities.

1696 004

REFERENCES:

1. Letter, D.G. Eisenhut to All Operating Nuclear Power Plants, "Followup Actions Resulting From NRC Staff Reviews Regarding the Three Mile Island Unit 2 Accident," September 13, 1979

2. Letter, Harold R. Denton to All Operating Nuclear Power Plants, "Discussion of Lessons Learned Short-Term Requirements," October 30, 1979

3. Letter, Thomas D. Keenan to D.G. Eisenhut, "BWR Owner's Group Positions on NUREG 0578," October 17, 1979

4. Letter, D.G. Eisenhut to Thomas D. Keenan, November 14, 1979

5. Letter, Larry D. Root, Iowa Electric Light and Power Company, to Harold R. Denton, November 20, 1979

6. Letter, Thomas D. Keenan to D.G. Eisenhut, "BWR Owner's Group Implementation of NUREG 0578, Requirements 2.1.2," December 11, 1979

7. Letter, Thomas D. Keenan to D.G. Eisenhut, "BWR Owner's Group Positions on NUREG 0578," also enclosed letter, R.H. Buchholz (GE) to D.G. Eisenhut, November 15, 1978

8. Letter, D.G. Eisenhut to Thomas D. Keenan, November 27, 1979

9. Letter, Thomas D. Keenan to Dr. D.F. Ross, Jr. "Revised NEDO-24708 Small Break Operator Guidelines," November 16, 1979

10. Letter, Larry D. Root, Iowa Electric, to Norman C. Moseley, June 11, 1979

1696 005

APPENDIX A

DUANE ARNOLD ENERGY CENTER

PROCEDURES TO IMPLEMENT

NUREG-0578 for JANUARY 1, 1980

Iowa Electric Light

and Power Company

January 1, 1980

1606 006

APPENDIX A

DAEC Procedures To Implement NUREG-0578 for January 1, 1980

2.1.1 EMERGENCY POWER SUPPLY

Operating Order 2-82

2.1.3.a DIRECT INDICATION OF POWER-OPERATED RELIEF VALVE AND SAFETY VALVE POSITION FOR PWRs AND BWRs

Integrated Plant Operating Instruction Volume 2 Section II Procedure B.10

2.1.3.b INSTRUMENTATION FOR DETECTION OF INADEQUATE CORE COOLING

Integrated Plant Operating Instruction Volume 2 Section II Procedures B.5, B.9, B.10, B.11, B.12

2.1.6.a INTEGRITY OF SYSTEMS OUTSIDE CONTAINMENT LIKELY TO CONTAIN RADIOACTIVE MATERIALS FOR PWRs AND BWRs

Procedure for immediate and continuing leak reduction program

2.1.8.a IMPROVED POST-ACCIDENT SAMPLING CAPABILITY

1) Radiation Protection Procedure 13.3, 13.4
2) Operating Order 2-82

2.1.8.b INCREASED RANGE OF RADIATION MONITORS

1) Radiation Protection Procedure 13.2
*2) Radiation Protection Procedure 13.5, 13.6

2.1.8.c IMPROVED IN-PLANT IODINE INSTRUMENTATION UNDER ACCIDENT CONDITIONS

*1) Radiation Protection Procedure 13.7, 13.8

2.1.9 TRANSIENT AND ACCIDENT ANALYSIS

Integrated Plant Operating Instruction Volume 2 Section II Procedures B.11 and B.12

Denton REACTOR COOLANT SYSTEM VENTING

1) Integrated Plant Operating Instruction Volume 1 Section V
2) Operating Instruction 62/80/83

1696 007

2.2.1.a SHIFT SUPERVISOR RESPONSIBILITIES

1) Quality Directive 1302.2 Rev. 3
2) Administrative Control Procedure 1404.1 Rev. 2

2.2.1.b SHIFT TECHNICAL ADVISOR

1) Administrative Control Procedure 1404.1 Rev. 2

2.2.1.c SHIFT AND RELIEF TURNOVER PROCEDURES

Basic Statistics Procedures No. 5

2.2.2.a CONTROL ROOM ACCESS

1) Special Order 298

2.2.2.b ONSITE TECHNICAL SUPPORT CENTER (TSC)

1) Special Order 298

2.2.2.c ONSITE OPERATIONAL SUPPORT CENTER

1) Preparedness Plan Section 4.2
2) Preparedness Plan Implementation Procedure 2

* Procedure will be implemented when equipment is delivered and available for use.

1696 008

APPENDIX B

CRITERIA AND OBJECTIVES

FOR

DESIGN REVIEW OF PLANT SHIELDING

NUREG 0578, SECTION 2.1.6.b

IOWA ELECTRIC LIGHT AND POWER COMPANY

JANUARY 1, 1980

1696 009

NUREG 0578 Section 2.1.6.b - Design Review of Plant Shielding

Duane Arnold Energy Center

I. CRITERIA FOR DESIGN REVIEW OF PLANT SHIELDING OF SPACES FOR POST-ACCIDENT OPERATIONS FOR BWR's

Identification of systems outside the primary containment which may contain high concentrations of radioactive materials following an accident have been classified into two categories as follows:

A. Those systems designed to mitigate a design basis loss of coolant accident.

B. Those systems directly connected to the reactor coolant system, which, while neither designed to nor expected to contain high concentrations of radioactive material post-accident, are postulated to contain such material.

Category A Systems

1. Portions of the RHR system during the following accident modes;

   a. LPCI Mode - the portions used to recirculate the suppression pool water, including containment spray piping.

   b. Shutdown Cooling Mode - Portions to provide "normal" residual heat removal service.

   c. Steam Condensing Mode - Portions used for condensing steam in the RPV vapor dome.

   d. Suppression Pool Cooling Mode - Portions used to cool the suppression pool water.

2. Portions of the core spray (CS) system used to recirculate the suppression pool water.

3. Portions of the HPCI system as follows:

   a. Steam supply piping and turbine exhaust piping, including these portions of the HPCI system barometric condenser which carry non-condensible gases to the standby gas treatment system.

   b. Pump suction and discharge piping during suppression pool recirculation.

1696 010

4. Portions of the reactor core isolation cooling (RCIC) systems as follows:

   a. Steam supply piping and turbine exhaust and associated piping.

   b. Pump suction and discharge piping during the steam condensing mode of RPV cooldown.

   c. Pump suction and discharge piping during reactor makeup from suppression pool.

5. Portions of the sampling systems required to meet the intent of NUREG 0578 Section 2.1.8.a.

6. Primary containment purge piping to the standby gas treatment system.

7. Standby Gas Treatment System filters.

<u>Category B Systems</u>

1. Portions of the reactor water cleanup (RWCU) system up to the second isolation valve.

2. The control rod drive hydraulic system (CRDHS) scram discharge piping.

Detailed depiction of the piping involved is shown in the attached figures.

II. IDENTIFICATION OF POTENTIAL SOURCES

The following releases have been used as a basis for determining the concentrations for shielding:

A. Source A*: containment airborne: 100% noble gases, 25% halogens

B. Source B*: reactor liquid: 100% noble gases, 25% halogens

C. Source C: suppression pool liquid: 50% halogens, 1% solids

D. Source D: reactor steam: 100% noble gases, 25% halogens

*From H. R. Denton clarification letter of October 30, 1979.

III. APPLICATION OF SOURCES TO SHIELDING

In applying the releases defined in Section II to generate activity concentrations, consideration will be given to dilution and decay as follows:

A. Decay

The time after the accident when operator action or system operation is required can be the most significant factor in evaluating shielding. For analyses or personnel exposures in vital areas outside the control room, radioactive decay equivalent to the plant specific licensing basis LOCA delay will be assumed. For analysis of integrated exposures to equipment, radioactive decay equivalent to the minimum time to initiation of system operation may be used.

B. Dilution

The volume used for dilution can al o be important, affecting the calculations of dose rate in a line r fashion.

1. Source A - drywell free volume

2. Source B - for systems recirculating the reactor coolant, the reactor coolant system normal liquid volume (Based upon reactor coolant density at the operating temperature and pressure) was used.

3. Source C - for systems recirculating suppression pool water, the lume of water present at the time of recirculation (reactor coolant system + suppression pool water) is used.

4. Source D - for steam condensing and turbine operations, reactor coolant system normal vapor volume is used.

C. Sources to be used for piping and equipment in each system.

In defining the limits of the connected piping subject to such contamination, normally shut valves are assumed to remain shut.

<u>Category A Systems</u>

1. RHR - mode 1.a[(1)] - use source C with credit for decay available in the FSAR

   mode 1.b[(1)] - use source B with credit for decay available in the FSAR

   mode 1.c - use source D with credit for decay available in FSAR

   mode 1.d - use source C with credit for decay available in FSAR

2. CS - use source C with credit for decay available in the FSAR.

3. HPCI steam - mode 3.a use source D with credit for decay available in the FSAR - mode 3.b use source C

4. RCIC - mode 4.a[(1)] - use source D with credit for decay available in the FSAR

   mode 4.b[(1)] - use source D with credit for decay available in the FSAR.

   mode 4.c - use source C with credit for decay available in the FSAR.

5. Sampling - the sources in the samples will be dependent upon the time the sample is drawn. For this evaluation, samples were assumed to be taken at time 1 hour after the accident.

1696 012

(a) Containment Air Sample - Source A

(b) Reactor Coolant Sample - Source B

(c) Suppression Waters Sample - Source C

6. Primary containment purge piping - source A with credit for decay as available in the FSAR.

7. Standby Gas Treatment System - filters use source A with credit for decay as available in FSAR.

Category B Systems

1. RWCU - source B with credit for decay available per the FSAR.

2. CRDHS - source B with credit for decay per the FSAR.

D. Airborne Sources

The design review of shielding is performed to determine:

(1) personnel access to areas within the reactor building.

(2) personnel access to areas outside the reac - building (e.g., control room) and,

(3) integrated acicdent doses for equipment qualification.

In the first case, shielding cannot necessarily be provided to reduce the airborne dose rate. Therefore a realistic assessment of the feasibilty of access is required. In the second and third cases additional shielding can be provided, and assurance of access/qualification is essential. Therefore a conservative assessment in these cases is appropriate.

Accordingly, the following source terms will apply:

A. Personnel access to reactor building:

1. Airborne source is system leakage rate from system integrity tests (NUREG Position 2.1.6.a) after corrective action has been taken plus containment leakage rate form last full pressure integrated leak rate test minus one-half the lastest MSN local leak rate valve. (Post-Accident, the MSN leakage control systme exhausts to SGTS. Local leak rate valves are conservative, therefore subtraction form the total leak rate valve would by non-conservative. One-half the local valve is considered a reasonable estimate). This leakage includes the small amount of condensate that will drain from atmospheric monitoring system to equipment drain sumps in the reactor building.

2. SGTS filter loading is based on the system leak rates after connective action plus containment leak rate from last full pressure test.

1696 013

3. Calculated dose is external dose only. Full respiratory protection is assumed.

4. Doses are calculated taking room size into acocunt. An "infinite cloud" is not assumed.

B. Personnel access outside reactor building and equipment qualification:

1. Airborne Source is twice system leakage rate form system integrity tests after corrective action has been taken plus design containment leak rate (2%/day).

2. GTS filter loading is based on (1) above.

IV. ESTIMATING INTEGRATED DOSES FOR EQUIPMENT QUALIFICATION

Estimated integrated exposures are based on the single highest surface dose rate in the area, either 1) piping or component surface for internal sources, or 2) wall surface for external sources. Credit has been taken for decay and actual time requirements for operability of equipment or other effects as applicable. Integrated doses are estimated by assuming dose rates are proportional to total energy release rates above 300 keV. Therefore, a multiplication factor is obtained which when multiplied by the dose rate at t = 0 will yield the total integrated dose. For the purposes of the short term review, the maximum exposure time considered shall be 30 days. Inasmuch as most organic materials (insulation, seals, etc.) are not affected by gamma doses below 10^5 rads safety related equipment receiving an integrated exposure less than 10^5 rads in six months are not considered further. An exception concerns solid state devices such as transistors, diodes, microprocessors, etc., which can fail at $5x10^2$ rad. Safety related instruments containing these are considered separately. Safety-related equipment receiving higher exposures are evaluated further to assure that the safety function of the equipment can be performed before the qualification limits on the equipment are exceeded.

V. ACCESSIBILITY CONSIDERATIONS FOR PERSONNEL EXPOSURE ESTIMATES

Any area which will or may require occupancy to permit an operator to aid in the mitigation of or recovery from an accident has been designated as a required access area.

It is necessary to consider not only the access area locations requiring personnel access but pathways for access must be identified as well. Operator actions required for safe shutdown or long term cooling will be identified to define the basis for identifying each required access area.

Access areas which will be evaluated include:

1. control room

2. sampling and monit[illegible] as per 2.1.8a and 2.1.8.b.

3. on-site technical s[illegible]t, temporary and permanent, center per 2.2.2b

4. on-site operational support center per 2.2.2c

1696 014

5. Radwaste building

6. Turbine building effluent exhaust monitors

In addition to these access areas both Category A and Category B systems will be reviewed to identify any required manual operations.

There may be some other systems where operation may be required. Pathways to the equipment and access will be considered.

Examples: 1) Portions of HVAC
2) Portions of Component Cooling Water System
3) Diesel Generator Operation
4) Motor Control Center Operation
5) System Control Panels
6) Switchgear
7) Instrument areas

VI. PERSONNEL RADIATION EXPOSURE GUIDANCE

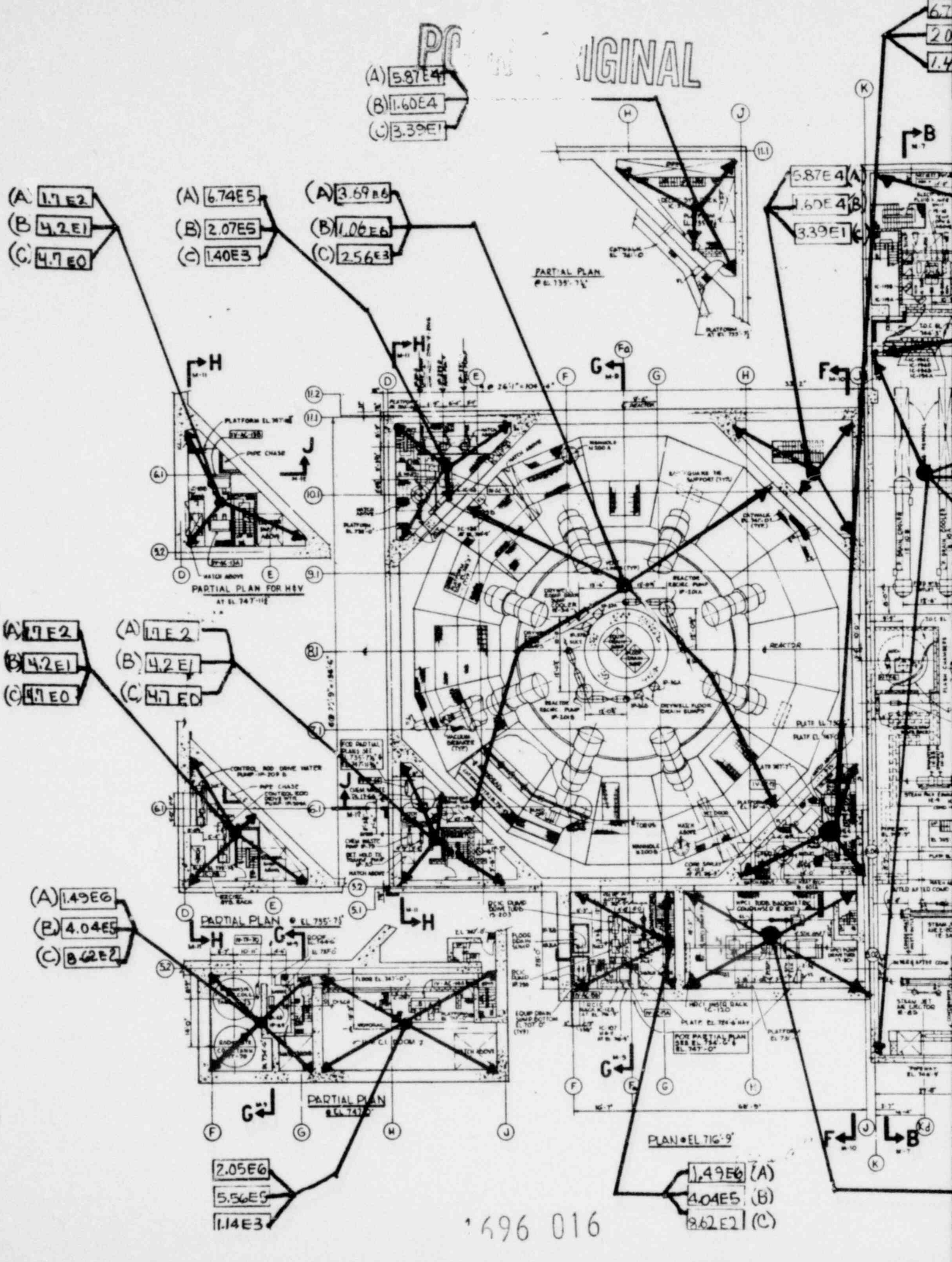
For short term post accident shielding reviews, designs will be evaluated based upon 10 CFR 50, Appendix A, GDC 19. The following radiation limit guidelines will be used to evaluate occupancy and accessibility of plant vital areas. Average dose rates may be used rather than maximum surface dose rates. However, if average dose rates are used contributions from all sources will be considered. Operator exposure should be evaluated using time and motion analysis.

A. Access areas requiring occupancy

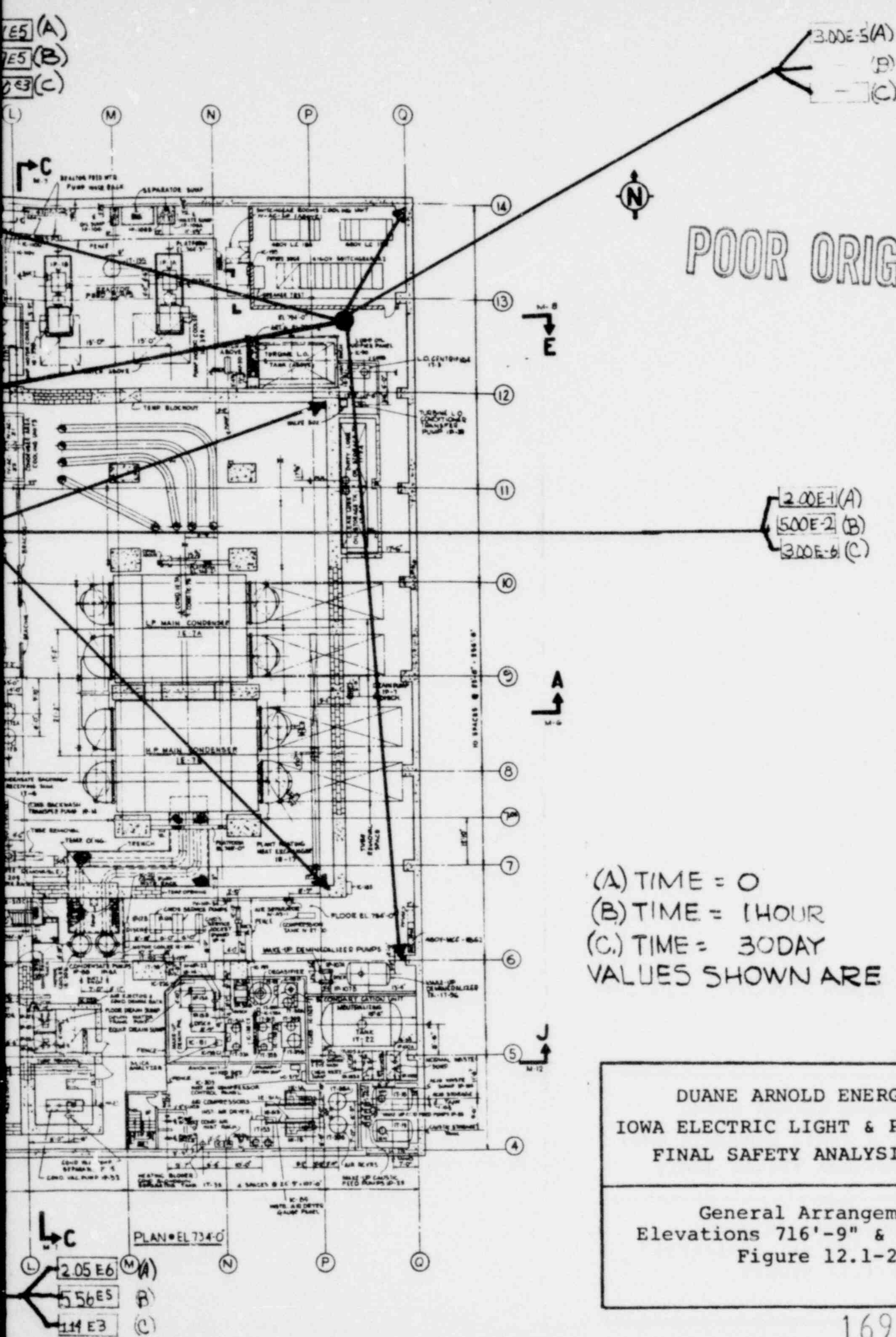
The short term review will verify that for vital areas such as control room and the on-site technical support center, the direct dose rate shall be less than 15 mr/hr at all times from all Category A and Category B systems.

B. Access areas requiring infrequent access or passageways to these access areas.

The review will be on the basis that for areas which may require access on a regular basis, but not continuous occupancy (such as radiochemical/chemical analysis labs, sample stations etc.) that the dose rate will not exceed the guidelines of GDC 19 (i.e., 5 rem total whole body dose). For dose rates greater than 100 mr/hr, occupancy time will be determined to insure that the integrated exposure for an operator action shall not exceed 5 rem. For dose rates less than 100 mr/hr, occupancy time determination is not required.



POOR ORIGINAL
(A) 5.87E4
(B) 1.60E4
(C) 3.39E1
(A) 1.7E2
(B) 4.2E1
(C) 4.7E0
(A) 6.74E5
(B) 2.07E5
(C) 1.40E3
(A) 3.69E6
(B) 1.06E6
(C) 2.56E3
PARTIAL PLAN
PARTIAL PLAN FOR H&V
REACTOR
(A) 1.7E2
(B) 4.2E1
(C) 4.7E0
(A) 1.7E2
(B) 4.2E1
(C) 4.7E0
5.87E4 (A)
1.60E4 (B)
3.39E1 (C)
(A) 1.49E6
(B) 4.04E5
(C) 8.62E2
PARTIAL PLAN @ EL 735'-7½
PARTIAL PLAN
PLAN @ EL 716'-9"
2.05E6
5.56E5
1.14E3
1.49E6 (A)
4.04E5 (B)
8.62E2 (C)
1696 016



POOR ORIGINAL

2/19/79

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General Arrangement
Elevations 716'-9" & 734'-0"
Figure 12.1-2

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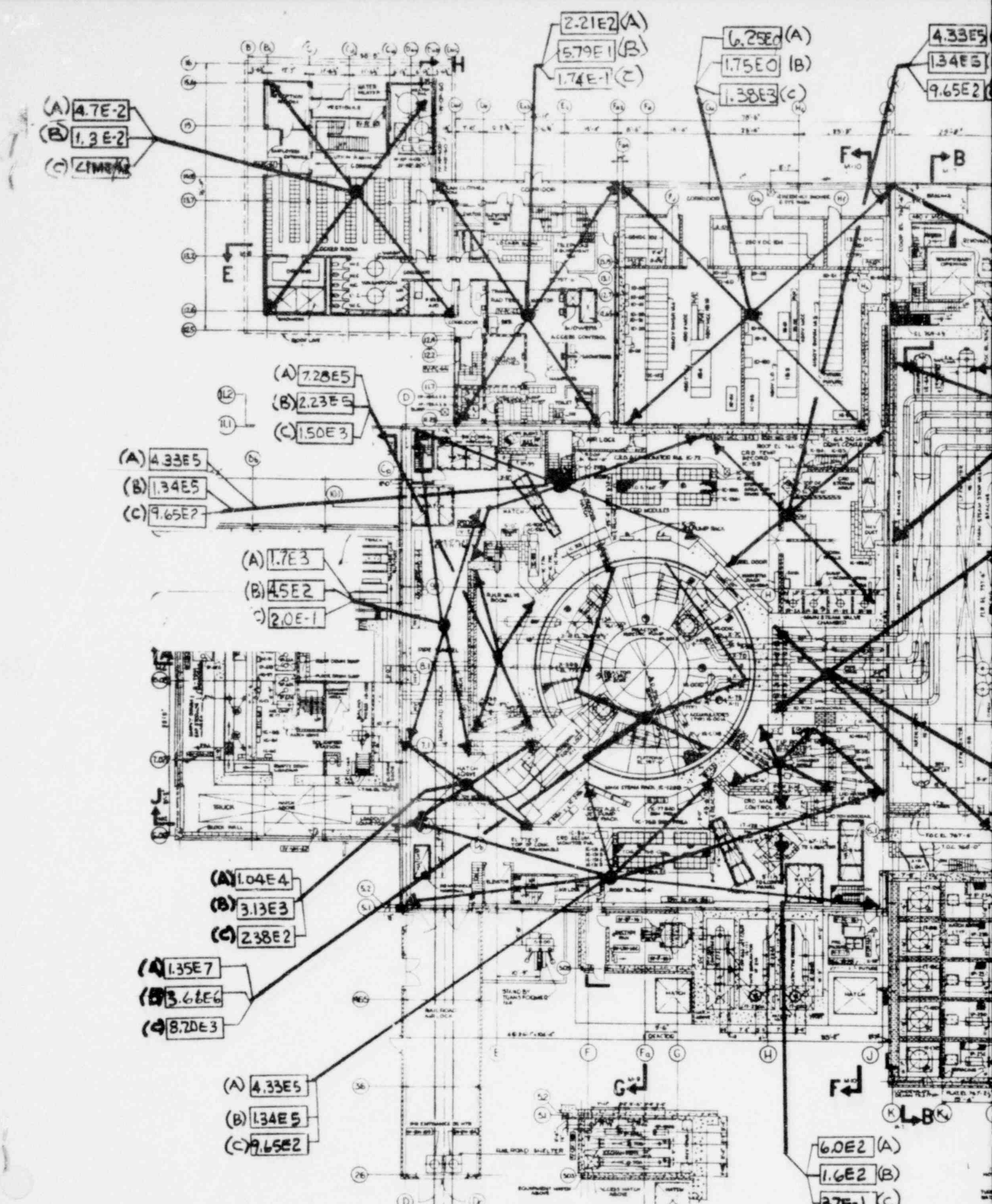
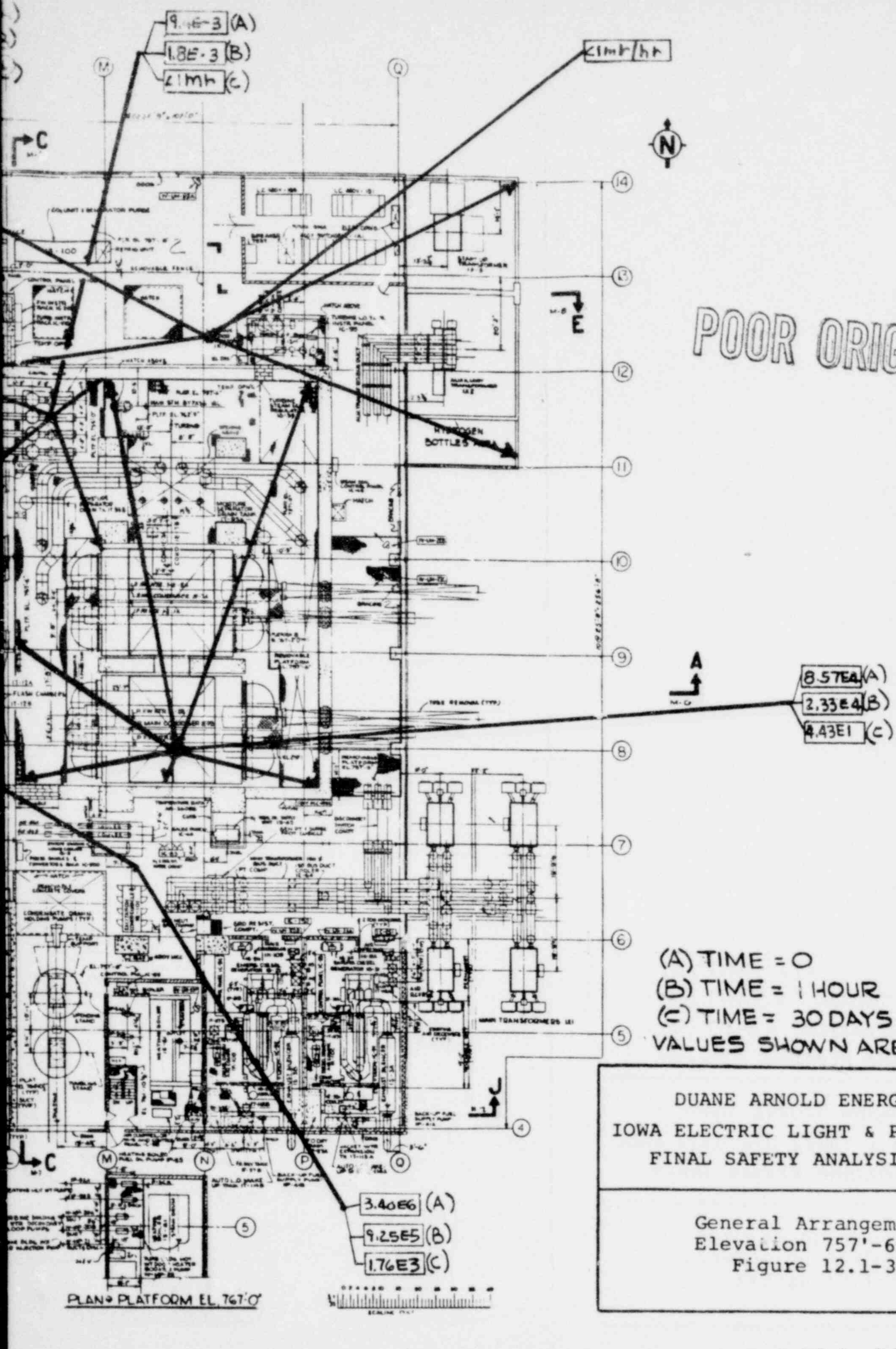
1696 017



(A) 2.21E2
(B) 5.79E1
(C) 1.74E-1
(A) 6.25E0
(B) 1.75E0
(C) 1.38E3
4.33E5
1.34E5
9.65E2
(A) 4.7E-2
(B) 1.3E-2
(C)
(A) 7.28E5
(B) 2.23E5
(C) 1.50E3
(A) 4.33E5
(B) 1.34E5
(C) 9.65E2
(A) 1.7E3
(B) 4.5E2
(C) 2.0E-1
(A) 1.04E4
(B) 3.13E3
(C) 2.38E2
(A) 1.35E7
(B) 3.66E6
(C) 8.20E3
(A) 4.33E5
(B) 1.34E5
(C) 9.65E2
6.0E2 (A)
1.6E2 (B)
3.7E-1 (C)
E
F
B
G
J
H
K

1696 018



POOR ORIGINAL

12/19/79

(A) TIME = 0
(B) TIME = 1 HOUR
(C) TIME = 30 DAYS
VALUES SHOWN ARE Rem/Hr

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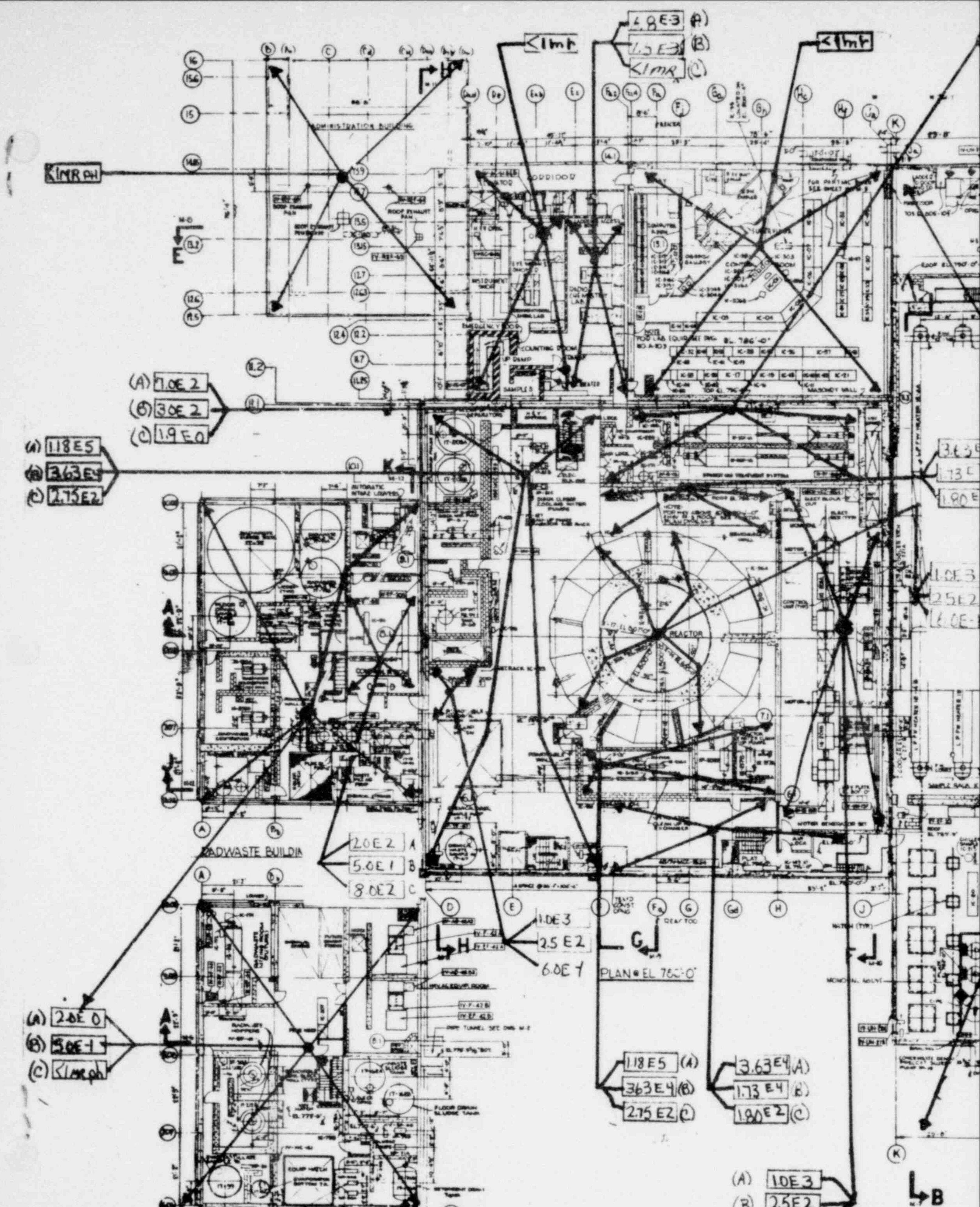
General Arrangement
Elevation 757'-6"
Figure 12.1-3

12/74



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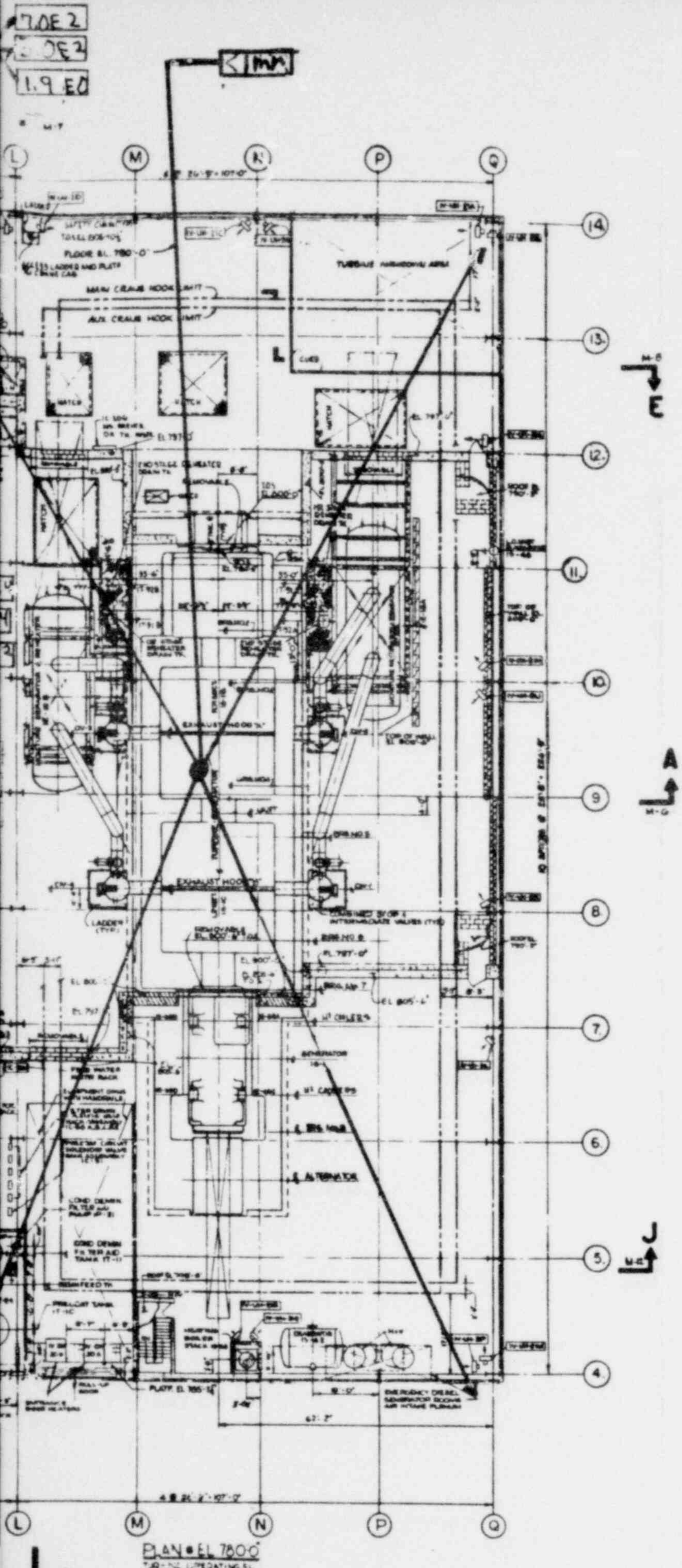

(A) TIME = 0
(B) TIME = 1 HOUR
(C) TIME = 30 DAYS
VALUES SHOWN ARE Rem/hr

12/26/79

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IOWA ELECTRIC LIGHT & POWER COMPANY
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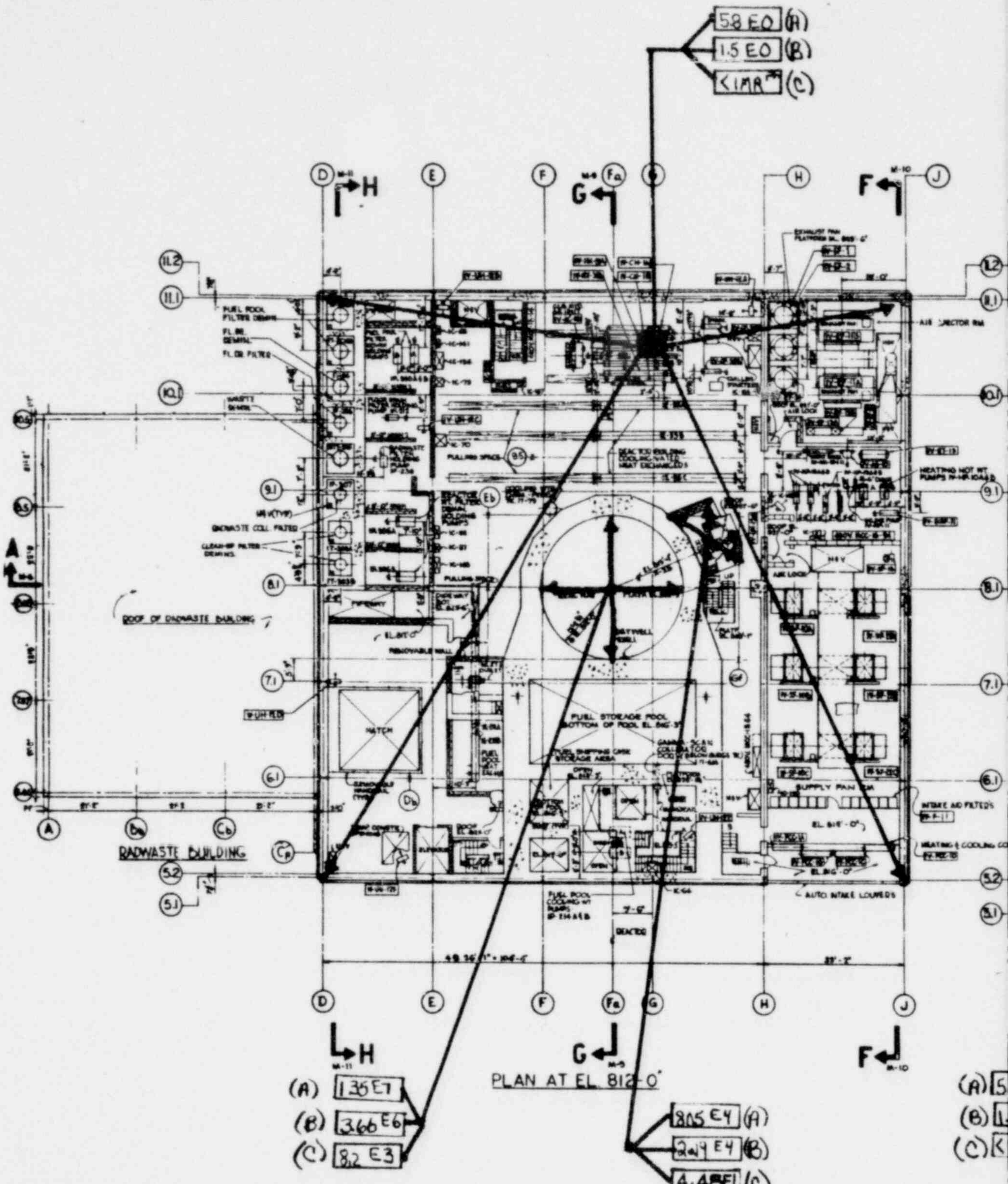
General Arrangement
Elevations 780-0"
786'-0" and 773'-6"
Figure 12.1-4 12/74

1696 021

POOR ORIGINAL



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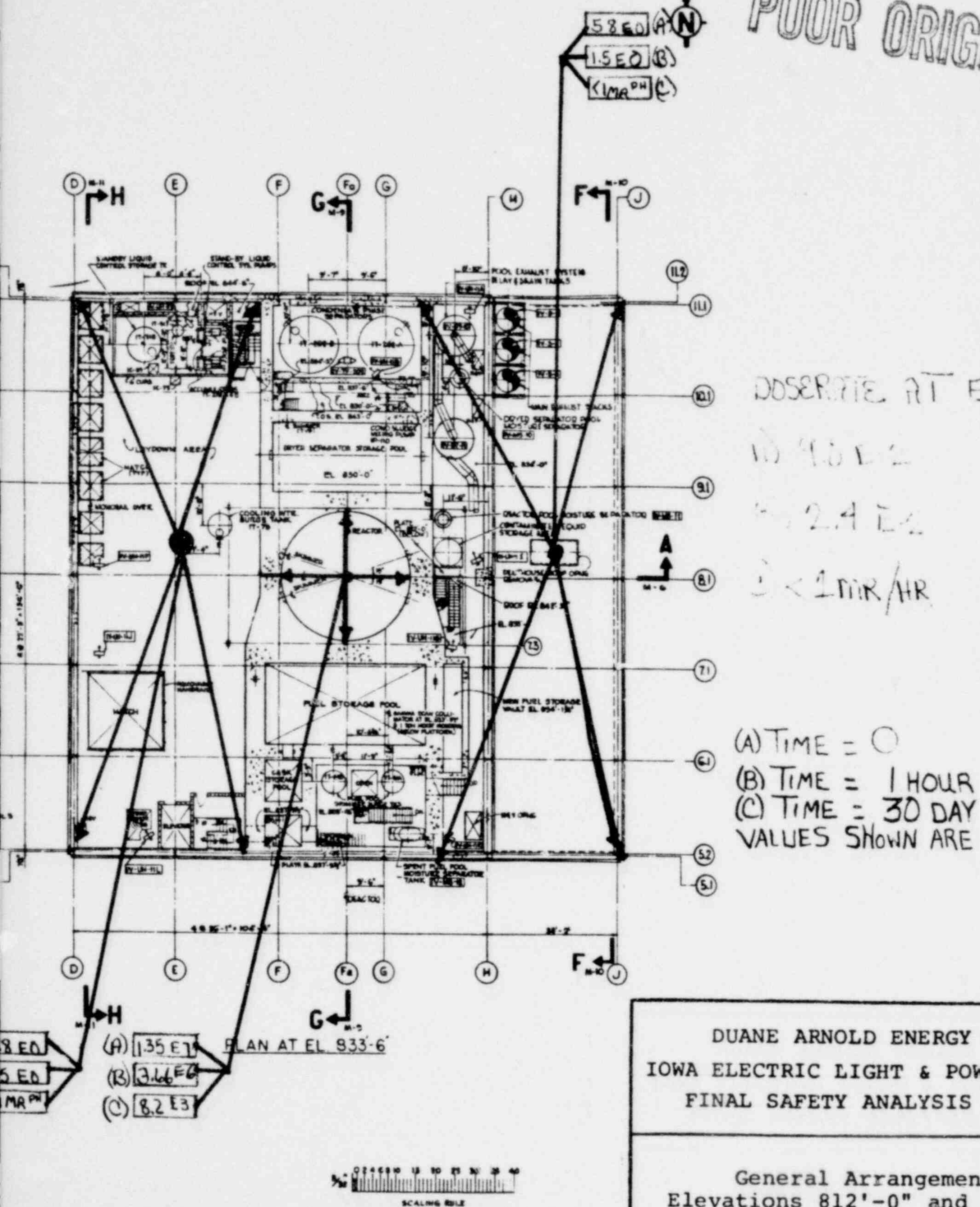

12/19/79

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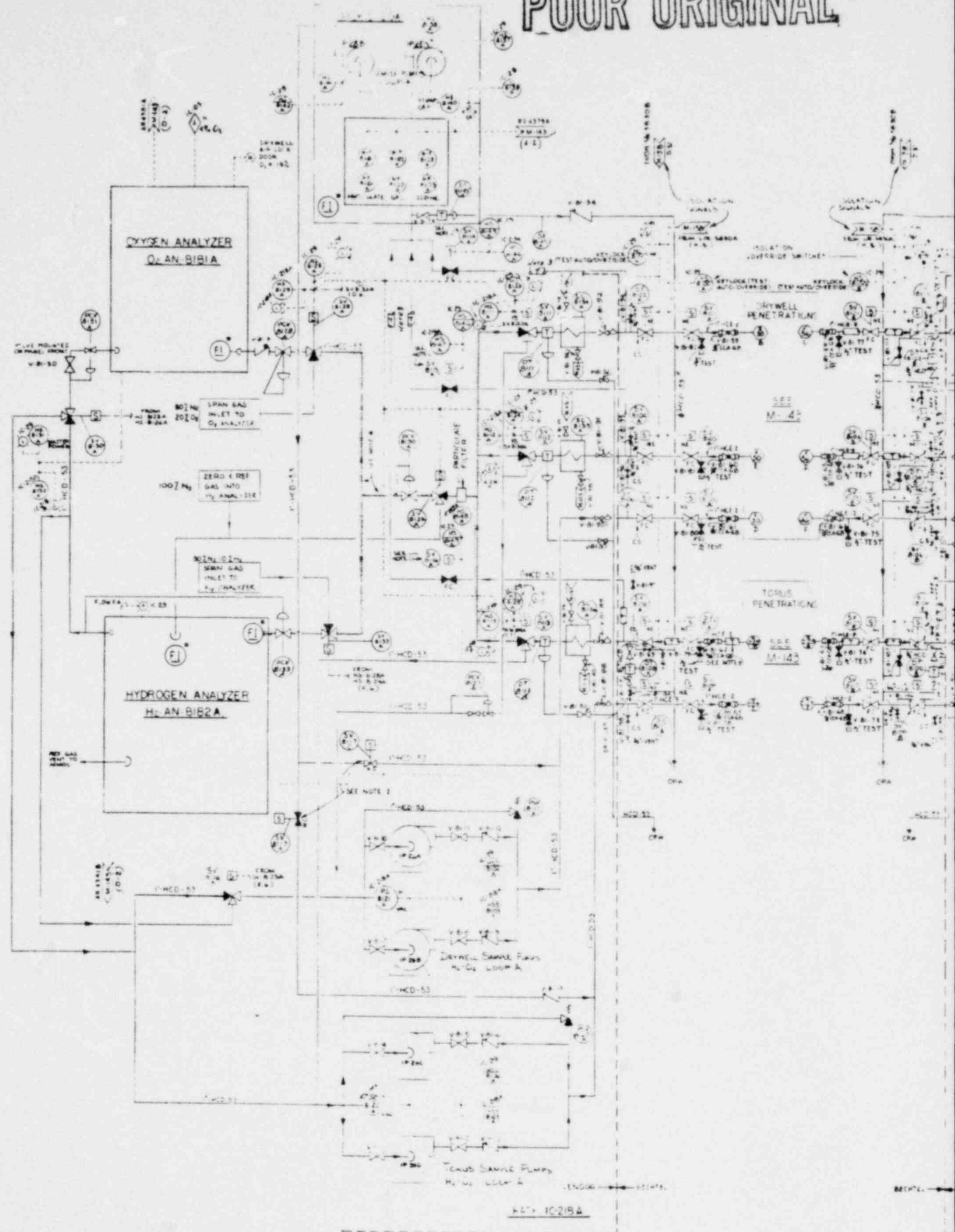
General Arrangement
Elevations 812'-0" and 833'-6"
Figure 12.1-5

12/74

1696 023

POOR ORIGINAL



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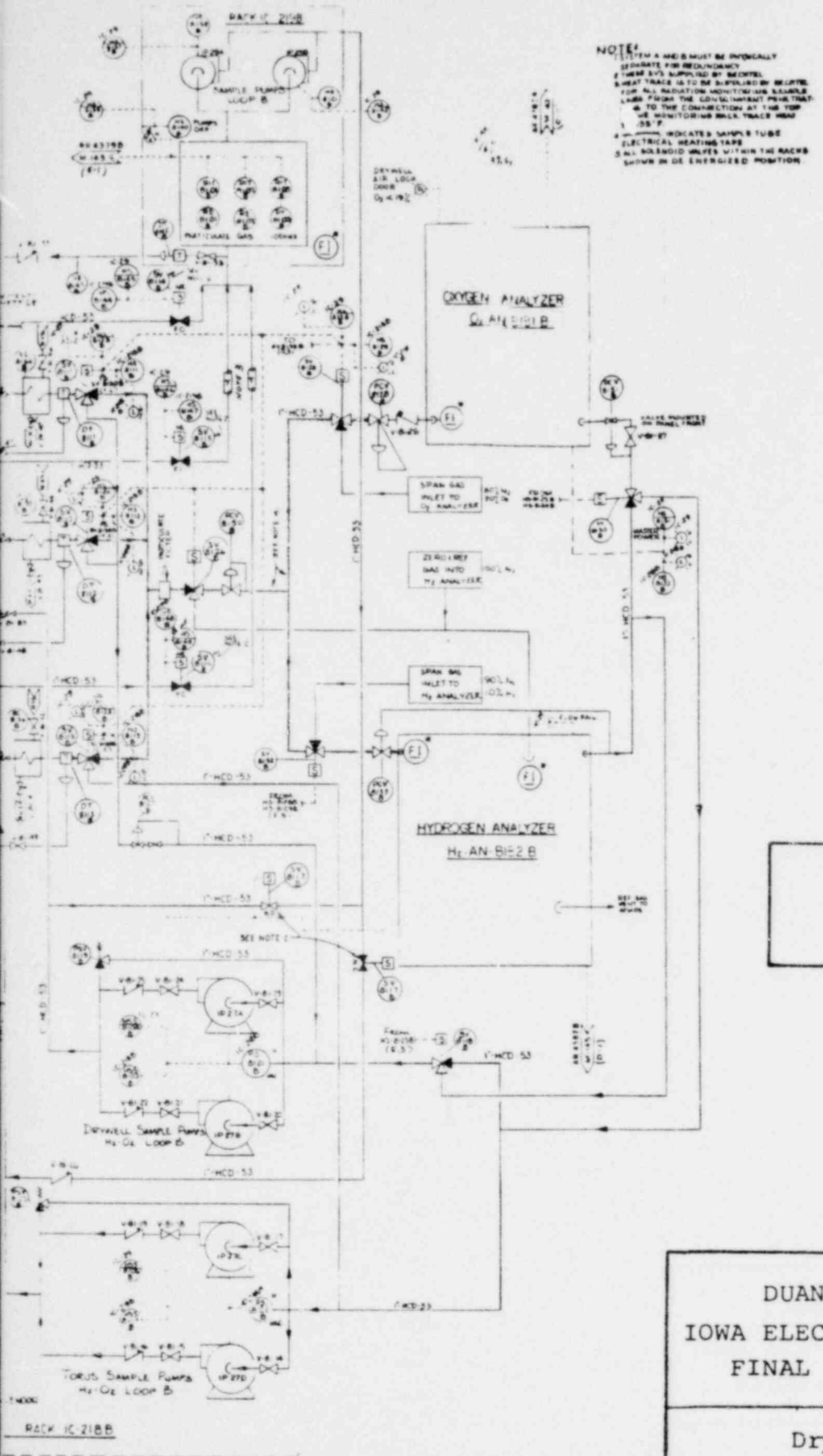

POOR ORIGINAL

ALL PIPING & EQUIPMENT
ARE SOURCE "A" CONTAINMENT
AIRBORNE

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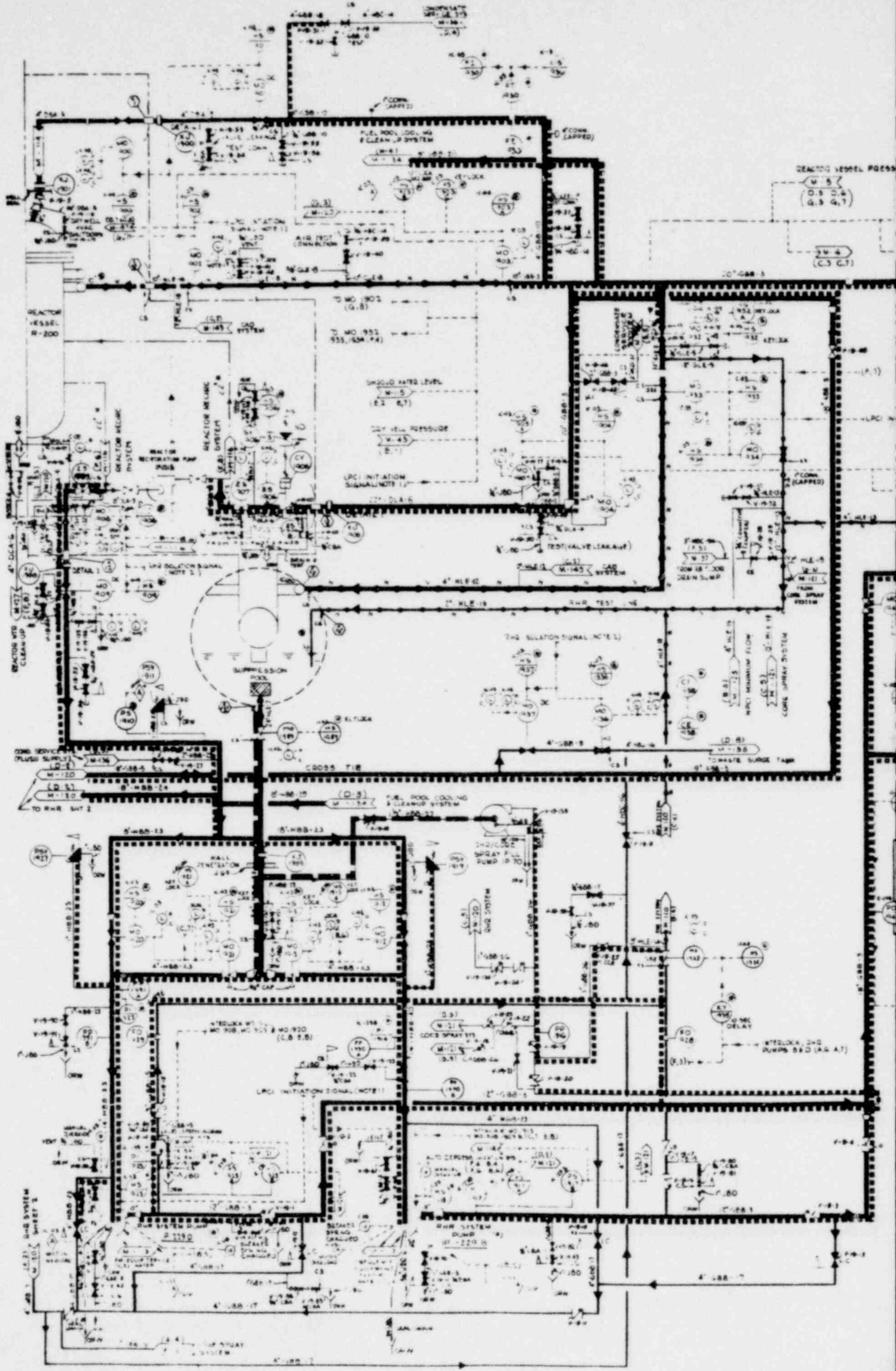
Drywell Atmospheric
Monitoring System
FIGURE 7-G7.3-1

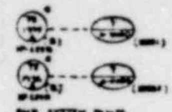
12/74

1696 025



POOR ORIGINAL

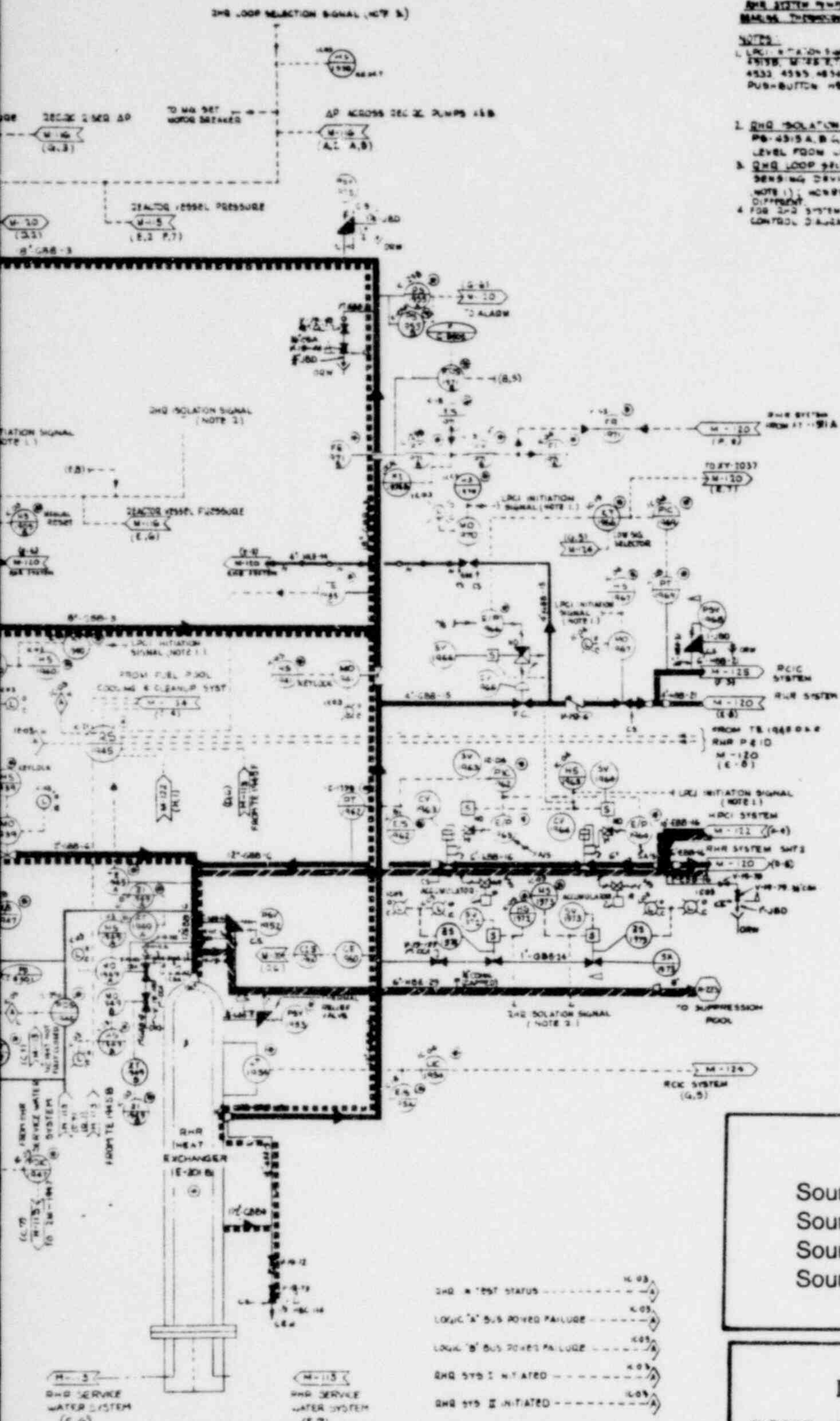
1696 026



RHR SYSTEM PUMPS
BEARING THERMOCOUPLES

NOTES:

1. LPCI INITIATION SIGNAL - DRYWELL HIGH PRESSURE PS 4310B, 4611B, 4312B, 4313B, M-125(F,7) AND/OR REACTOR LOW WATER LEVEL FROM LS 4531, 4532, 4533, 4534 M-115(F,2 F,7). SIGNAL MAY BE RESET WITH PUSH-BUTTON HS 1361 LOCATED ON PANEL 1C-03
2. RHR ISOLATION SIGNAL - DRYWELL HIGH PRESSURE FROM PS-4315A, B, C, D, M-143(E,7) AND/OR REACTOR LOW WATER LEVEL FROM LIS 4535, 4536, 4537, 4538, M-118 (G,2 G,7)
3. RHR LOOP SELECTION SIGNAL - RECEIVED FROM SAME SENSING DEVICES THAT SEND LPCI INITIATION SIGNALS (NOTE 1); HOWEVER REACTOR LOW WATER LEVEL SET POINTS ARE DIFFERENT.
4. FOR RHR SYSTEM LOGIC, SEE G.E. RHR FUNCTIONAL CONTROL DIAGRAM 729E630CA



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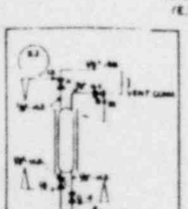

DETAIL 1
(TYPICAL EXPANSION JOINT TEST CONNECTIONS)

TABLE A EXPANSION JOINT TEST VALVES

EXPANSION JOINT NO.	VALVE NO. A	VALVE NO. B	VALVE NO. C	VALVE NO. D	VALVE NO. E
EJ-1900	V-19-105	V-19-104	V-19-10[illegible]	V-19-106	V-19-107
EJ-1906	V-19-108	V-19-109	V-19-110	V-19-111	V-19-112
EJ-1908	V-19-113	V-19-114	V-19-115	V-19-116	V-19-117

LEGEND

Source A - Containment Airborne	·▬·▬·
Source B - Reactor Liquid	▪▪▪▪▪▪
Source C - Suppression Pool Liquid	▬ ▬ ▬
Source D - Reactor Steam	▬▬▬

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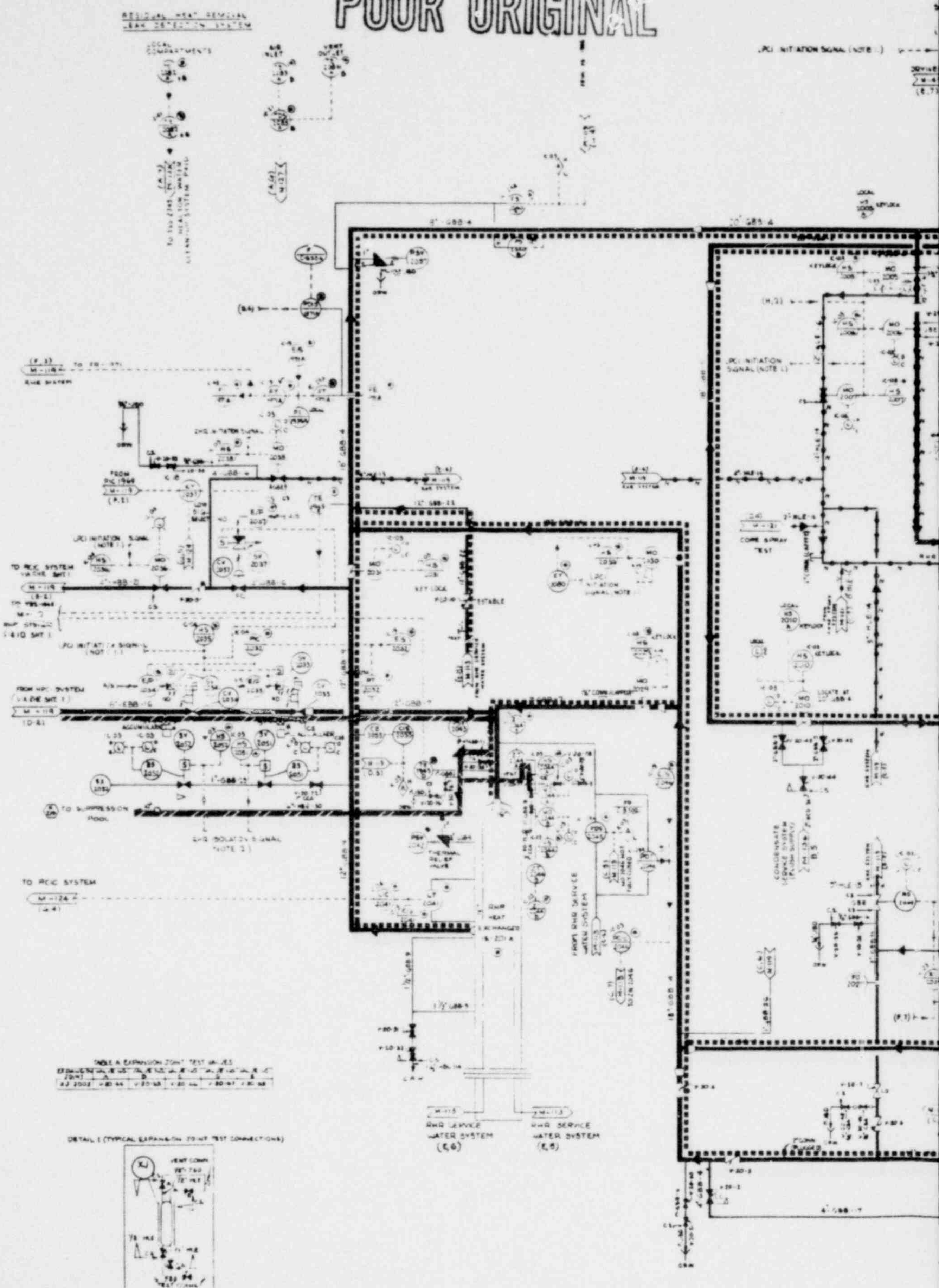
Residual Heat Removal System
P&ID
FIGURE 4.8-2 SHEET 1 12/74

1696 027

POOR ORIGINAL



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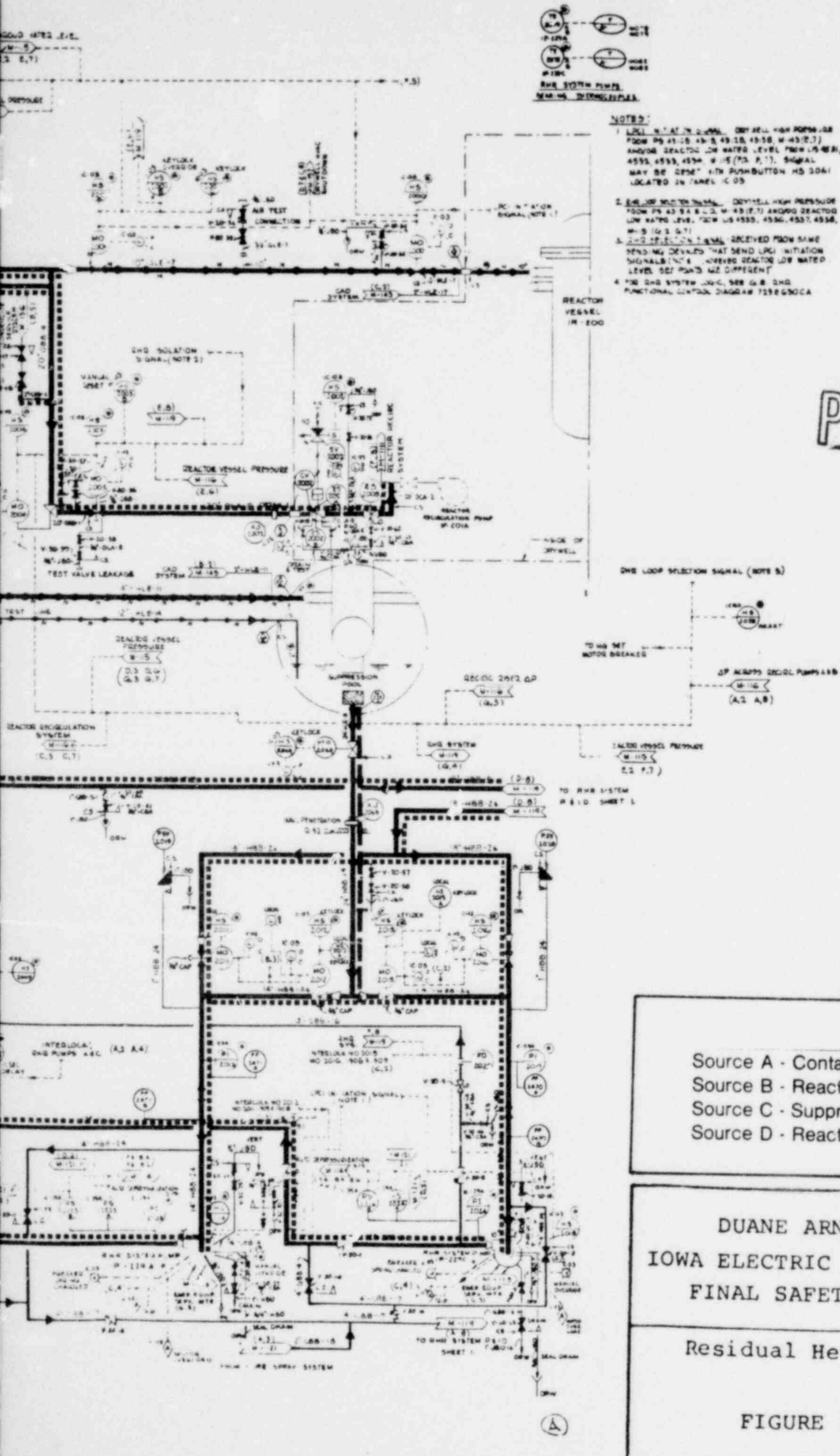

POOR ORIGINAL

LEGEND

Source A - Containment Airborne	— - — -
Source B - Reactor Liquid	··········
Source C - Suppression Pool Liquid	— — —
Source D - Reactor Steam	═══

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Residual Heat Removal System
P&ID

FIGURE 4.8-2 SHEET 2

12/74

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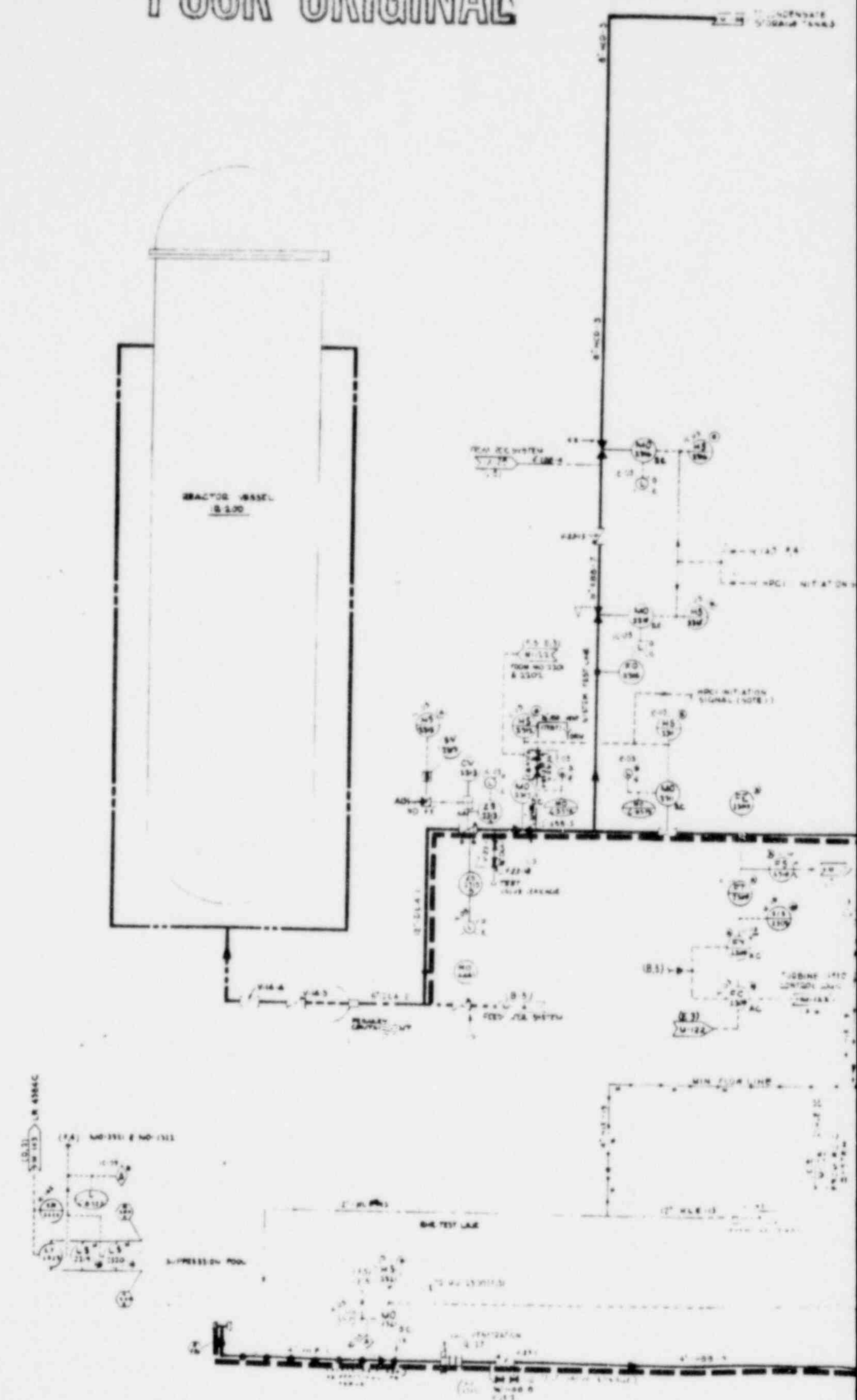
POOR ORIGINAL



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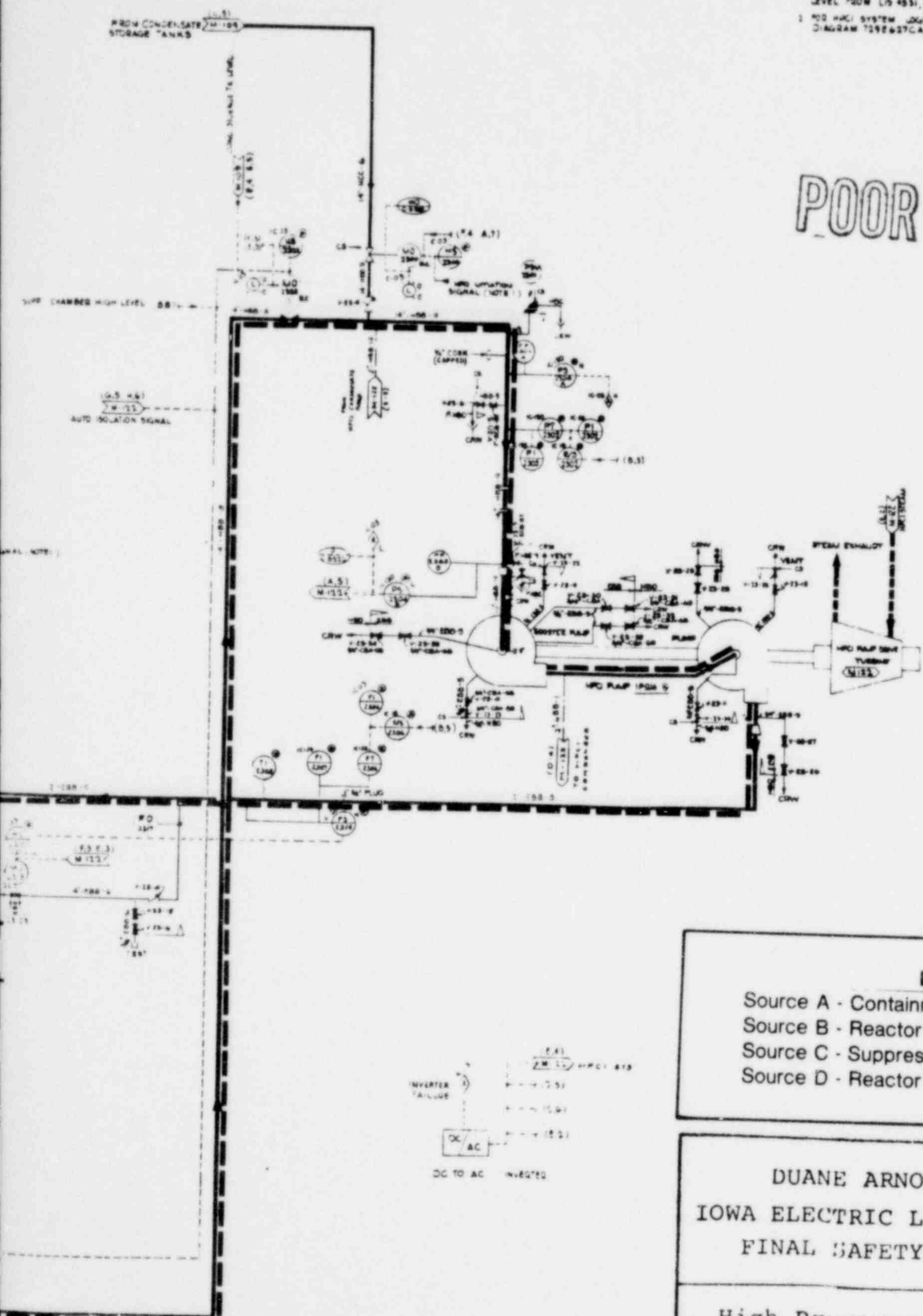
VOL III

NOTES:

1. HPCI INITIATION SIGNAL DRYWELL HIGH PRESSURE FROM PS 4301B, 4311B, 4312B, 4313B, M-45(E,7) AND/OR REACTOR LOW WATER LEVEL FROM LIS 4531, 4532, 4533, 4534, M-115 (F,5 F,6)
2. FOR HPCI SYSTEM LOGIC SEE G.E. FUNCTIONAL CONTROL DIAGRAM 729E627CA



LEGEND

Source A - Containment Airborne

Source B - Reactor Liquid

Source C - Suppression Pool Liquid

Source D - Reactor Steam

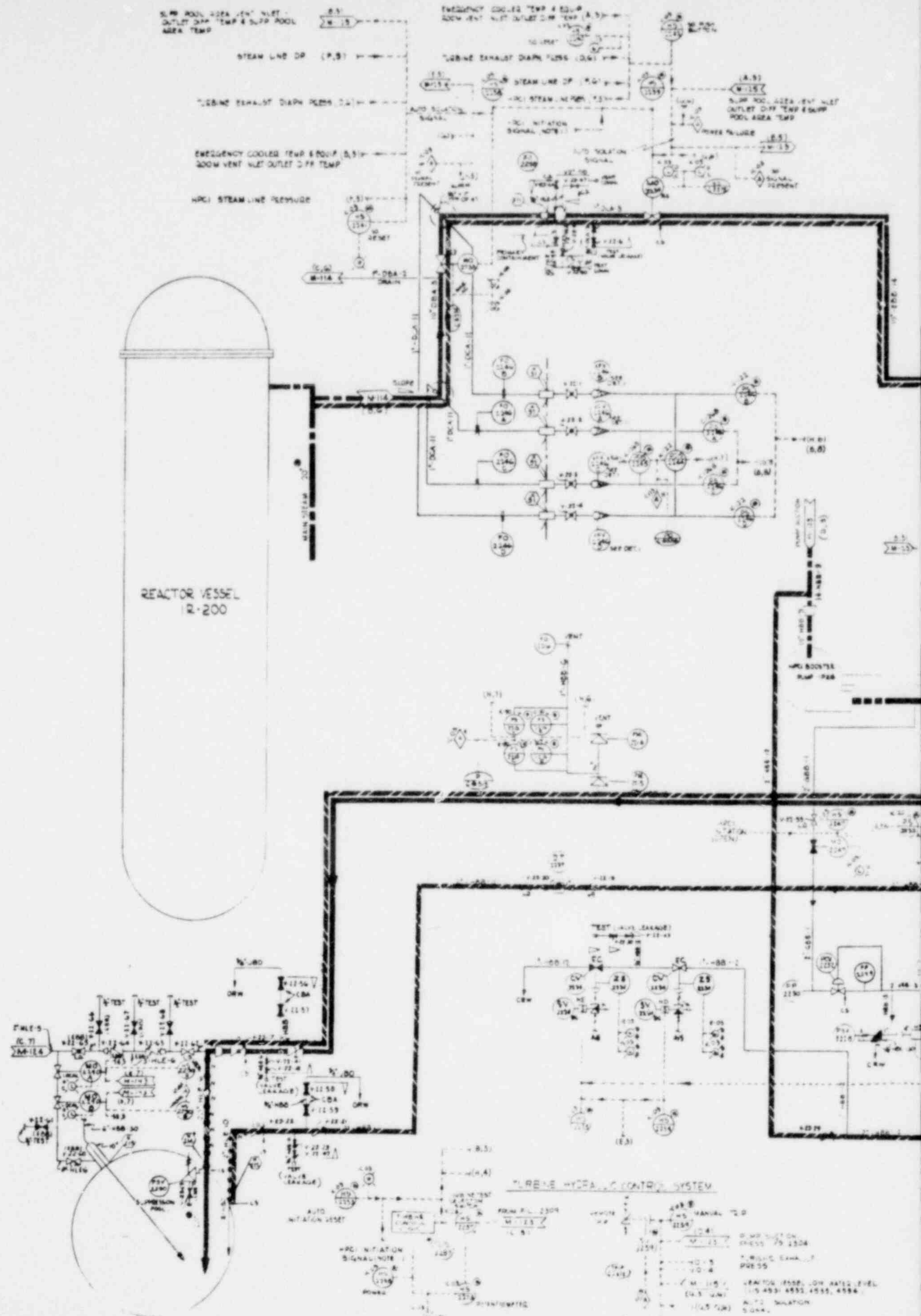
DUANE ARNOLD ENERGY CENTER
IOWA ELECTRIC LIGHT & POWER COMPANY
FINAL SAFETY ANALYSIS REPORT

High Pressure Coolant Injection System - P&ID

FIGURE 7.4-2 SHEET 2

12/74

1696 031

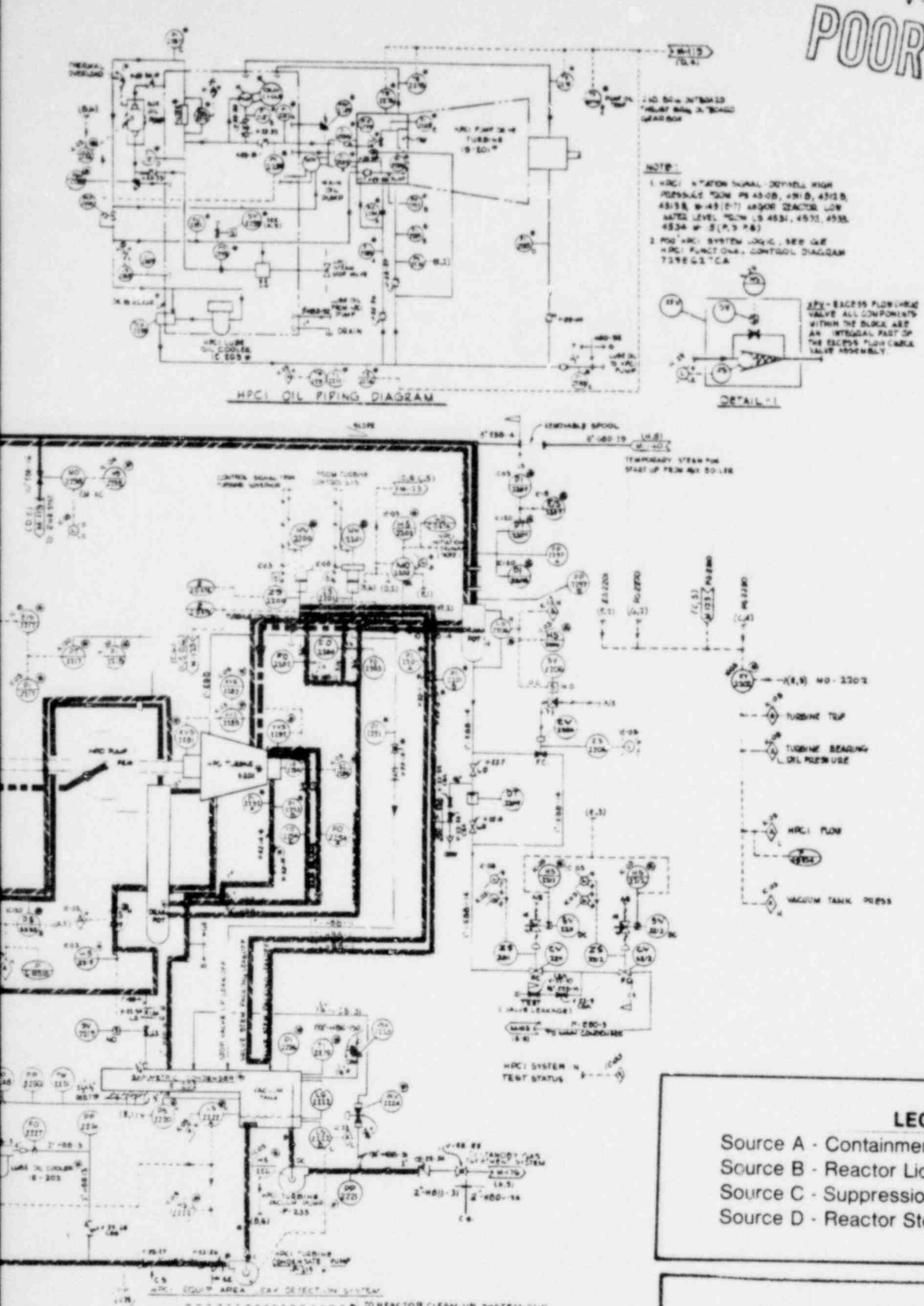


POOR ORIGINAL

1696 032

VOL. III

POOR ORIGINAL



LEGEND

Source A - Containment Airborne	
Source B - Reactor Liquid	
Source C - Suppression Pool Liquid	
Source D - Reactor Steam	

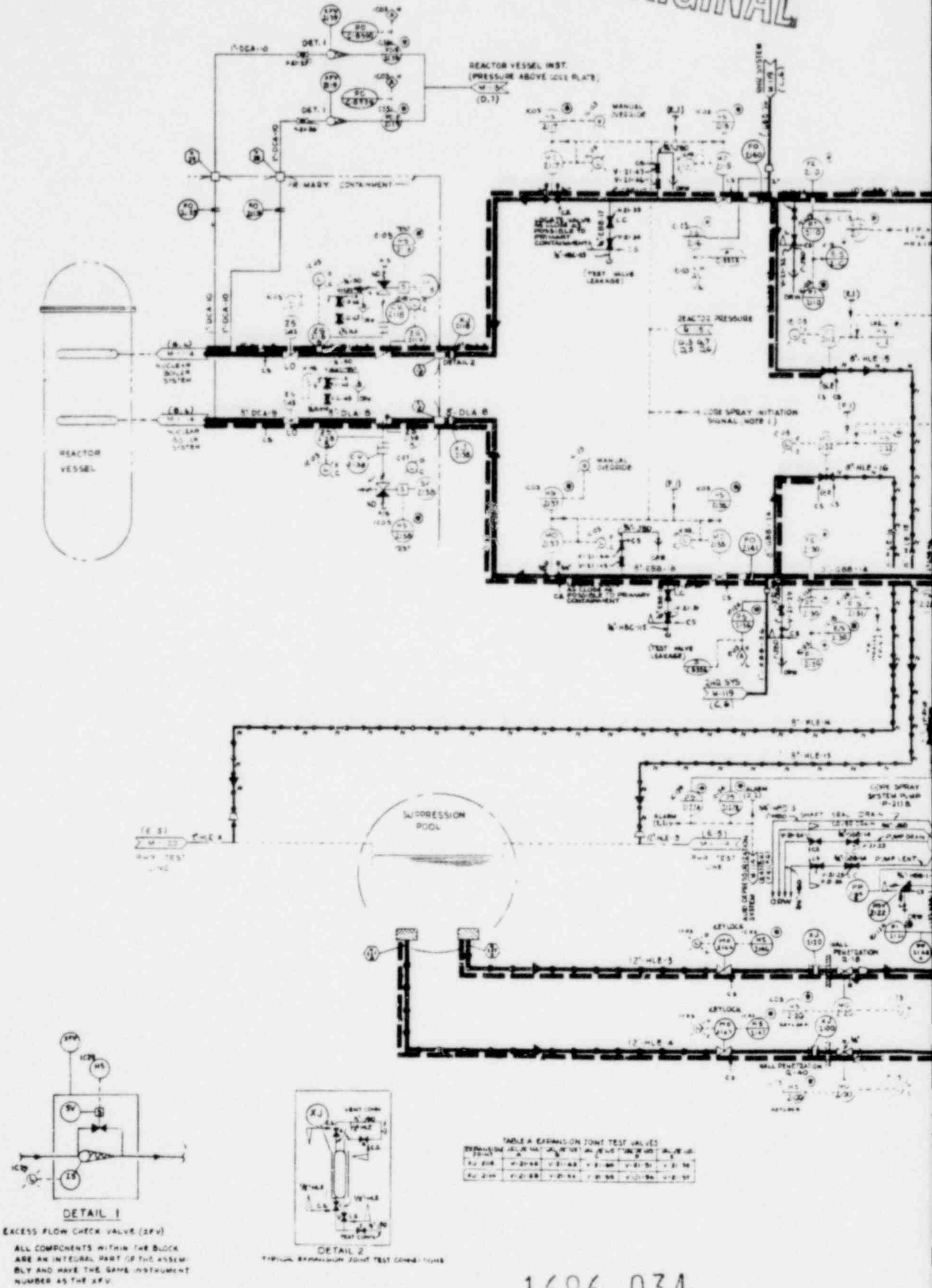
DUANE ARNOLD ENERGY CENTER
IOWA ELECTRIC LIGHT & POWER COMPANY
FINAL SAFETY ANALYSIS REPORT

High Pressure Coolant Injection System - P&ID

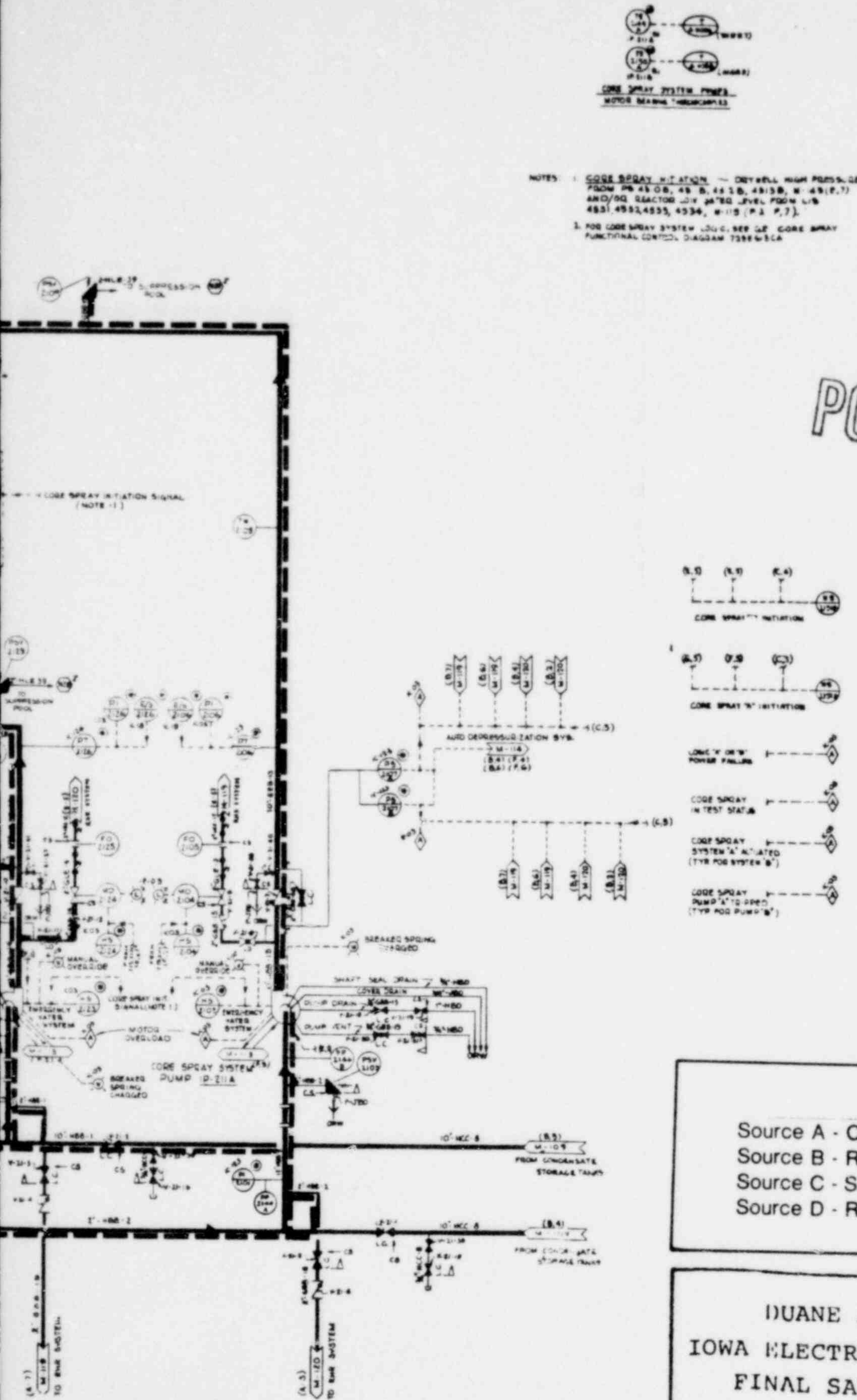
FIGURE 7.4-2 SHEET 1

12/74

1696 033


REACTOR VESSEL INST. (PRESSURE ABOVE CORE PLATE)
PRIMARY CONTAINMENT
REACTOR VESSEL
NUCLEAR BOILER SYSTEM
REACTOR PRESSURE
CORE SPRAY INITIATION SIGNAL (NOTE 1)
MANUAL OVERRIDE
TEST VALVE LEAKAGE
SUPPRESSION POOL
RHR TEST LINE
CORE SPRAY SYSTEM PUMP P-211B
SHAFT SEAL DRAIN
KEYLOCK
WALL PENETRATION
DETAIL 1
EXCESS FLOW CHECK VALVE (XFV)
ALL COMPONENTS WITHIN THE BLOCK ARE AN INTEGRAL PART OF THE ASSEMBLY AND HAVE THE SAME INSTRUMENT NUMBER AS THE XFV.
DETAIL 2
TYPICAL EXPANSION JOINT TEST CONNECTIONS
TABLE A EXPANSION JOINT TEST VALVES

1696 034



LEGEND

Source A - Containment Airborne	·—·—·—
Source B - Reactor Liquid	··········
Source C - Suppression Pool Liquid	— — —
Source D - Reactor Steam	———

DUANE ARNOLD ENERGY CENTER
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Core Spray System
P&ID
FIGURE 7.4-5 12/74

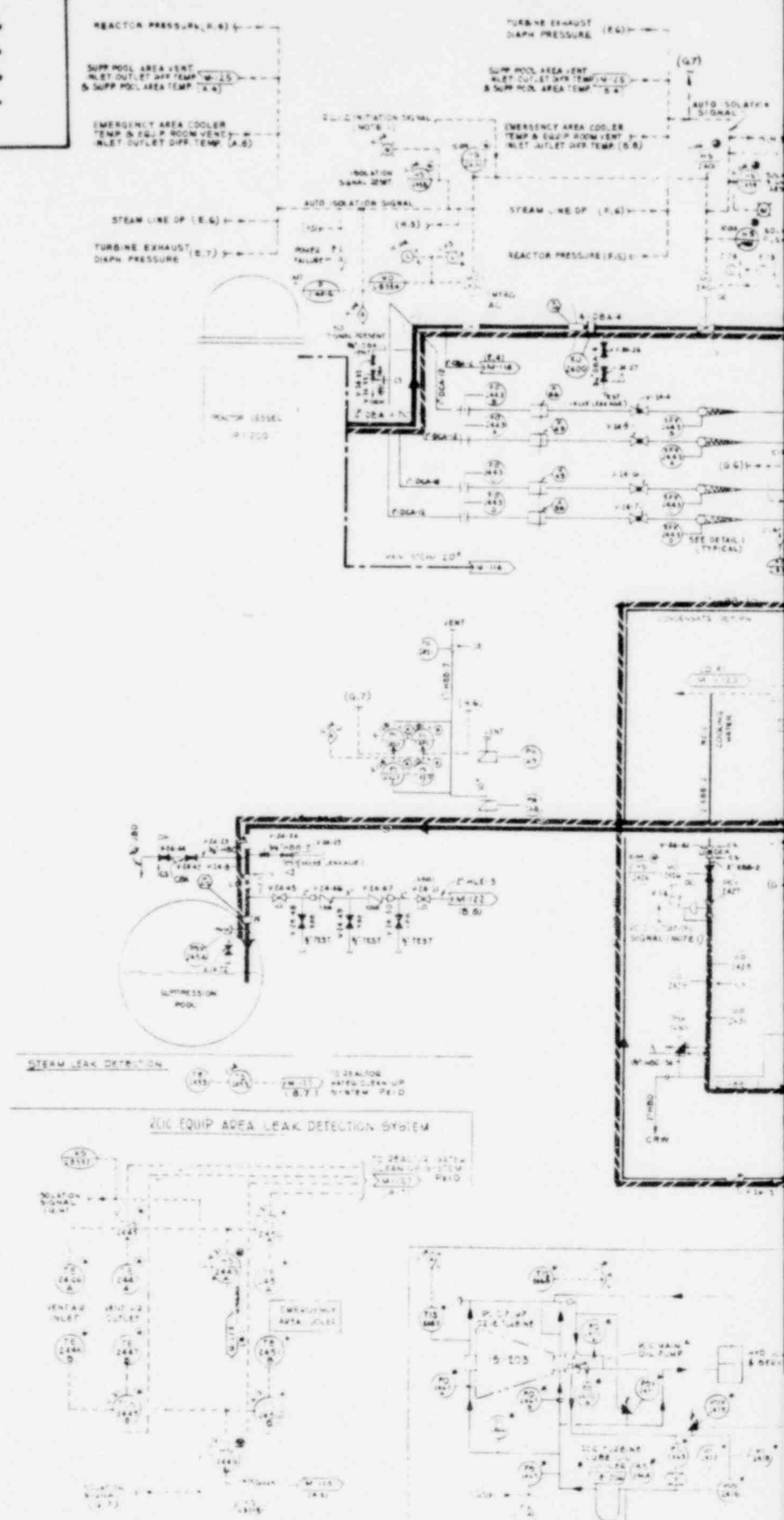
LEGEND

Source A - Containment Airborne

Source B - Reactor Liquid

Source C - Suppression Pool Liquid

Source D - Reactor Steam



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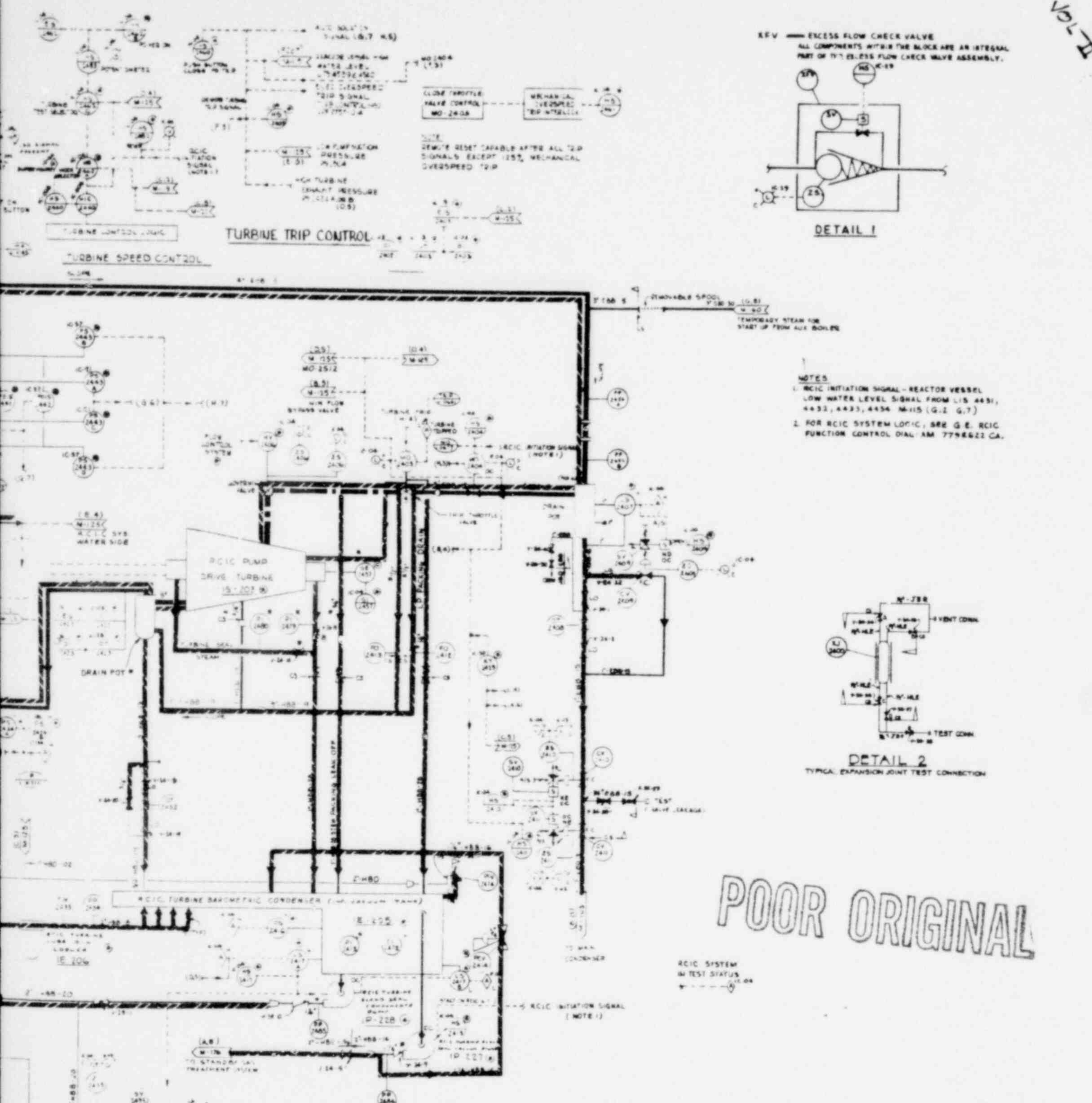

DUANE ARNOLD ENERGY CENTER
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Reactor Core Isolation Cooling System - P&ID
FIGURE 4.7-1 SHEET 1

12/74

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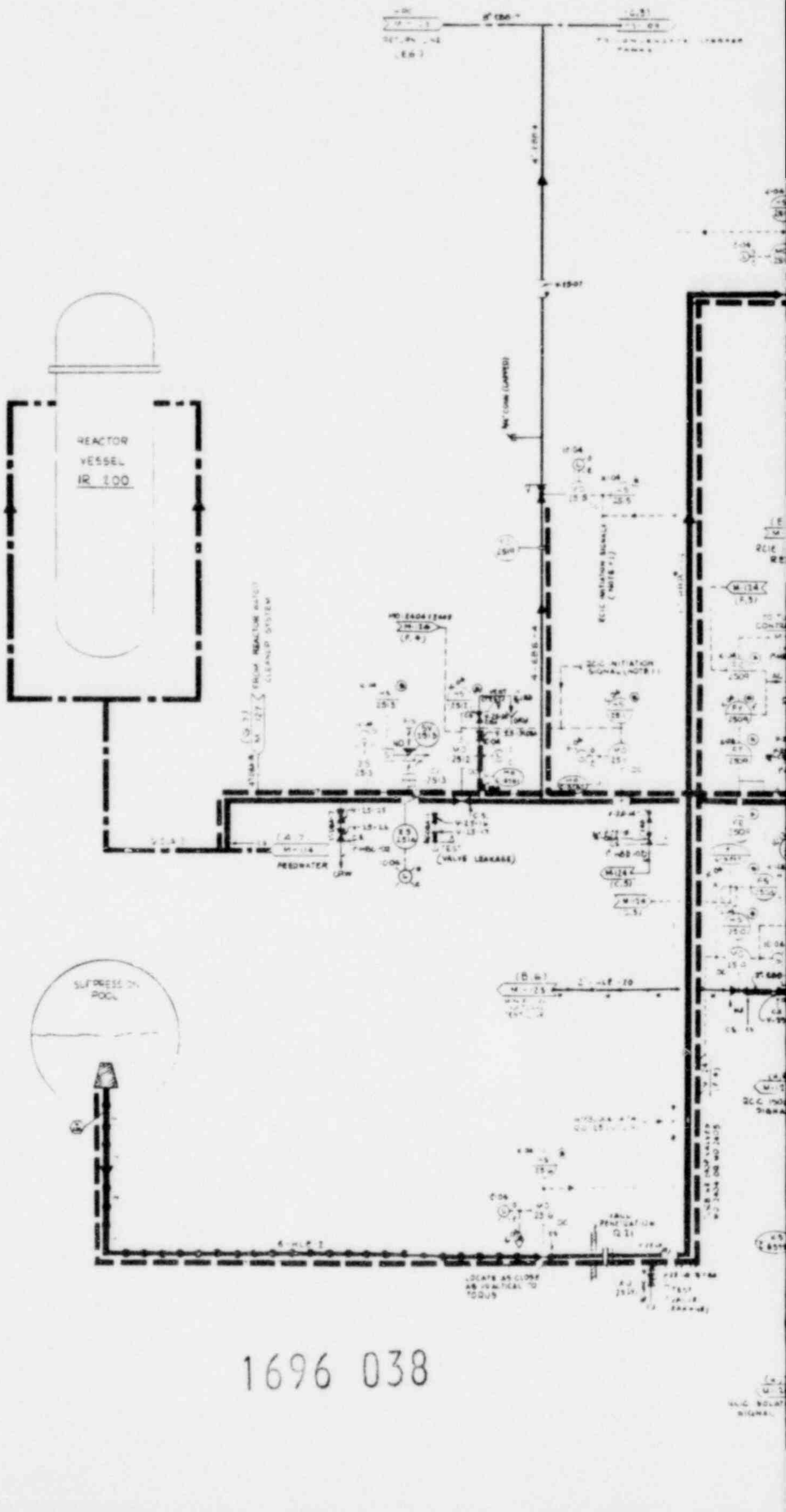
POOR ORIGINAL



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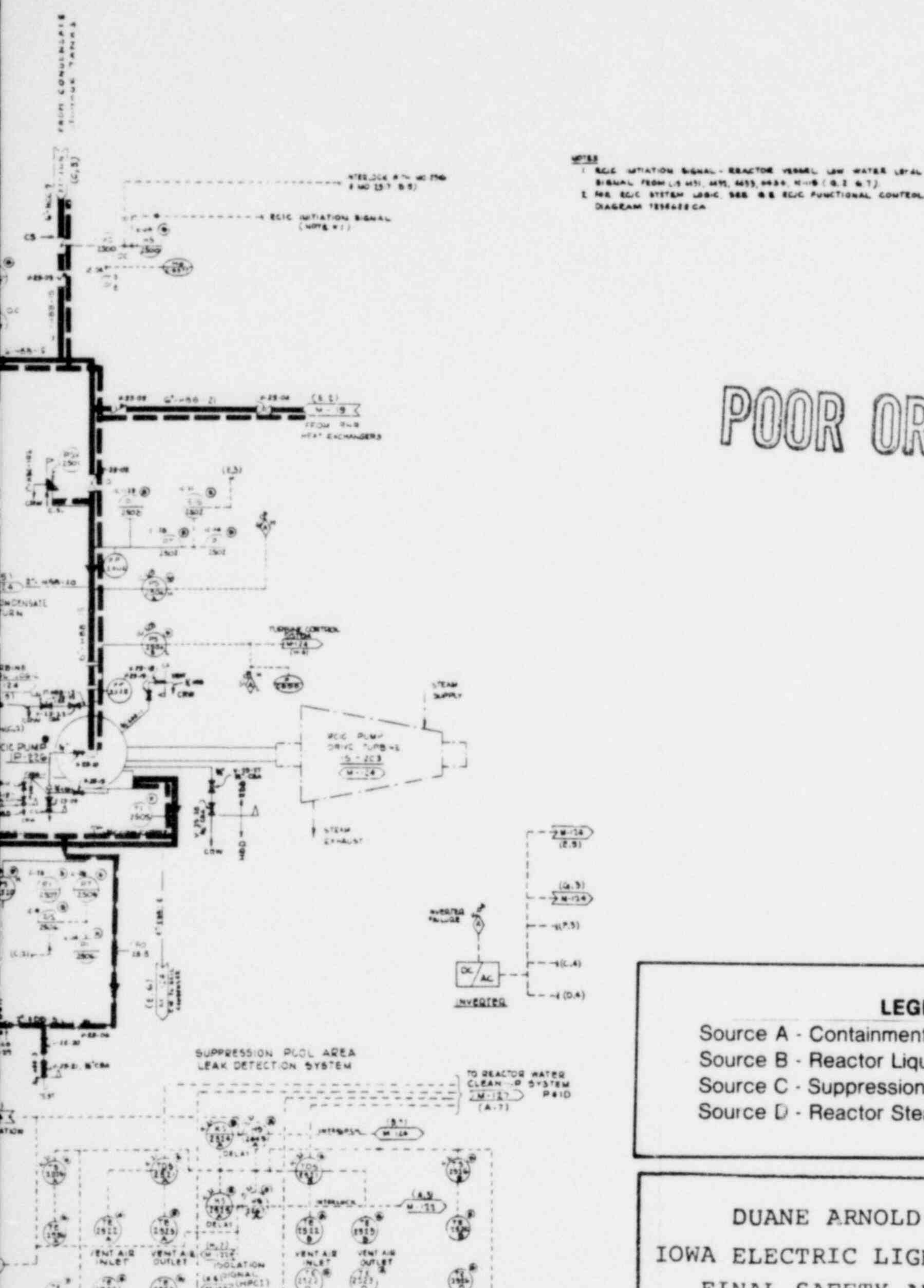
VOL II



POOR ORIGINAL

LEGEND

- Source A - Containment Airborne
- Source B - Reactor Liquid
- Source C - Suppression Pool Liquid
- Source D - Reactor Steam

DUANE ARNOLD ENERGY CENTER
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Reactor Core Isolation
Cooling System - P&ID
FIGURE 4.7-1 SHEET 2

12/74

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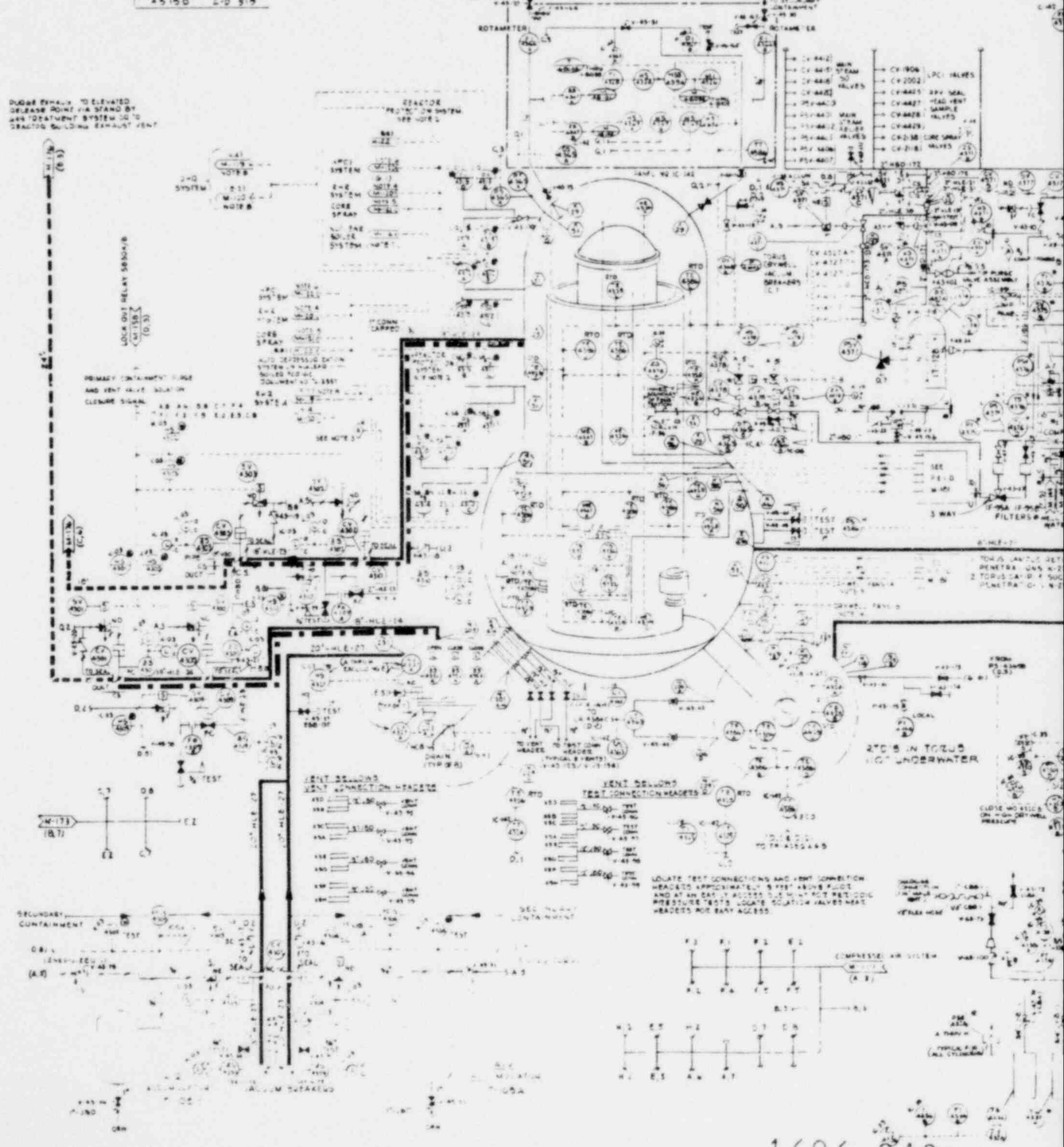
TABLE I

COMPUTER POINTS FOR PRIMARY CONTAINMENT PRESSURE SWITCHES

SWITCH	POINT
43:0A	6-B526
43:0B	6-B507
43:0C	6-B512
43:1A	6-B514
43:1B	6-B521
43:1C	6-B546
43:2A	6-B550
43:2B	6-B574
43:2C	6-B591
43:3A	6-B592
43:3B	6-B593
43:3C	6-B594
43:5A	6-D512
43:5B	6-D513
43:5C	6-D514
43:5D	6-D515

NOTES:-

1. CONTAINMENT ATMOSPHERIC DILUTION SYSTEM IS A SEISMIC I DESIGN
2. REACTOR PROTECTION SYSTEM [illegible] DOCUMENT [illegible] CONTAINMENT [illegible] ISOLATION FUNCTIONS [illegible] NUCLEAR BOILER [illegible] DOCUMENT [illegible] 557
3. CONTAINMENT HIGH PRESSURE APPROACH ALARM AT 5 PSIG
4. LPCI INITIATION SIGNAL & 2HD SELECTION SIGNAL SEE M-120 NOTES 4 & 5
5. CORE SPRAY INITIATION SEE M-120 NOTE 1
6. HPCI INITIATION SEE M-122 NOTE 1
7. AUTO DEPRESSURIZATION SYSTEM ON NUCLEAR BOILER [illegible] DOCUMENT [illegible] 557
8. CONTAINMENT SPC. [illegible] DRYWELL PRESSURE NOT LOW
9. SHIFT DRYWELL VENTILATION FANS TO SLOW SPEED AT 2 PSIG
10. LINES TO DRYWELL ATMOSPHERE RADIOACTIVITY DETECTOR MUST BE AS SHORT AS POSSIBLE WITH TEST BENDS AND STAINLESS STEEL
11. THESE INSTRUMENTS CONTROL COOLING WATER TO HAV COOLERS. THE INSTRUMENTS ARE LOCATED UNDER THE REACTOR VESSEL.
12. THESE VALVES ARE EQUIPPED WITH A MANUAL DEVICE FOR OPENING
13. THERE MUST BE AS A MINIMUM A STRAIGHT UNOBSTRUCTED RUN OF AT LEAST 15 PIPE DIAMETERS UPSTREAM OF THE FLOW ELEMENT AND 5 PIPE DIAMETERS DOWNSTREAM
14. THIS SECTION OF LINE MUST INCLUDE A STRAIGHT RUN OF PIPE SUFFICIENT FOR THE ACCURATE OPERATION OF A FLOW-METER IS AT LEAST 25 DIAMETERS LONG
15. A JUNCTION BOX WITH 15 FT OF COILED CABLING IS PROVIDED FOR EACH TEMPERATURE AND MOISTURE ELEMENT LOCATED IN THE CONTAINMENT
16. ALL VALVES AND INSTRUMENTS FOR THE NITROGEN SYSTEMS LOCATED IN THE YARD TO BE HOUSED IN A CLOSED SHEET METAL CABINET



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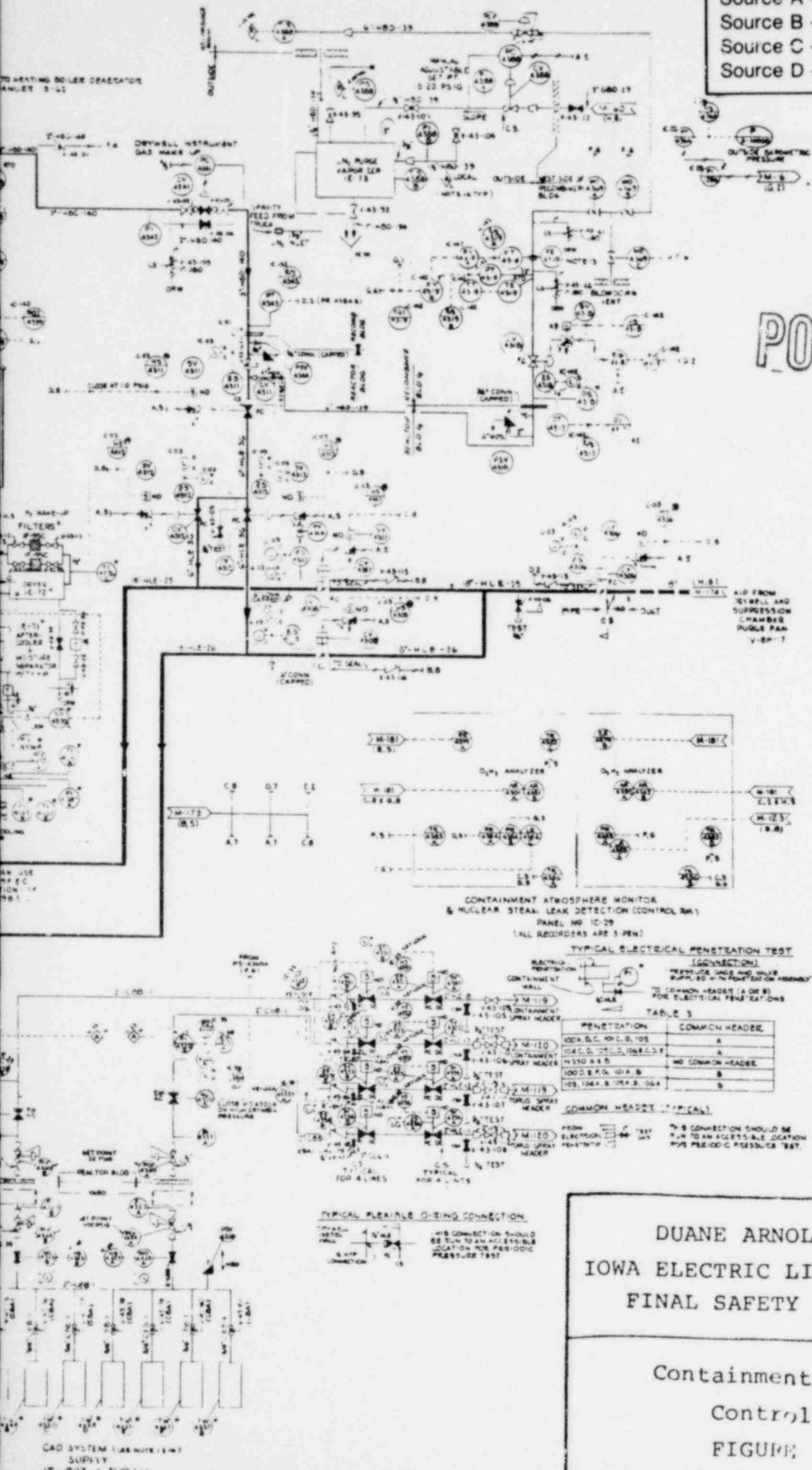
POOR ORIGINAL

LEGEND

Source A - Containment Airborne	— · — ·
Source B - Reactor Liquid	···········
Source C - Suppression Pool Liquid	— — — —
Source D - Reactor Steam	━━━━



POOR ORIGINAL

DUANE ARNOLD ENERGY CENTER
IOWA ELECTRIC LIGHT & POWER COMPANY
FINAL SAFETY ANALYSIS REPORT

Containment Atmospheric
Control System
FIGURE 4.10-2

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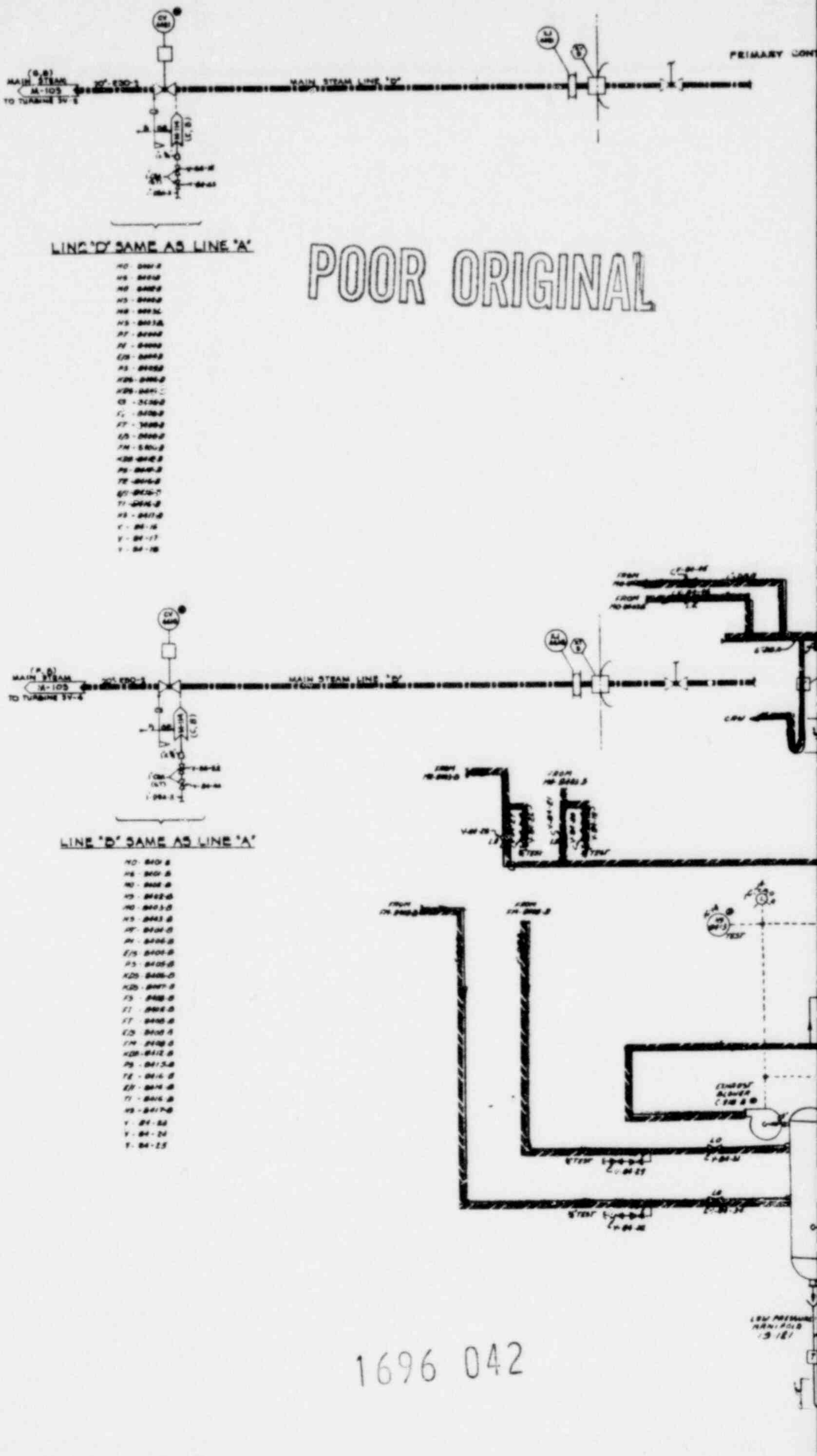
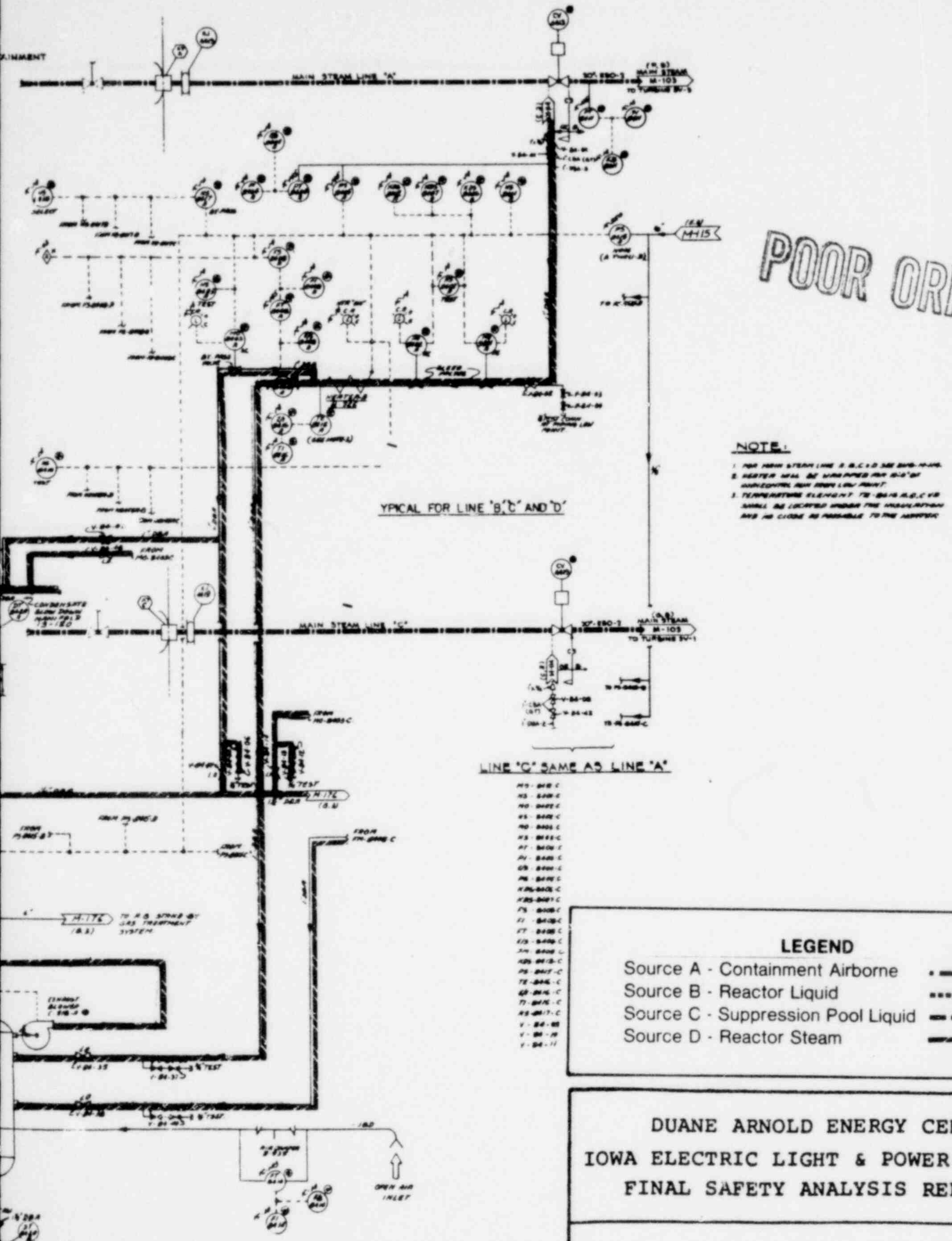
1696 041



POOR ORIGINAL

1696 042



LEGEND

Source A - Containment Airborne	— - — -
Source B - Reactor Liquid	▪▪▪▪▪▪▪▪
Source C - Suppression Pool Liquid	— — — —
Source D - Reactor Steam	═══════

DUANE ARNOLD ENERGY CENTER
IOWA ELECTRIC LIGHT & POWER COMPANY
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MSIV Leakage Control System
Flow Diagram
Figure 13-5.8-1

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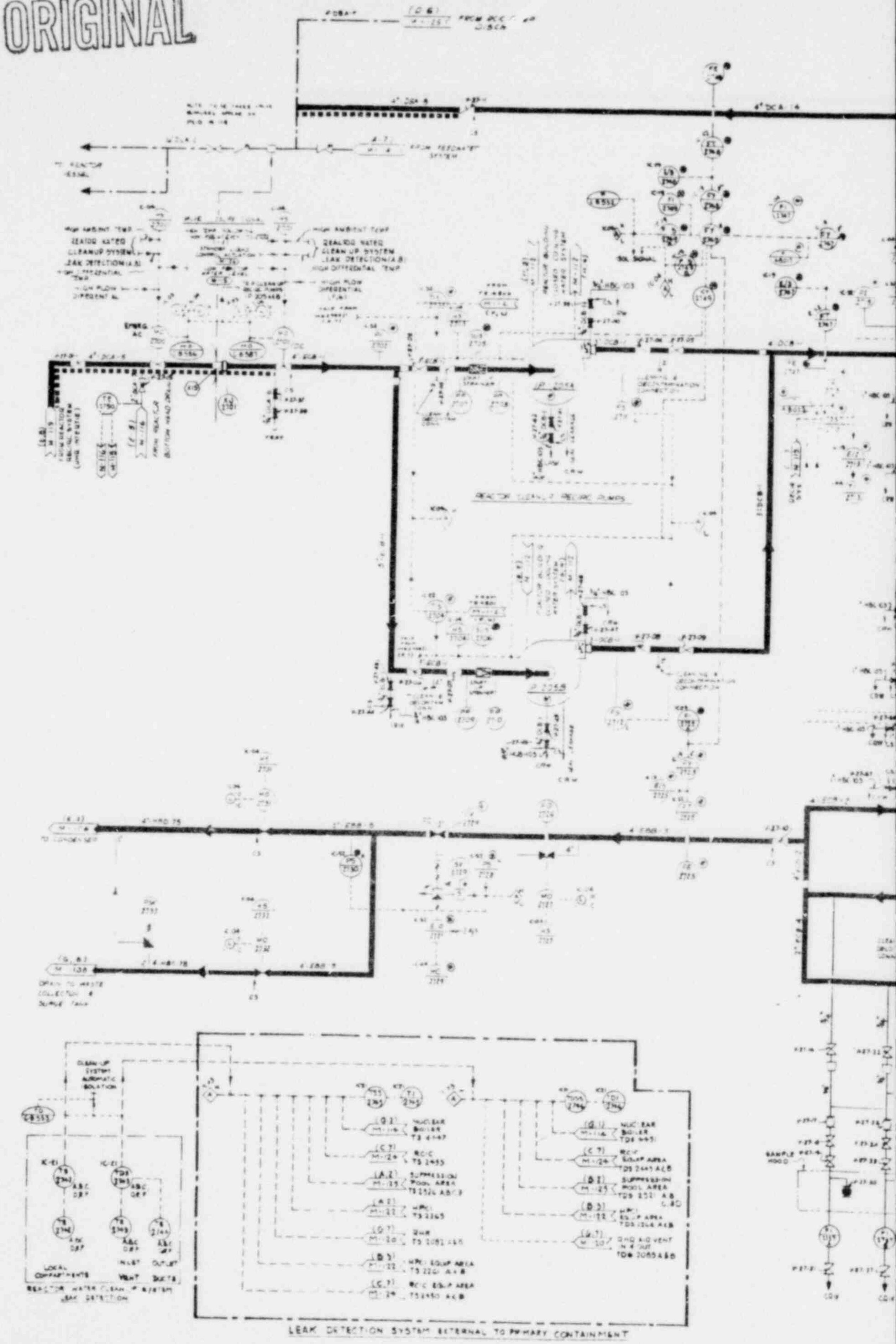
POOR ORIGINAL



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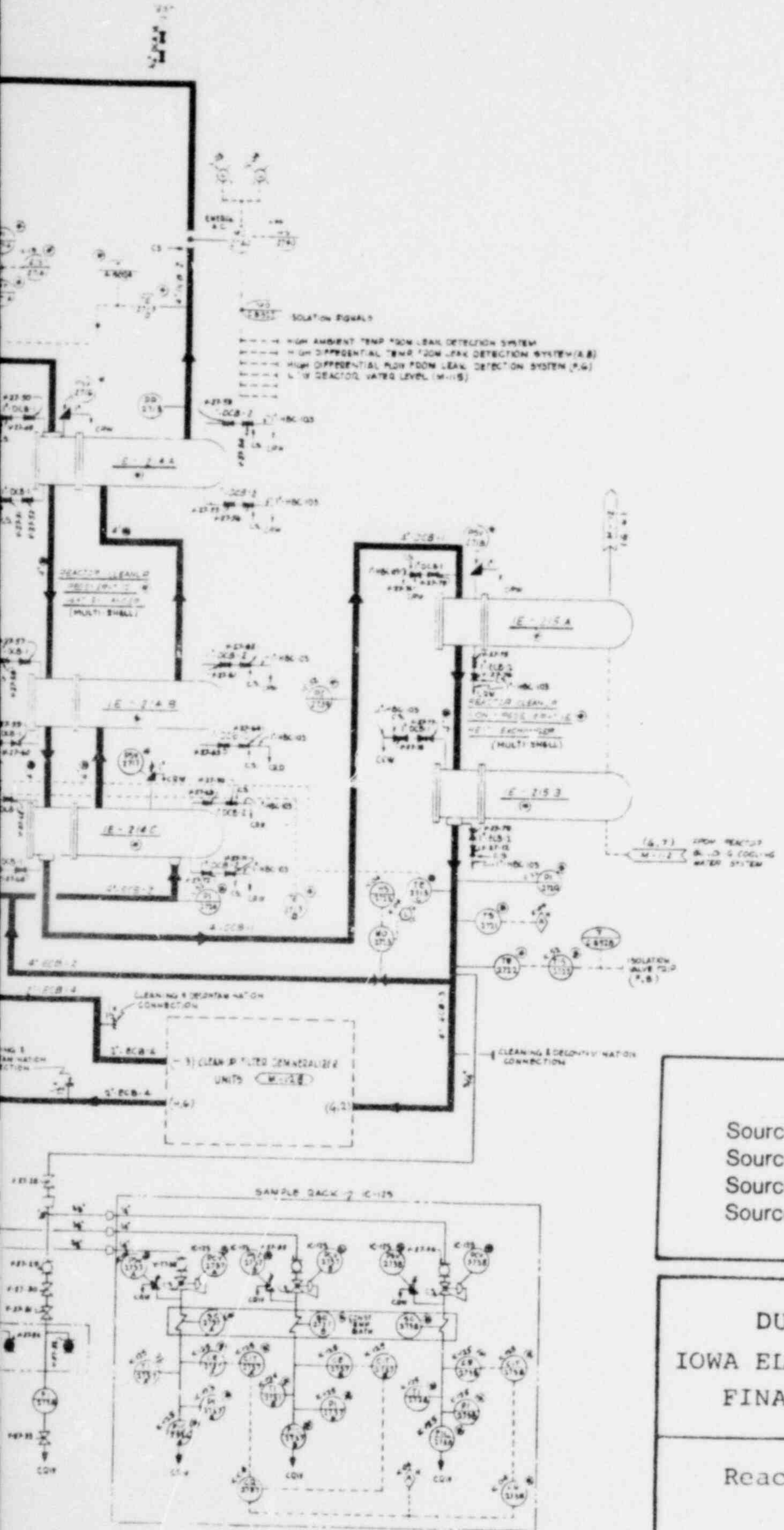
POOR ORIGINAL



NOTE:

ALTHOUGH PIPING WAS FABRICATED TO ANSI B31.7 CLASS II, VALVES CONFORM TO ANSI B31.7 CLASS III THUS, SYSTEM AS A WHOLE CONFORMS TO ANSI B31.7 CLASS III

LEGEND

Source A - Containment Airborne
Source B - Reactor Liquid
Source C - Suppression Pool Liquid
Source D - Reactor Steam

DUANE ARNOLD ENERGY CENTER
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FINAL SAFETY ANALYSIS REPORT

Reactor Water Cleanup System
P&ID
FIGURE 4.9-1 SHEET 1

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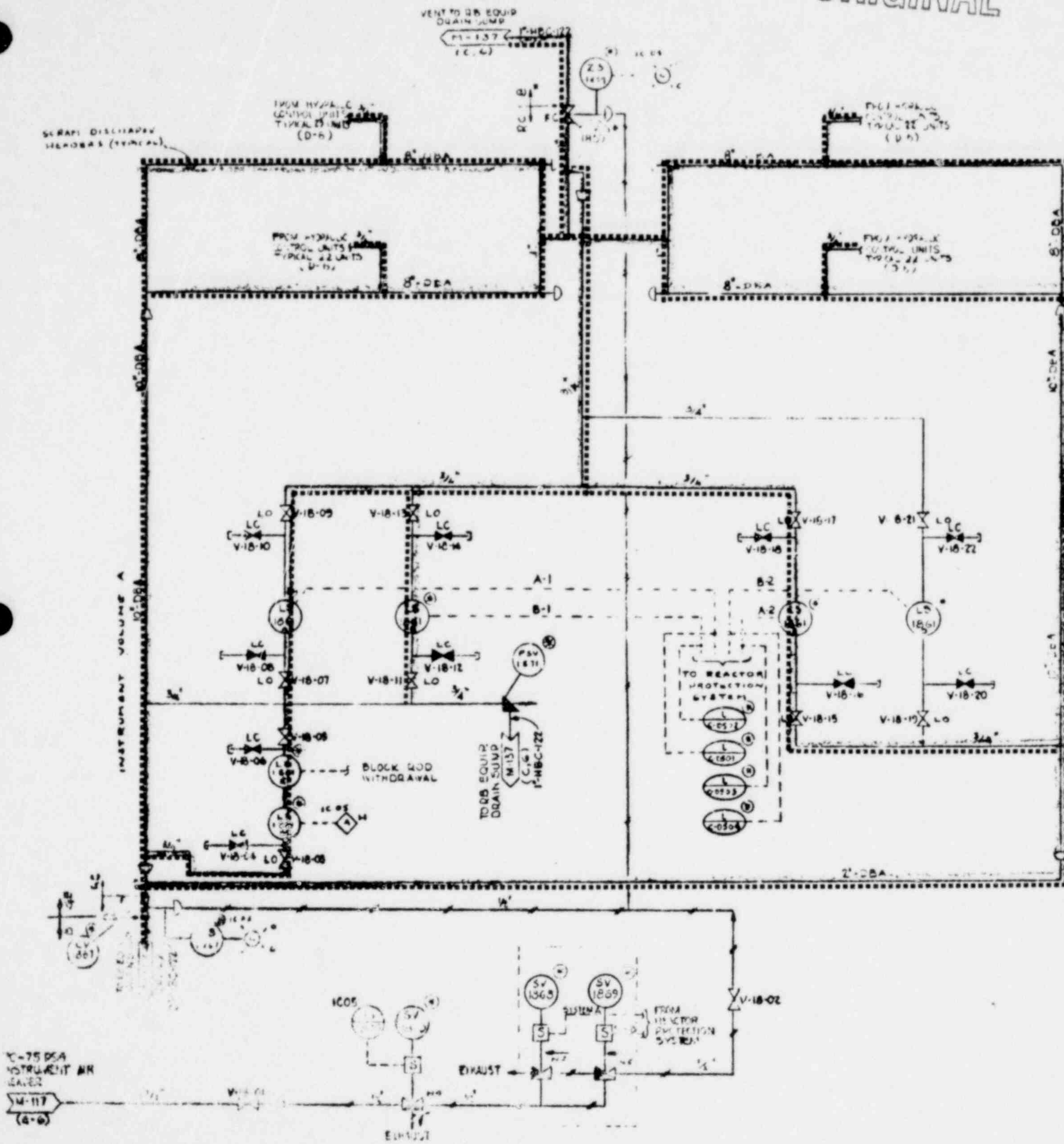
POOR ORIGINAL

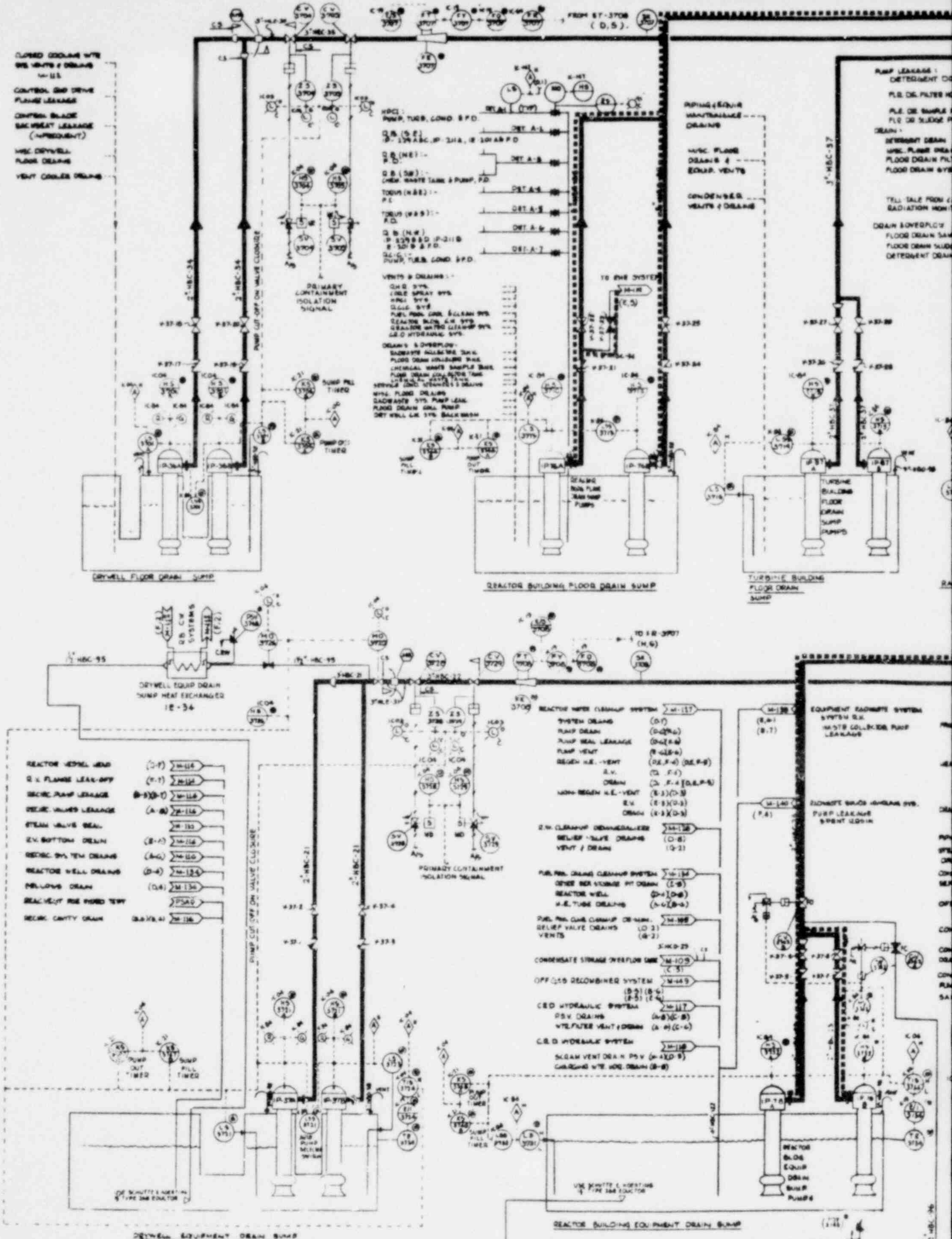


SCRAM DISCHARGE PIPING

LEGEND

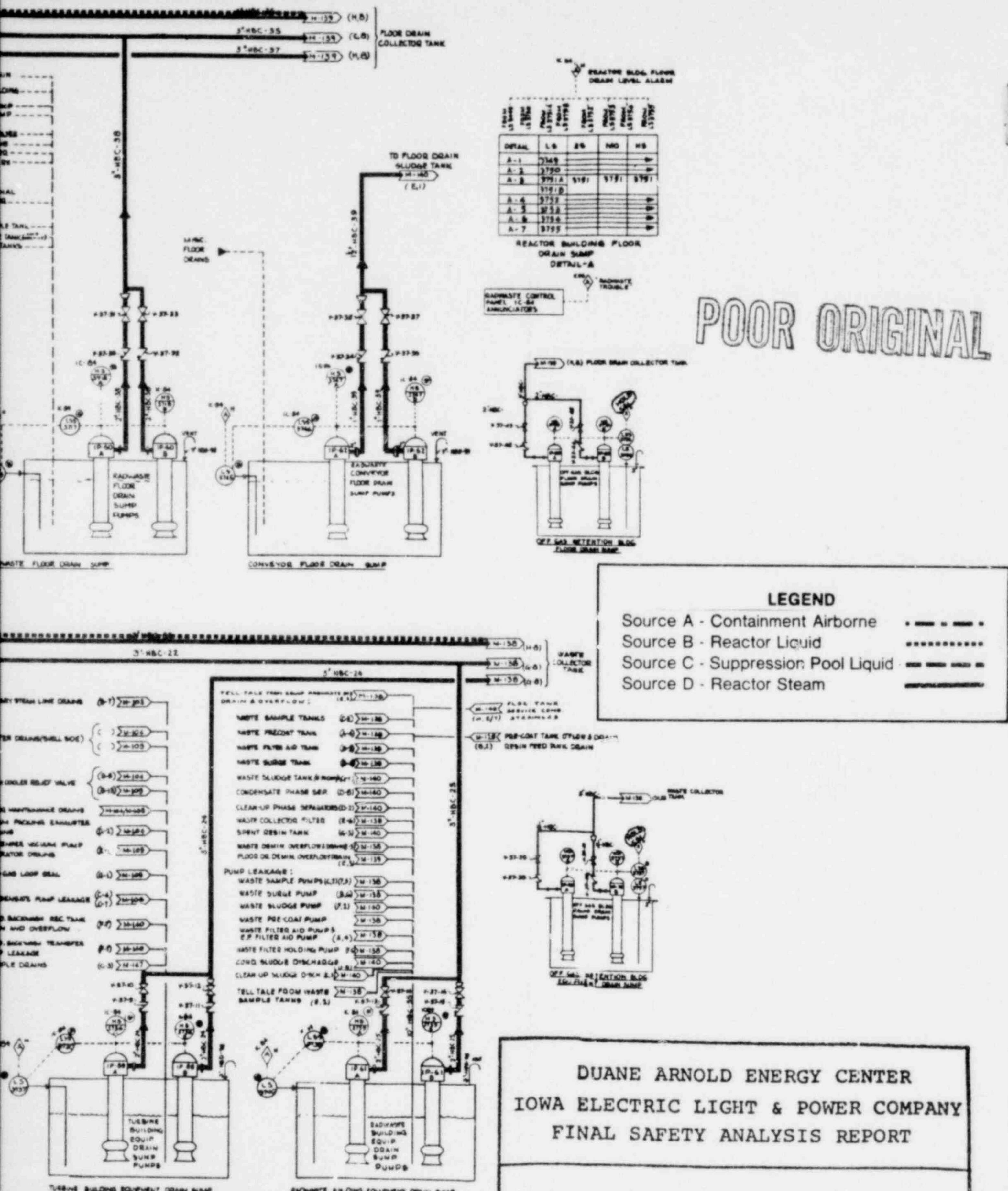
Source A - Containment Airborne	— - — -
Source B - Reactor Liquid	········
Source C - Suppression Pool Liquid	— — — —
Source D - Reactor Steam	————

1696 046



POOR ORIGINAL

1696 047



POOR ORIGINAL

LEGEND

Source A - Containment Airborne
Source B - Reactor Liquid
Source C - Suppression Pool Liquid
Source D - Reactor Steam

DUANE ARNOLD ENERGY CENTER
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FINAL SAFETY ANALYSIS REPORT

Liquid Radwaste System - P&ID

FIGURE 1-9.1-1

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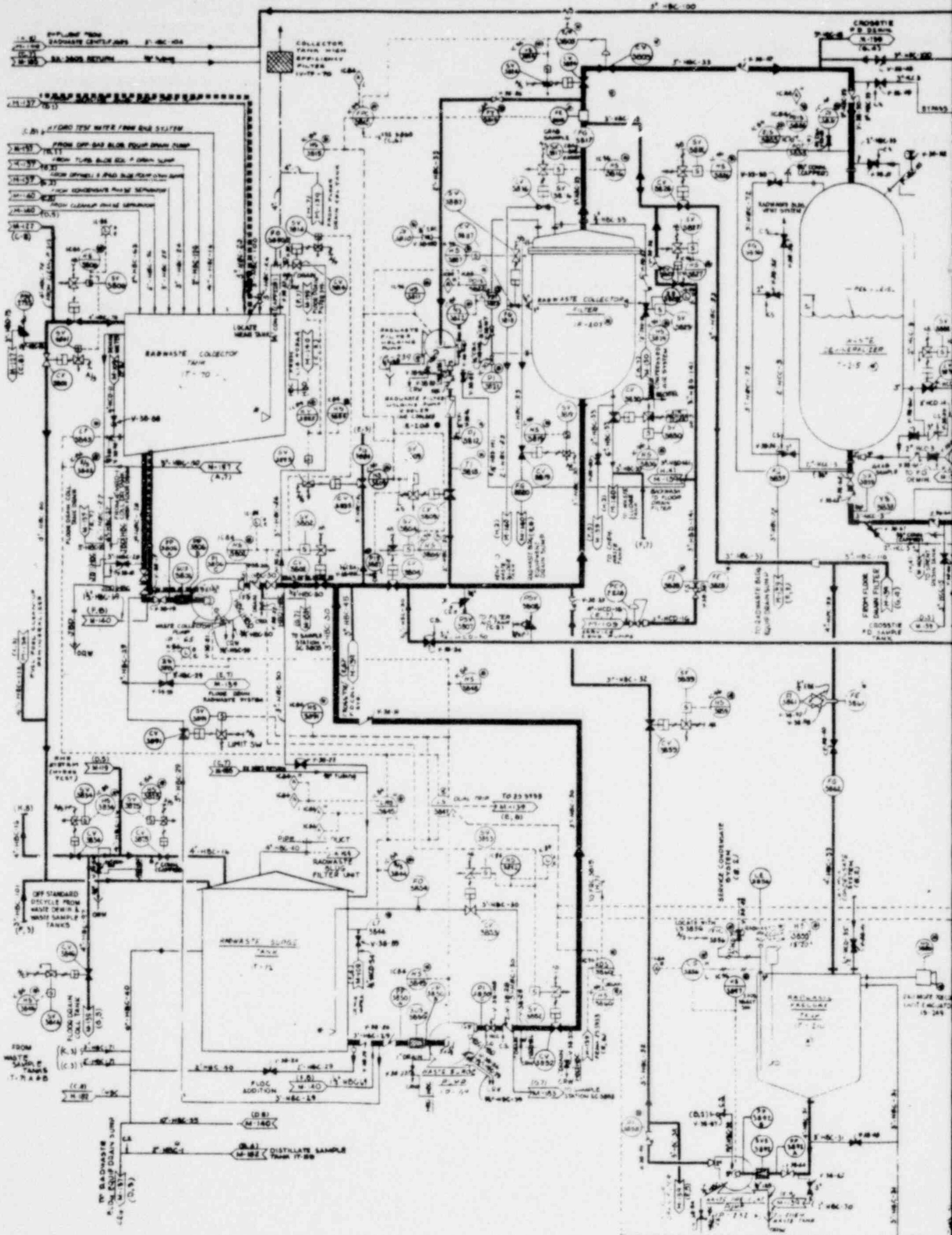
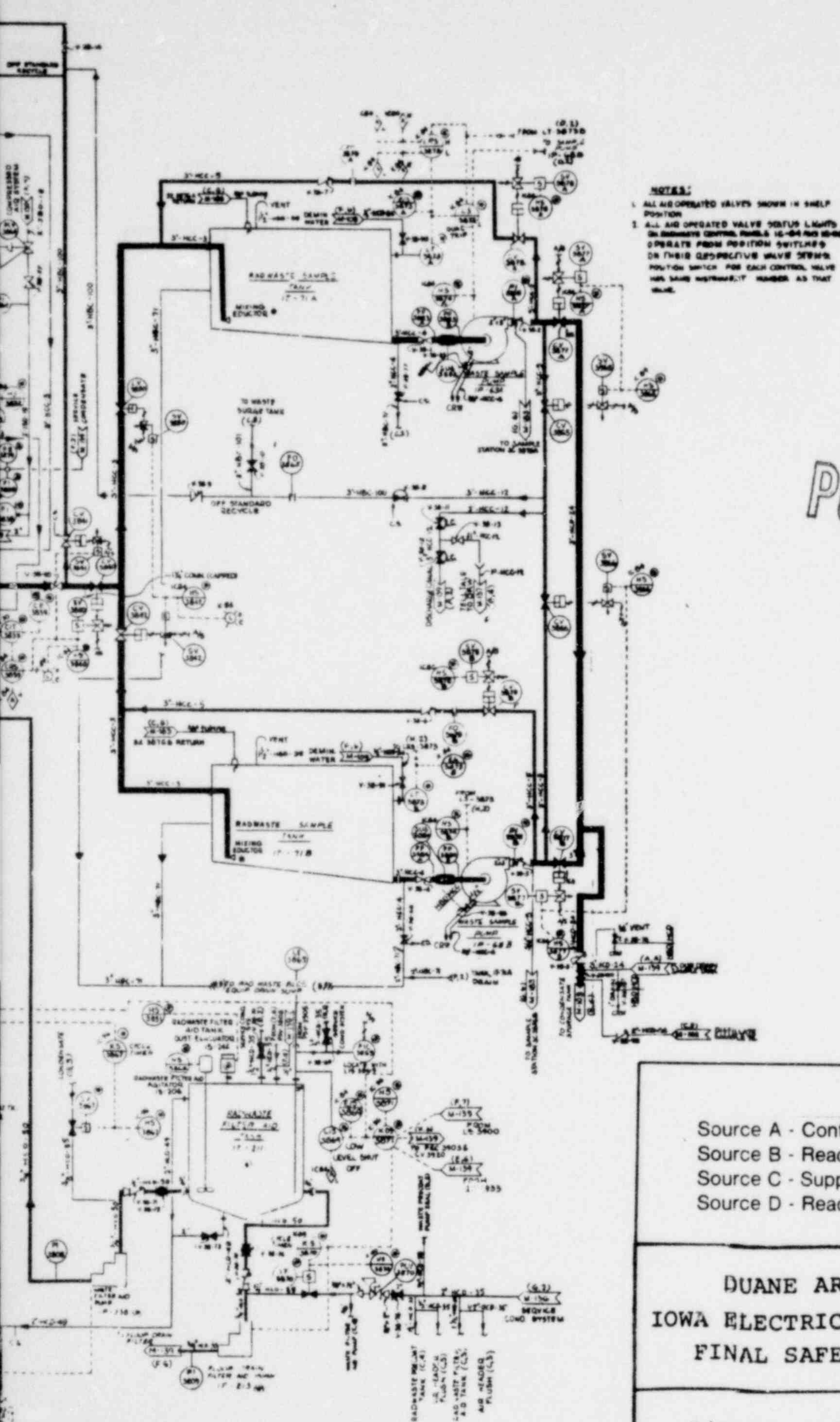
1696 048



1696 049

POOR ORIGINAL



POOR ORIGINAL

LEGEND

Source A - Containment Airborne

Source B - Reactor Liquid

Source C - Suppression Pool Liquid

Source D - Reactor Steam

DUANE ARNOLD ENERGY CENTER

IOWA ELECTRIC LIGHT & POWER COMPANY

FINAL SAFETY ANALYSIS REPORT

Liquid Radwaste System - P&ID

FIGURE 1-9.1-2

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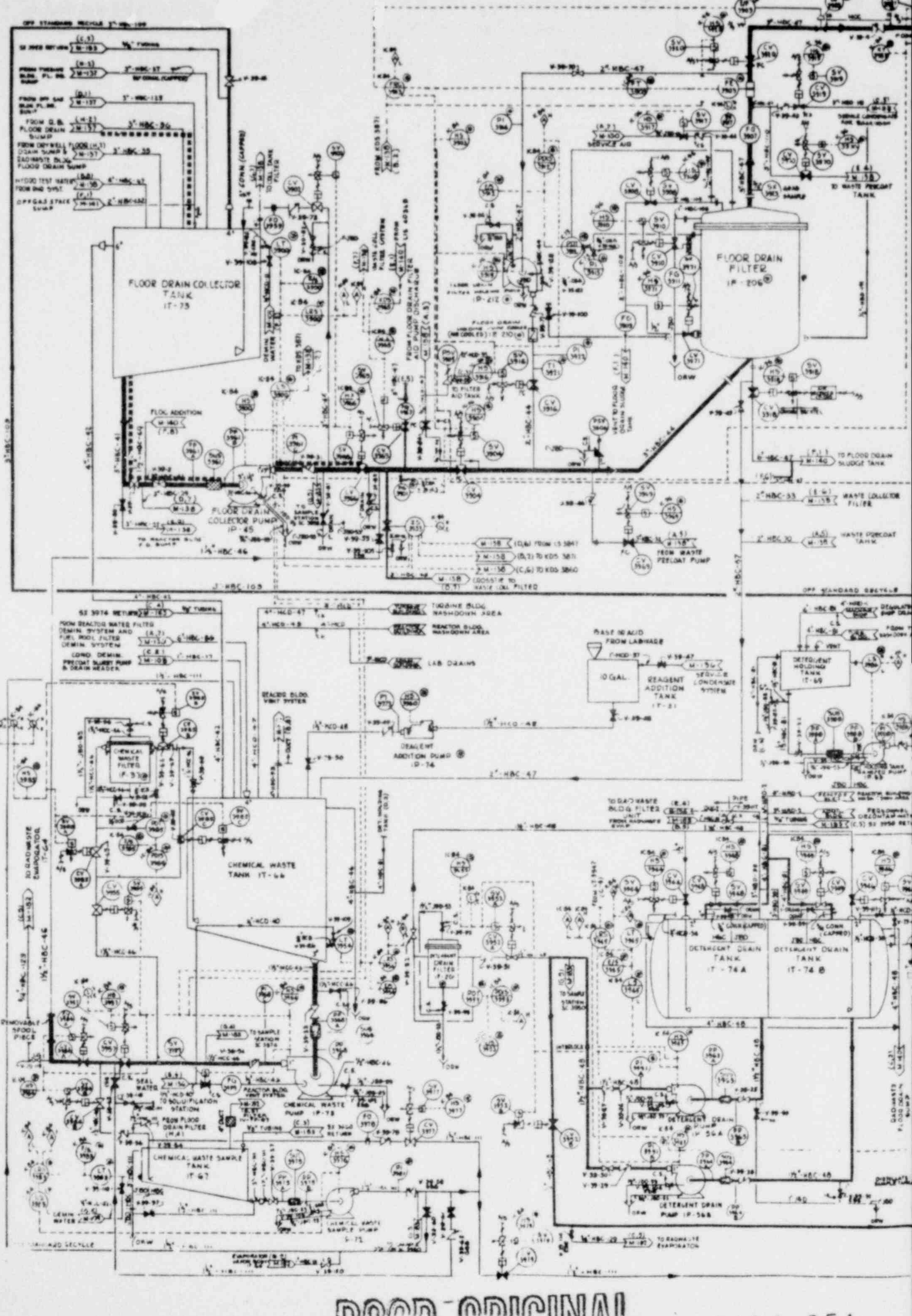
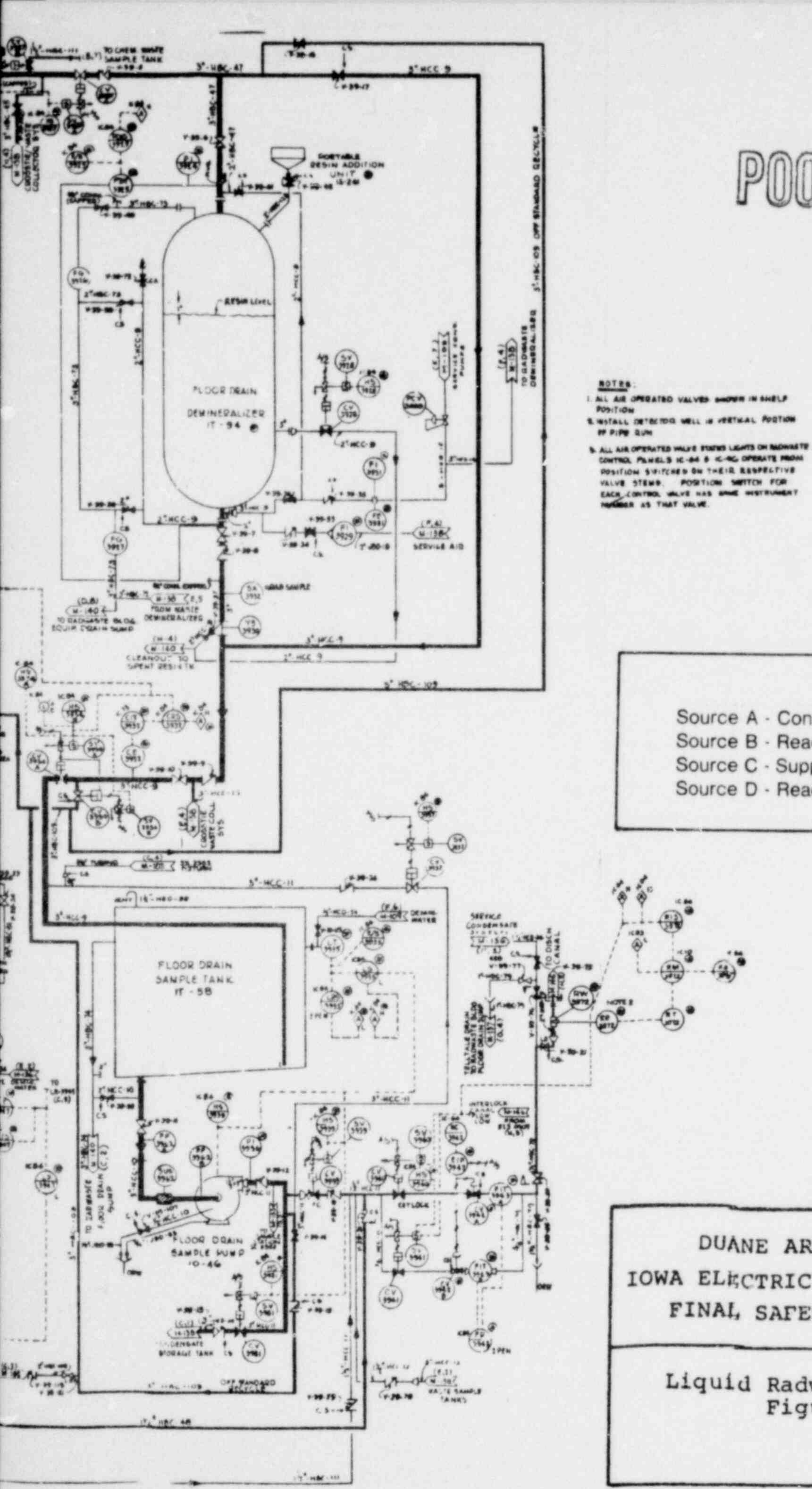


POOR ORIGINAL

1696 051

POOR ORIGINAL



LEGEND

Source A - Containment Airborne	
Source B - Reactor Liquid	
Source C - Suppression Pool Liquid	
Source D - Reactor Steam	

DUANE ARNOLD ENERGY CENTER
IOWA ELECTRIC LIGHT & POWER COMPANY
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Liquid Radwaste System - P&ID
Figure 1-9.1-3

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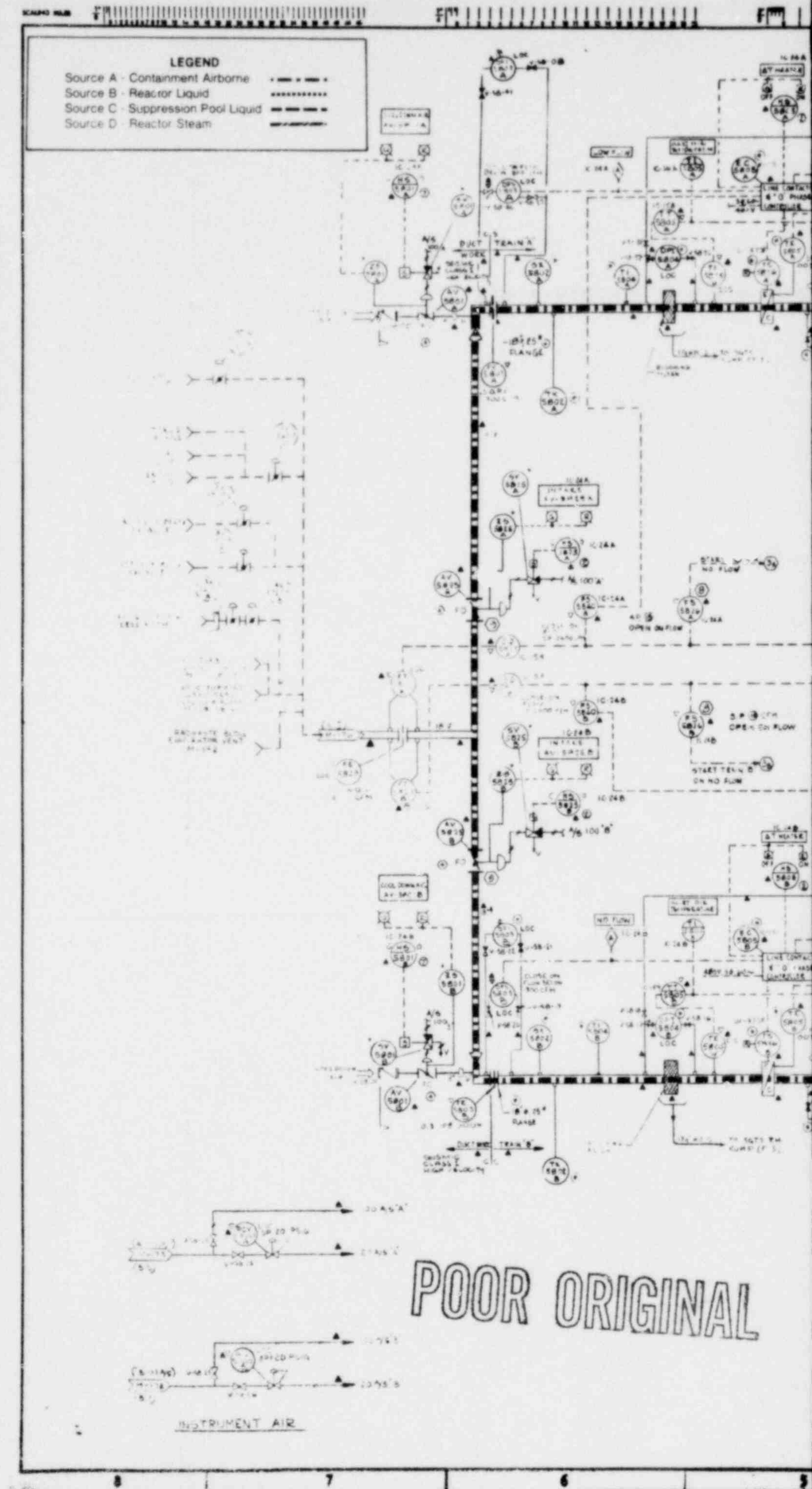
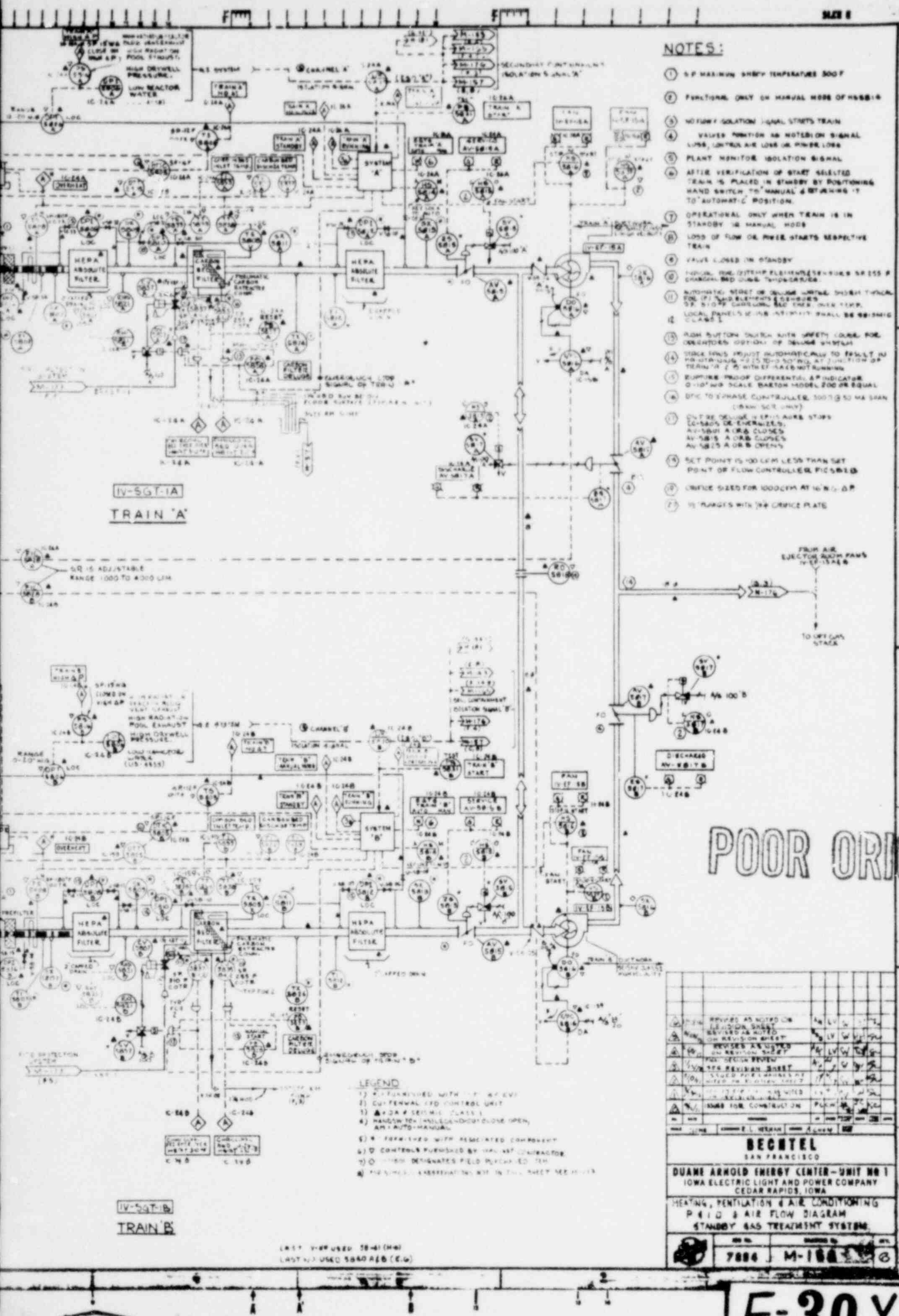

LEGEND
Source A - Containment Airborne
Source B - Reactor Liquid
Source C - Suppression Pool Liquid
Source D - Reactor Steam
INTAKE AV SPC B
COOL DOWN AC AV SPC B
NO FLOW
INSTRUMENT AIR
POOR ORIGINAL
8
7
6
5

1696 053

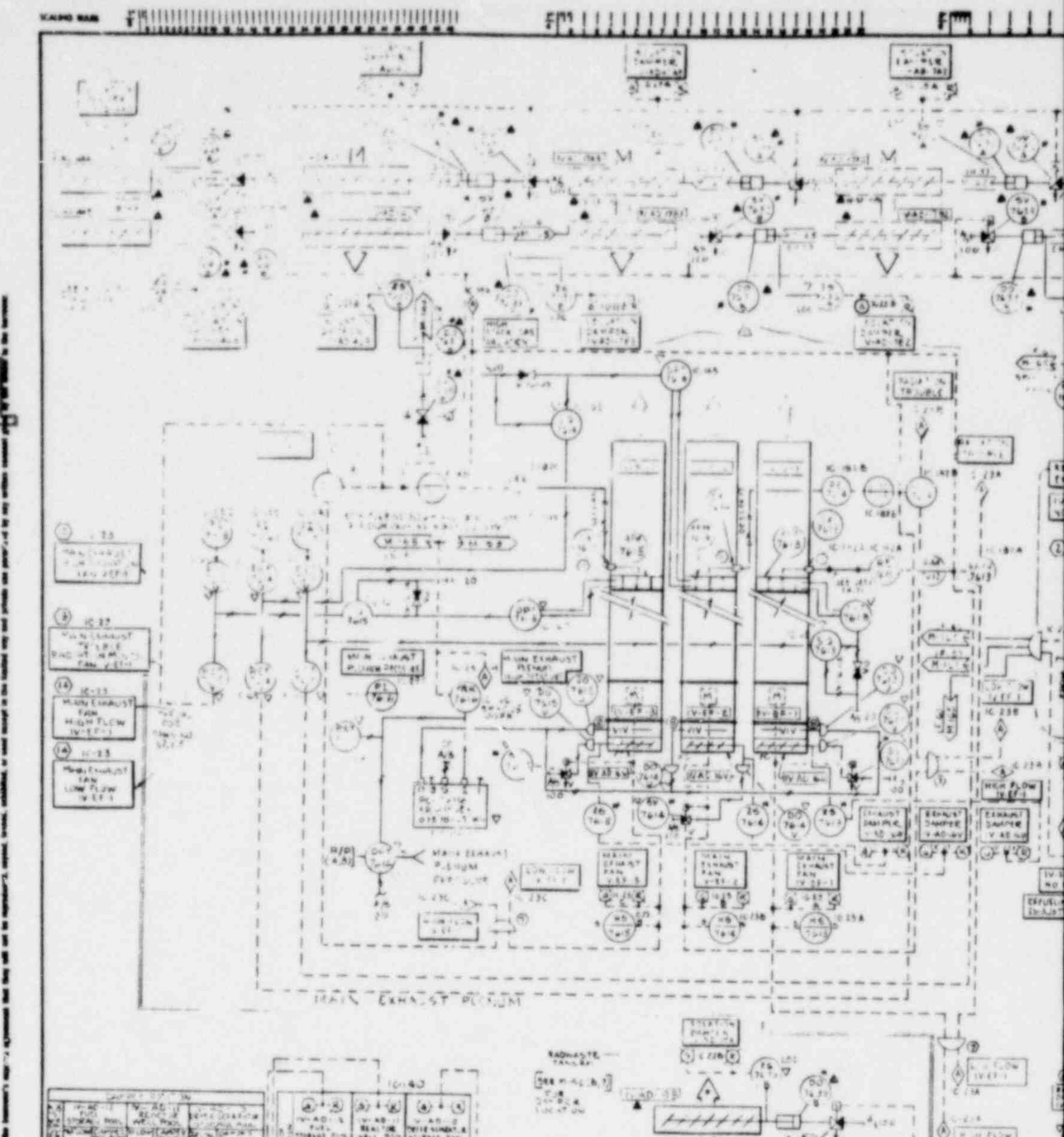


POOR ORIGINAL

TE-30X

1696 054

POOR ORIGINAL



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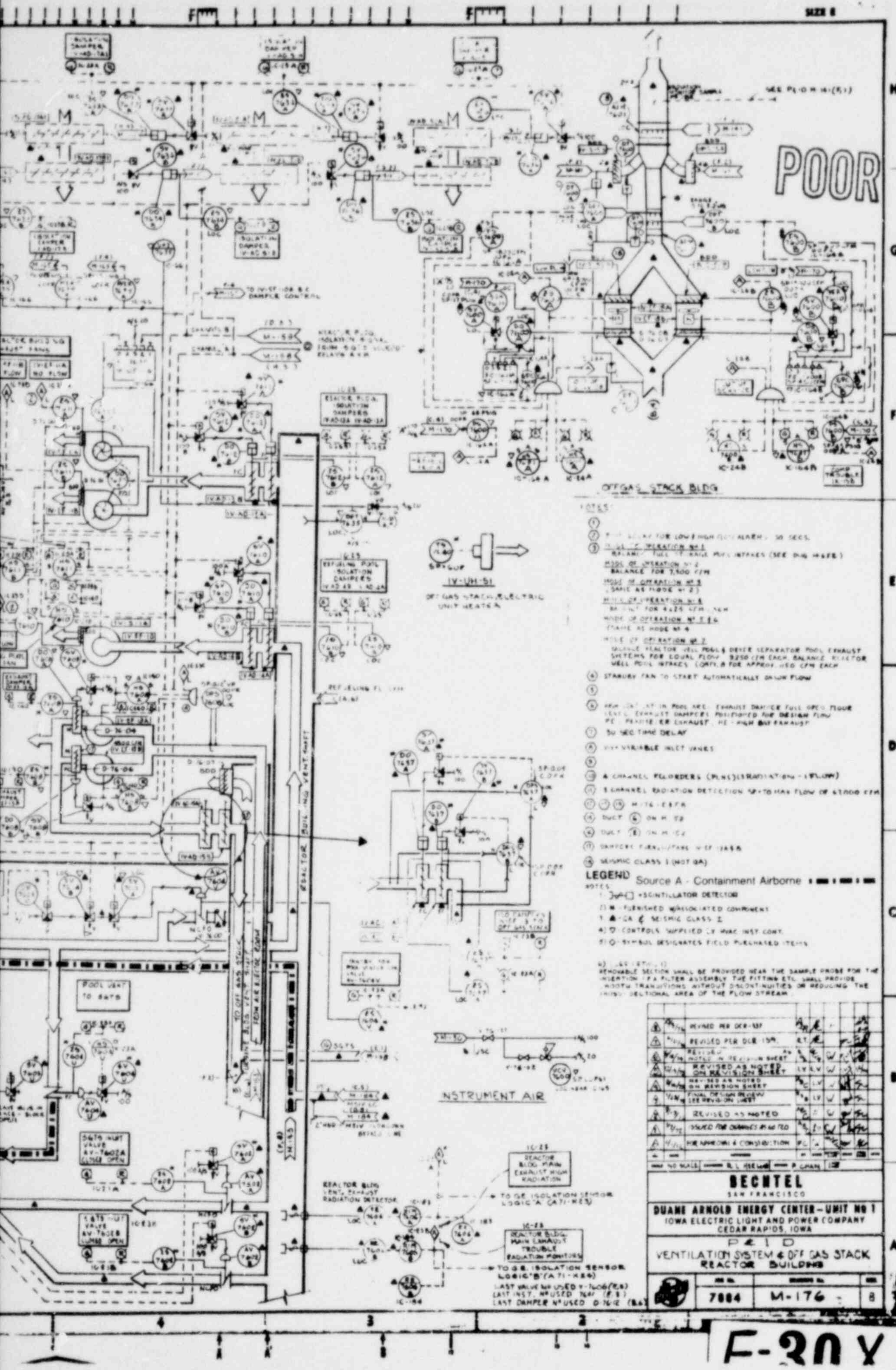

POOR ORIGINAL

F-30Y

1696 056

APPENDIX C

DESIGN CRITERIA

AND

CONCEPTUAL DESIGN DESCRIPTION

FOR THE

TECHNICAL SUPPORT CENTER

FOR

IOWA ELECTRIC LIGHT & POWER COMPANY

DUANE ARNOLD ENERGY CENTER

Prepared by
Bechtel Power Corporation

December 20, 1979

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TABLE OF CONTENTS

ATTACHMENTS

1.0 SCOPE AND OBJECTIVES

This report provides design criteria and a conceptual design for the Duane Arnold Energy Center (DAEC) technical support center (TSC). The TSC will be designed to provide the technical support function in response to the emergency conditions as described in the DAEC Emergency Plan.

2.0 FUNCTION

The function of the TSC is to reduce the need for control room access during emergency conditions and to provide support to the reactor command and control function following a plant accident.

In the event the TSC is not available, the TSC functions would be carried out in the control room.

The TSC provides:

a) An alternative to control room monitoring and diagnosis of accident conditions

b) A location for technical and management review and approval of offsite emergency activities

c) Communication of plant status to the offsite emergency operations center, NRC, and government agencies

2.1 ALTERNATIVE TO CONTROL ROOM MONITORING AND DIAGNOSIS OF ACCIDENT CONDITIONS

Personnel located within the TSC will:

a) Establish the location, type, and cause of the accident.

b) Assess damage resulting from the accident and determine the status of plant power block and engineered safety features.

c) Predict facility response and determine post-accident performance.

d) Determine and approve corrective control room actions required to isolate and contain defective systems, and to bring the reactor to a cold shutdown condition.

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2.2 TECHNICAL AND MANAGEMENT REVIEW AND APPROVAL OF OFFSITE EMERGENCY PROCEDURES

Personnel located within the TSC will:

a) Determine the immediate effect of the accident on the health and safety of the public, and recommend actions to minimize adverse effects.

b) Monitor key parameters to assure the continued health and safety of the public during the entire post-accident period.

c) Plan logistics for personnel and materials for emergency procedures.

3.0 ARCHITECTURAL AND STRUCTURAL CRITERIA

3.1 GENERAL ARCHITECTURAL DESIGN CRITERIA

The TSC will consist of a space or adjacent spaces large enough to accommodate a team of 25 people and supporting equipment.

The people will work in this space for 8 to 16 hours per day; no sleeping accommodations are provided in the TSC. Maximum use of the station's existing facilities for eating, washing, and toilet accommodations is planned; therefore, no lunch, shower, or toilet space is provided within the TSC. Bottled water and paper cups will provide drinking water only.

The habitability requirements of the space including shielding, air conditioning, lighting, and acoustics, will be designed to minimize environmental stresses. The TSC will provide potassium iodide pills and protective breathing apparatus as required.

The TSC work station area may be used as a training area or a temporary work area. The TSC will not be used for permanent office work space.

3.2 SITE LOCATION AND DESCRIPTION

The TSC will be located within the site security boundary; however, the TSC area is not vital to the plant security system.

The TSC is sited to allow physical access to the station control room within a walking distance of 5 minutes or less. The locations under consideration for the TSC are shown in Figure 1.

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3.3 STRUCTURAL CONCEPTS

a) General

The TSC shall be designed in accordance with the criteria established herein. The TSC shall be evaluated for consistency with the criteria established for the remainder of the plant, i.e., appropriate sections of the plant FSAR shall be complied with.

b) Seismic Classification

The TSC is not a Seismic Category I structure as defined in NRC Regulatory Guide 1.29, Seismic Design Classification. The TSC shall be designed for seismic loads in accordance with UBC requirements for Zone 1, using an importance factor of 1.5.

c) Natural Phenomena

Severe natural phenomena such as floods and tornadoes shall be evaluated for consistency with the criteria established for the remainder of the plant. Analysis procedures shall be followed to assure that the structure will not collapse.

d) Shielding Requirements

Minimum concrete thicknesses shall be established to limit personnel exposures to 5 rem to the whole body over a 30-day period.

3.4 CIVIL/STRUCTURAL DESIGN CRITERIA

Appendix B provides a description of quality standards, shielding requirements, and design loads.

4.0 HEATING, VENTILATING, AND AIR CONDITIONING

4.1 FUNCTION

The technical support center HVAC system performs the following functions:

a) Heating, ventilating, and air conditioning to provide inside design conditions for personnel comfort and safety at all times, including in the event of a nuclear incident

b) Automatically shifts ventilation dampers in the makeup air path to a filtration/adsorption mode upon detecting high airborne radiation contamination

c) Minimizes infiltration from outside

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4.2 DESIGN BASES

a) The system is designed to perform the technical support center heating, ventilating, air conditioning, filtration, and radiation adsorption in a safe, reliable, efficient, and economical manner. Charcoal adsorption will be provided for use under accident conditions.

b) The equipment and components are not Q-listed or manufactured to Seismic Category I requirements. Redundant ventilation systems are not provided.

4.3 DESCRIPTION

The HVAC system will be constant volume, low velocity, single zone-type with supply, return, and outside air ductwork to distribute air to the various areas.

5.0 <u>INSTRUMENTATION</u>

5.1 INPLANT DATA

The TSC will be provided with the following methods of monitoring plant data:

a) CCTV monitors of the control room panels, with pan and zoom capability

b) Computer readouts using line printer and CRT display of plant parameters and alarm conditions

The CCTV monitoring system will consist of camera(s), as required, to be located in the control room and a television display in the TSC. The zoom and pan controls for the camera(s) are also in the TSC. The camera(s) will monitor the control board surfaces available to the operator.

The computer equipment in the TSC will develop its inputs from a data analysis system. Personnel in the TSC will be able to call up plant data for display without using the plant operator and control room displays.

The TSC will provide the following types of data:

a) Plant safety parameters:

1) Reactor coolant system

2) ECCS system

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3) Feedwater and makeup system

4) Containment

b) In-plant radiological parameters:

1) Reactor coolant system

2) Containment

3) Effluent treatment

4) Release paths

c) Offsite radiological data:

1) Meteorology

2) Offsite radiation level

In addition, radiation monitors and a portable air sampler will be provided to monitor radiation levels and airborne activity concentrations within the TSC.

5.2 OFFSITE RADIOLOGICAL DATA

Appropriate meteorological data will be available within the TSC. Calculating capability will exist within the TSC for assessment of meteorological conditions and offsite doses.

6.0 <u>ELECTRICAL POWER SUPPLY</u>

The TSC shall be considered a separate entity from the existing power block. A normal source of electrical power will be taken from an existing non-Class 1E 480-volt load center that is part of the station common services. When normal power is lost, a backup power supply will be provided.

The electrical design for the TSC is to consist of the power distribution, electrical controls, lighting, and grounding systems and shall be designed in accordance with the latest applicable sections of the National Electrical Code.

7.0 <u>RECORDS STORAGE/REPRODUCTION FACILITIES</u>

Records will be stored within the TSC and/or will be readily available at the station records storage area located in the adjacent administration building. Records stored in the TSC will include design documents as required to diagnose plant problems at the system level.

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The documents will be stored on aperture cards, microfiche, or in hard copy form, as appropriate. Reproduction facilities will be located within the TSC.

8.0 COMMUNICATIONS

Offsite communications will be provided via standard telephone and microwave link. Approximately 10 telephones will be provided for communication to the emergency offsite response center, NRC, and government agencies. A microwave communications link will be available between the TSC and IE headquarters in Cedar Rapids.

Capability to transmit data from the TSC to the NRC, nuclear steam system supplier, architect-engineer, or IE engineering staff in Cedar Rapids will be investigated.

Communications between the TSC and the control room will be provided via dedicated telephone and sound powered telephone.

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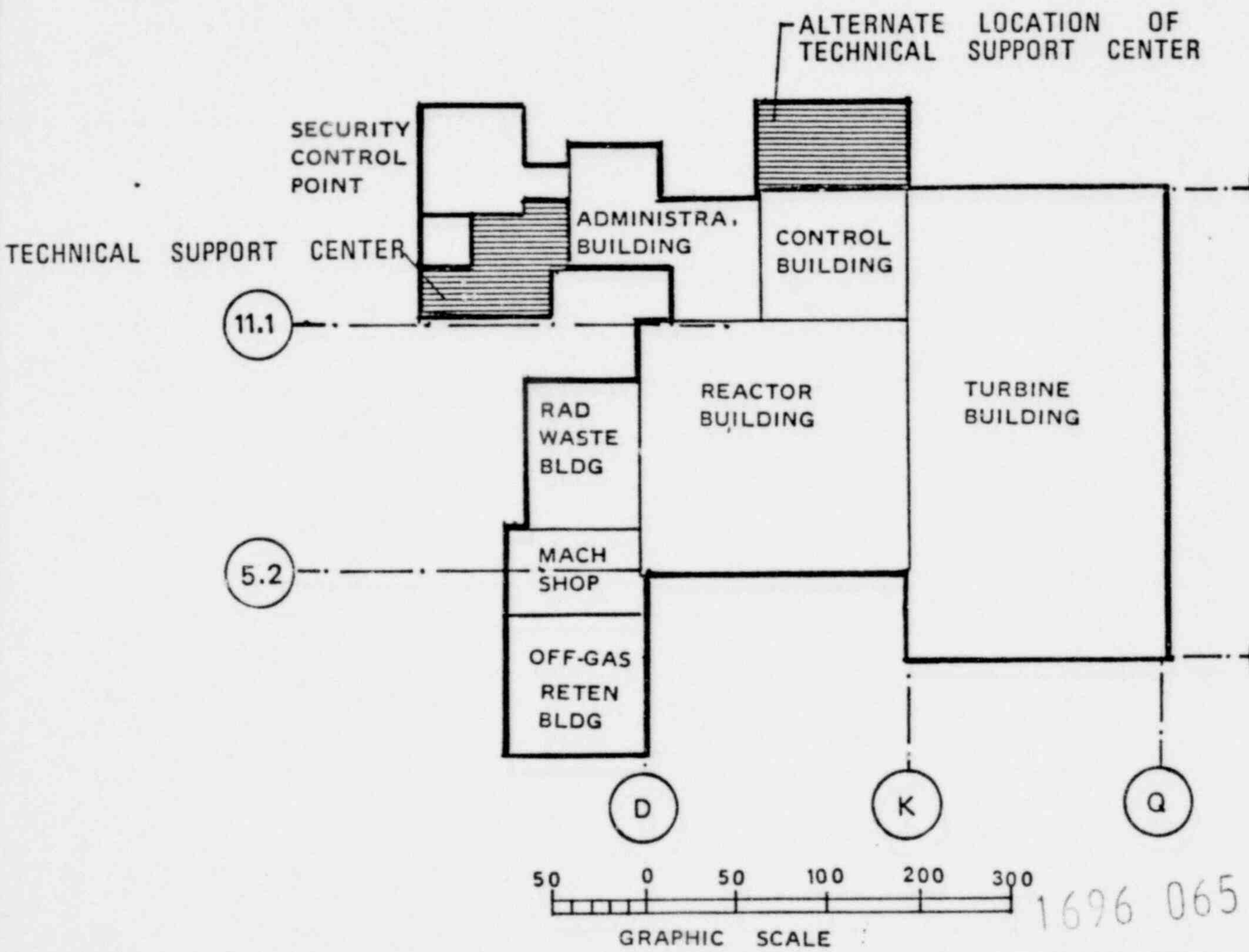
NOTES

1. TRAVEL DISTANCE BETWEEN TSC AND CONTROL ROOM= APPROXIMATELY 28'-6" VERTICAL AND 110' HORIZONTAL.
2. FLOOR AREA OF TSC APPROXIMATELY 2400 SQUARE FEET.
3. FLOOR AREA OF TSC HVAC APPROX. 630 SQUARE FEET.
4. IF NEW EMERGENCY POWER SOURCE IS REQUIRED TO SUPPORT TSC, ADDITIONAL SPACE WILL BE REQUIRED FOR 100 KW DIESEL GENERATOR, DAY TANK, BATTERY AND CHARGER, MOTOR CONTROL CENTER AND/OR SWITCHGEAR.



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No.	DATE	REVISIONS	BY	CH'K	APPR
A	12/17/79	ISSUED FOR CLIENT APPROVAL	MH	MH	

ORIGIN			
	BECHTEL	IOWA ELECTRIC LIGHT AND POWER COMPANY DUANE ARNOLD ENERGY CENTER TECHNICAL SUPPORT CENTER LOCATION PLAN	JOB No. 11186-215 FIGURE 1 REV A

APPENDIX D

LEAK RATE TEST RESULTS

NUREG 0578, SECTION 2.1.6.a

DUANE ARNOLD ENERGY CENTER

IOWA ELECTRIC LIGHT AND POWER COMPANY

JANUARY 1, 1980

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DUANE ARNOLD NUCLEAR ENERGY CENTER

SYSTEM RHR -A LOOP INSPECTOR: Dale Snyder & D W Marquardt

Test Results-D-1

SYSTEM MODE OPERATING PRESSURE TEST DATE 12-13-79

SYSTEM INTEGRITY TEST

SHEET 1 OF 1

SYSTEM LOCATION S.E CORNER TORUS & RHR VALVE ROOM

COMPONENT NO.	COMPONENT DESCRIPTION	NORMAL MODE COMP. MODE (ON/OFF) (OPEN/SHUT)	LEAK RATE	CORRECTIVE ACTION DATE	REMARKS
MO 2007	GLOBE VALVE	OPEN	—		WATER AROUND STEM
MO 2005	GATE VALVE	OPEN	—		WATER & RUST STAIN
MO. 2003	GATE VALVE	OPEN	—		RUST STAINS
MO 2004	90° GATE VALVE	CLOSED	—		
MO2011					RUST AND WATER ALSO STANDING WATER
MO 2012	GATE VALVE	OPEN	—		RUST
V-20-11	GATE VALVE	CLOSED	1 DROP/MIN.		WATER ON BOLT
V-20-14	GATE VALVE	CLOSED	—		WATER & RUST
V-20-13	GATE VALVE	CLOSED	—		WATER STAINS
1P-229 A	RHR SYSTEM PUMP	—	—		RUST ON BOLTS
1P-229C	RHR SYSTEM PUMP	—	—		WATER IN BASE OF PUMP
MO-2029	GATE VALVE	CLOSED	—		WATER IN BASE OF PUMP
V-20-4	GATE VALVE	CLOSED	—		RUST STAIN
					VALVE STEM PACKING WET.

POOR ORIGINAL

1696 067

1696 068

DUANE ARNOLD NUCLEAR ENERGY CENTER

Test Results-D-2

SYSTEM INTEGRITY TEST

SHEET 1 OF 2

SYSTEM RHR LOOP "B"

INSPECTOR: [illegible] & Dale [illegible]

SYSTEM MODE OPERATING PRESSURE TEST

DATE 12/13/79

SYSTEM LOCATION NW CORNER RM., TORUS RM., RHR VALVE RM.

COMPONENT NO.	COMPONENT DESCRIPTION	NORMAL MODE COMP. MODE (ON/OFF) (OPEN/SHUT)	LEAK RATE	CORRECTIVE ACTION DATE	REMARKS
MO-1[illegible]03	GLOBE VALVE	SHUT	< 5 DROPS/MIN.		
MO-1932	GATE VALVE	SHUT	—		STANDING WATER ON INSULATION
MO-1905	GATE VALVE	SHUT	220 CC/MIN.		—
MO-1939	GATE VALVE	OPEN	—		WATER STAIN ON INSULATION
V-19-48	GATE VALVE	OPEN	8 DROPS/MIN.		
MO-1927	GLOBE VALVE	SHUT	—		RUST STAINS AROUND STEM
MO-1921	GATE VALVE	OPEN	< 5 DROPS/MIN.		
V-19-04	GATE VALVE	OPEN	—		RUST STAINS AROUND STEM
MO-1912	GATE VALVE	SHUT	—		~~STANDING WATER~~ AROUND STEM
MO-1941	GATE VALVE	OPEN	—		RUST STAINS
V-19-06	CHECK VALVE		—		RUST STAINS
CV-1963	CHECK VALVE	SHUT	—		RUSTY & WET AROUND STEM
V-19-07	GATE VALVE	SHUT	24 DROPS/[illegible]		—
CV-1726	CHECK VALVE		—		STANDING WATER

POOR ORIGINAL

DUANE ARNOLD NUCLEAR ENERGY CENTER

Test Results-D-3

SYSTEM INTEGRITY TEST

SHEET 2 OF 2

SYSTEM RHR LOOP "B" INSPECTOR: [illegible] & Dale [illegible]

SYSTEM MODE OPERATING PRESSURE TEST DATE 12/13/79

SYSTEM LOCATION N.W. CORNER RM., TORUS AREA, RHR VALVE RM.

COMPONENT NO.	COMPONENT DESCRIPTION	NORMAL MODE COMP. MODE (ON/OFF) (OPEN/SHUT)	LEAK RATE	CORRECTIVE ACTION DATE	REMARKS
V-19-11	CHECK VALVE		—		RUST STAINED
V-19-23	CHECK VALVE				RUST STAINED
V-19-124	CHECK VALVE		—		RUST STAINED
MO-1943	VALVE		—		RUST STAINED

POOR ORIGINAL

1696 069

DUANE ARNOLD NUCLEAR ENERGY CENTER

SYSTEM HPCI (Water side) INSPECTOR: [illegible] & DW Marquardt

SYSTEM MODE STAND BY DATE 12-12-79

SYSTEM LOCATION HPCI ROOM

Test Results-D-4

SYSTEM INTEGRITY TEST

SHEET 1 OF 1

COMPONENT NO.	COMPONENT DESCRIPTION	COMP. MODE (ON/OFF) (OPEN/SHUT)	LEAK RATE	CORRECTIVE ACTION DATE	REMARKS
MO-2318	GLOBE VALVE	SHUT	184 cc/min.		—
MO-2311	GATE VALVE	OPEN	—		STANDING WATER AT VALVE STEM
MO 2316	GATE VALVE	SHUT	—		RUST ON VALVE STEM
MO 2315	GLOBE VALVE	SHUT	—		RUST ON VALVE STEM
1P216	HPCI BOOSTER PUMP	—	122cc/MIN.		—
1P-216	HPCI PUMP	—	10 DROPS/MIN		—
MO 2300	GATE VALVE	OPEN	—		RUST ON VALVE STEM
V-23-5	GATE VALVE	SHUT	—		RUST STAIN
PP 2305	—	—	—		RUST STAIN
PS 2304	—	—	—		RUST STAIN
V-23-2	GLOBE VALVE	SHUT	—		RUST STAIN
V-23-3	GLOBE VALVE	SHUT	—		RUST STAIN.

POOR ORIGINAL

1696 070

DUANE ARNOLD NUCLEAR ENERGY CENTER

SYSTEM RCIC (water side) INSPECTOR: DW Marquardt & Dale Snyder — Test-Result D-5

SYSTEM MODE STAND-BY DATE 12/17/79

SYSTEM INTEGRITY TEST

SHEET 1 OF 1

SYSTEM LOCATION RCIC ROOM

COMPONENT NO.	COMPONENT DESCRIPTION	COMP. MODE (ON/OFF) (OPEN/SHUT)	LEAK RATE	CORRECTIVE ACTION DATE	REMARKS
MO-2515	GLOBE VALVE	SHUT	345 cc/min		
MO-2511	GATE VALVE	OPEN	—		STAINS ON INSUL. BELOW
MO-2510	GLOBE VALVE	SHUT	10cc/min		
MO-2517	GATE VALVE	SHUT	—		STANDING WATER AROUND STEM
V-25-04	GATE VALVE	OPEN	< 5 DROPS/MIN		
V-25-23	GLOBE VALVE	SHUT	—		STAIN ON FLOOR BELOW
1P-226	RCIC PUMP	OFF			BASE HAS STANDING WATER IN IT

POOR ORIGINAL

1696 071

Test-Results-D-6

SYSTEM CORE SPRAY

INSPECTOR [illegible]

SYSTEM MODE LOW PRESSURE MODE

DATE 12/19/78

SHEET 1 OF 1

SYSTEM LOCATION NW, SE CORNER RM., [illegible] RM., 786 Rx LEVEL

COMPONENT ID	COMPONENT DESCRIPTION	COMP. POSITION (ON/OFF) (OPEN/SHUT)	LEAK RATE	CORRECTIVE ACTION DATE	REMARKS
MO 2137	GATE VALVE	SHUT	—		RUSTY (DRY)
MO 2135	GATE VALVE	OPEN	—		RUSTY & SMALL AMOUNT OF WATER
MO 2132	GLOBE VALVE	OPEN	—		WATER AROUND STEM
PSV 2129	90° RELIEF VALVE	SHUT	—		SMALL WATER LEAK
V-21-45	GATE VALVE	OPEN	—		RUST (DRY)
V-21-32	GATE VALVE	OPEN	—		RUST (DRY)
PSV 2122	90° RELIEF VALVE	OPEN	20 DROP/MIN.		RUST (DRY)
MO 2147	GATE VALVE	SHUT	—		RUST (DRY)
MO 2120	GATE VALVE	SHUT	—		RUST (DRY)
V-21-2	GATE VALVE	OPEN	—		SMALL AMOUNT OF WATER
V-21-1	GATE VALVE	OPEN	—		RUST (DRY)

POOR ORIGINAL

1696 072

Test Results-D-7

[illegible]: LIQUID REDWASTE INSPECTOR: [illegible signatures]

[illegible]: SUMP PUMP OPERATION DATE: 12/19/79

[illegible]: RCIC Room & RADWASTE TANK ROOM

[illegible]

SHEET 1 OF 1

[illegible]	COMPONENT DESCRIPTION	COMP. MODE (ON/OFF) (OPEN/CLOSED)	LEAK RATE	CORRECTIVE ACTION DATE	REMARKS
NO LEAKS FOUND					

POOR ORIGINAL

1696 073

SYSTEM/MODE: OPERATIONAL TEST DATE: 12/2[illegible]/79

SHEET 1 OF 2

Test Results
D-8

SYSTEM LOCATION: RCIC ROOM & TORUS ROOM
(steam side)

COMPONENT NO.	COMPONENT DESCRIPTION	COMP. MODE (ON/OFF) (OPEN/SHUT)	LEAK RATE	CORRECTIVE ACTION DATE	REMARKS
MO 2515	GLOBE VALVE	OPEN	INDETERMINATE, RATE TO HIGH		THIS VALVE IS ON THE WATER SIDE OF RCIC LEAK RATE IS MUCH HIGHER THAN PREVIOUS TEST UNDER STAND-BY MODE SHOWED
1E-205	RCIC TURBINE BAROMETRIC CONDENSER	—	HEAVY WATER LEAK FROM UNDER INSULATION		
1S-203	RCIC PUMP DRIVE TURBINE	—	HEAVY STEAM LEAK, PLUME TO CIELING		

POOR ORIGINAL

1696 074

SYSTEM MODE OCCASIONAL TEST DATE 12/20/79

SYSTEM LOCATION RCIC ROOM & TORUS ROOM (Steam side)

SYSTEM INTEGRITY TEST SHEET 2 OF 3

Test Results- D-9

COMPONENT NO.	COMPONENT DESCRIPTION	COMP. MODE (ON/OFF) (OPEN/SHUT)	LEAK RATE	CORRECTIVE ACTION DATE	REMARKS
—	DRAIN POT ON 4"-EBB-15		1'-6" PLUME		WITH WATER LEAK
CV-2409	CHECK VALVE		3'- PLUME		
CV-2410	CHECK VALVE		3'- PLUME		
CV-2411	CHECK VALVE		3'- PLUME		

DUE TO HEAVY LEAKAGE FROM MO-2515 FURTHER INSPECTION IS IMPOSSIBLE

POOR ORIGINAL

1626 075

Test Results D-10

SYSTEM MODE OPERATIONAL TEST DATE 12/21/79

SHEET 1 OF 1

SYSTEM LOCATION HPCI ROOM & TORUS ROOM (Steam Side)

COMPONENT NO.	COMPONENT DESCRIPTION	COMP. MODE (ON/OFF) (OPEN/SHUT)	LEAK RATE	CORRECTIVE ACTION DATE	REMARKS
MO-2315	GLOBE VALVE	OPEN	INDETERMINATE RATE TOO HIGH		THIS VALVE IS ON THE WATER SIDE OF HPCI. LEAK RATE UNDER THE TEST MODE IS TOO HIGH TO MEASURE.
CV-2211	CHECK VALVE		CONDENSATE ON MIRROR		
CV-2212	CHECK VALVE		CONDENSATE ON MIRROR		
?	VALVE	OPEN	CONDENSATE ON MIRROR		VALVE HAD NO TAG ON IT. (3/4" LINE NEAR TURBINE)

POOR ORIGINAL

1696 076