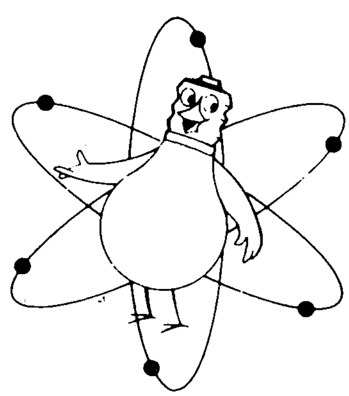
ANSWERS TO AEC QUESTIONS

DRESDEN NUCLEAR POWER STATION UNITS 2 AND 3

AMENDMENT NUMBER 16 FOR UNIT 2 AND AMENDMENT NUMBER 17 FOR UNIT 3

AMEND. 16 & 17



Commonwealth Edison Company

2566

TABLE OF CONTENTS

1. INTRODUCTION

This amendment provides additional information pertinent to Dresden Units 2 and 3. It includes design changes to the plant, answers to specific questions, and supplementary information for clarification of information presented in earlier amendments.

2. CONTAINMENT SPRAY ACTUATION CIRCUITRY

"The containment spray actuation circuitry needs to be made single failure proof."

The containment spray actuation circuitry does not meet the single failure criteria in regard to the two containment pressure switches which prevent the opening of the containment spray valves when containment pressure is less than 1 psig. This circuitry will be modified so that a single failure in this interlock circuitry will not prevent opening of the containment spray valves.

3. AC INTERLOCK

An additional interlock is being added to the automatic blowdown (ABD) system control logic. The design of this "ac interlock" is similar to that used on Oyster Creek. Use of the term ac interlock is somewhat of a misnomer and is better termed "pressure interlock" which describes both the purpose and the function more clearly.

Automatic blowdown requires both a high drywell and low low water signal to occur as well as to maintain the low low water signal for a two-minute period. In addition, the new design incorporates an added permissive in this chain. This permissive prevents actuation of the ABD function until the discharge pressure of at least one LPCI or one core spray pump exceeds 100 psig. This design is a direct indication that the low pressure ECCS pumps are operating and, therefore, that ac power is available to the ECCS systems.

The pressure permissive interlocks are designed to meet the requirements of single failure and separation. Two relays are activated when the pressure switches are closed and these relays provide dual signal paths in each of the two ABD channels. Conversely, failure of any switch or of either relay will not prevent occurrence of ABD should all the other necessary permissives be met.

4. FEEDWATER INTERLOCK TRIP POINT

The Jet Pump topical report, APED-5460, is applicable to many of the GE Boiling Water Reactor Plants. As a consequence, a number of the figures and discussions are generalized to encompass future as well as present designed plants. The typical operating map of Figure 4-1 and the discussion on page 4-20 of possible, although highly improbable, circumstance of operation at 100% recirculation pump speed at thermal power under 40% are examples of this generalization. Hence, in order to clarify these areas with direct reference to the Dresden application the following amplification is provided:

Cavitation as discussed on page 4-20 of the report does not refer to cavitation in the classical sense of the word. Jet pump operation at high temperature and pressure in the so-called cavitation region is indicated only by a minor decrease in output flow. This is substantiated by the tests discussed in Section 3 of the report. The line on Figure 4-1 shown as "Jet Pump NPSH Limit Line" is not an operating limit line but a line where a measurable drop in pump efficiency can be noted. The intercept point used for Dresden is that point where the jet pump output flow has decreased by less than 1% of the nominal flow. Figure 3-2 of the report shows the jet pump flow output as a function of subcooling. On this curve the Dresden "limit" line is equivalent to the 2.8 Btu/lb point. Operation of the pumps at lesser subcooling values has been demonstrated and, as shown on Figure 3-2, the result is only a further decrease in flow efficiency. At 0 Btu/lb subcooling, the efficiency in terms of flow has dropped less than 3%. Hence, operation as discussed on page 4-20 results in an automatic flow reduction. In essence this is a flow control line reduction that would activate the feedwater interlock and trip the recirculation pumps to 20% speed.

In addition, the area of the operating map under discussion is one that is not within the normal operating regime. Hence, it is concluded that the present 20% feedwater interlock trip point is satisfactory. Should the trip point be raised, it would not improve safety but would result in an unnecessary restriction in the *lower* power operating region which is within the operating map.

5. FLOW REFERENCE SCRAM

The flow reference scram being incorporated on the Dresden 2, 3 units is similar to that utilized in the Oyster Creek Nuclear Power Plant. The following is a description of this change:

1. The fixed scram reference signal will be disconnected from the trip circuit reference signal input in each of the six APRM's. The flow variable reference signal which is available from the existing flow network will then be connected to this same reference signal input.

2. The flow converter upscale and inoperative trips will be inserted into the scram circuitry to permit adjustment of the scram point if the flow signal fails.

6. STANDBY GAS TREATMENT INITIATION

The standby gas treatment system is initiated by two radiation monitors located on the periphery of the fuel pool. These monitors, on reading high radiation indicative of high fission product release such as in a refueling accident, will trip the isolation valves on the reactor building ventilation system. The location of these monitors is such that the transport of released fission products takes 12 seconds as discussed in Amendment 13/14, Response B.17b. The ventilation valves close in a period of 10 seconds. Thus with 0.5-second time constant for the monitors, the valves will be fully closed before fission products are transported to the valves. In this way there is assurance that fission products released into the secondary containment as a result of a refueling accident will be transported through the standby gas treatment system via the stack.

7. DIVERSITY OF ECCS VALVE INITIATION SIGNALS

The reactor low pressure circuits that permit opening the core spray and LPCI injection valves on a one-out-of-two logic basis have spatial diversity in that they are widely separated to reduce the probability of mechanical damage to both from a common event. They are designed to operate in the reactor building under all design basis environmental conditions. In addition two different types of pressure sensing devices are being employed to optimize immunity to common mode failure.

8. COMPARISON OF CONTAINMENT PENETRATION RESTRAINTS WITH THE OYSTER CREEK DESIGN

The following is to amplify a response to question I.C-1 of Amendment 9/10.

> *"Process piping primary containment penetration restraints fabricated for Dresden incorporated the design criteria established for the Oyster Creek Power Plant and submitted as Amendments 50 and 51 on Docket No. 50-219."*

For use on the Dresden project the design criteria delineated in Oyster Creek Amendment 51 were improved by changing the following criteria:

1. For Oyster Creek Case 1 (page 6) the load P_2 is less than the full break force, P. For Dresden the full break force P was utilized resulting in application of the greatest load possible to the drywell.

2. Values for P_R and M_S (page 8) for Oyster Creek were reduced when penetrations were not radial to the sphere in either plan or elevation. For Dresden values for P_R and M_S were reduced only when penetrations were not radial in plan. Therefore, these values were not always reduced as much as at Oyster Creek resulting in conservatism in Dresden calculations.

3. For calculating stresses at Point 4 (page 12), the Oyster Creek calculations treated the conditions as a solid attachment. For Dresden, the conditions were treated as a hollow attachment which result in more realistic stress values.

4. For Case VI, second paragraph (page 18); in the Dresden criteria the torsional loads were calculated on the basis of combined bending and torsion in the process piping whereas at Oyster Creek it was assumed the full strength of the pipe could be transferred in torsion.

9. SEISMIC PROVISIONS FOR INSTRUMENTS

A program is being initiated to analyze the instrument and control racks for capability to withstand seismic events. This program will include a physical test of representative samples of components (switches, recorders, indicators, etc.) and either a seismic analysis or a physical test of instrument racks including installed instruments and components. This program will be detailed in an answer to an AEC question on the Northern States Power Company's Monticello Plant, AEC Docket No. 50-263. The results of this program will be applied to the Dresden 2 and 3 Units.

10. POPULATION DATA FRONTING THE KANKAKEE RIVER ADJACENT TO THE DRESDEN SITE

"Provide staff with population in summer cottages and percentage of year cottages are occupied."

A survey was conducted of the number of houses in a direction south and southeast of the Dresden site. This survey followed the course of the Kankakee river. Edison assumed 3.5 people per house. Summer cottages are occupied between the middle of May to the end of September. Occupancy of the summer cottages vary during the summer months with peak occupancy occurring on the weekends.

Considering a distance of zero to one mile from the Dresden site south boundary there are a total of 14 permanent houses with 49 people and a total of 29 summer cottages with an occupancy of 102 people.

Following the Kankakee river out to a distance of 2½ miles from the site the number of permanent houses is 129 with a total of 451 people and the number of summer cottages is 191 with a total of 668 people. Total peak population would be 1119 people.

Therefore, within this area there are a total of 451 permanent residents. Referring to PDAR Volume III, page III-3-4, using the sectors south and southeast of the site with a distance of 0 to 5 miles the number of permanent residents are listed as 320 people. This number was based on the 1960 census.

In the FSAR Volume I, Section 2.3, a statement was made that as of June 1967 there was no significant change in population data. A statement was also made that there is a cluster of 20 cottages near the south boundary of the site.

There are now a total of 53 houses. The change in the number of houses represents both a change in permanent and summer houses over the 1967 figures.

The change in the number of permanent residents reflects the increase in population over the 1960 census. The percent increase is greater than the projected 10% increase in the population of Grundy County for a 10-year period. However, the Kankakee river is used for boating and fishing. Therefore it would be expected to be developed at a greater rate.

11. FUEL POOL DAMAGE PROTECTION

The fuel pool system incorporates design features to compensate for any damage to the fuel pool liner. In addition to the system design capabilities there are procedural controls to prevent damage from occurring.

The pool arrangement is such that major leakage due to some accidental happening should never occur. The heaviest object to be handled near the pool is the fuel cask. This cask will weigh about 85 tons. Analyses have been made to determine the damage effects on the pool should the cask be dropped. The analysis was based on a 100-ton cask load, a drop from the maximum height that the cask could be raised above the pool, and a minimum water level at the time of the drop. It was found for this case, more severe than could actually occur, that the 6-foot 3-inch thick reinforced concrete floor would not catastrophically fail. The analyses resulted in an estimate of 10 to 80 gpm leakage rate through crack paths that could develop as a result of the above postulated accident. The water would leak onto the floor beneath the pool and subsequently to the reactor building floor drain sumps. As noted previously, the sump capacity and the normal makeup capability are both greater than this calculated leakage. Any other equipment that is used in the pool would present less severe drop conditions than the cask. The subsequent section discusses the procedural controls that are delineated for Dresden operations to prevent any such occurrence.

The system design provides for minor cracks in the ¼-inch stainless steel liner. Beneath each liner seam weld is a drainage trough which directs leakage to the fuel pool liner drain network. These drains lead from beneath the liner to the reactor building floor drain system. Each pool drain outlet, of which there are a total of four, is valved closed and a flow glass is provided downstream of each valve. This arrangement aids in locating a problem area and provides a controlled flow to the reactor floor drain sumps. These drain sumps are capable of removing up to 100 gpm which is greater than any anticipated seam or liner crack leakage.

Depending upon the plant operating conditions at the time a leak is postulated to develop, there are various methods of supplying makeup water to the pool and to prevent the pool level from decreasing to an unsafe level above the fuel. The condensate transfer system is normally used to supply makeup water to the pool. This system has the least capacity of the various fill sources so only this system is discussed. There are two condensate transfer pumps in each unit and under normal conditions one pump can supply all the necessary plant makeup requirements. Should a circumstance occur which requires more than the capacity of one pump (275 gpm) the other pump can be started. This provides 550 gpm makeup. Should the event require greater capacity the two pumps of the other unit can be used. These systems are already cross tied and require only starting the pumps. Thus, capacity can be increased to above 1000 gpm. These values are in excess of any leakage that can conceivably occur.

To minimize the potential of a cask dropping into the fuel storage pool a detailed procedure of inspection and load testing of the reactor building crane will be performed at the time of a fuel cask transfer or at a minimum of every six months. In addition, the fuel cask travel path will never be over the reactor vessel or the fuel racks in the fuel storage pool. Travel over the fuel storage pool will be limited to that small area allocated for cask storage.

Prior to cask lifting operations a detailed visual inspection is made of all mechanical and electrical components of the crane. Following the visual inspection an operational test will be conducted with no load on the hook. This test will verify that all controls are operating correctly. Following these tests a load test will be conducted by raising the fuel cask approximately one foot off its rail car. The prime purpose of this test is to verify that there is no load movement after a fixed period of time. Hoisting and lowering rates will then be determined to see that they comply with the vendor's recommendations.

After confirmation of the operational acceptability of the crane the fuel cask will be hoisted to the refueling floor and moved over a controlled path to its position in the fuel storage pool.

12. SPRAY PROTECTION FOR 4160-VOLT SWITCH GEAR

"What is the potential for damage to 4160-volt switch gear located at elevation 570 feet 0 inches as a result of leakage from piping runs above the switch gear?"

Although these piping runs which are for service water and reactor building closed cooling water, are low pressure lines and no leakage is anticipated, flow deflectors will be installed above the switch gear area to assure that any leakage from the piping will not impinge on the 4160-volt switch gear.

13. MODIFICATION DUE TO DRESDEN DAM FAILURE

The spool piece presently located in the crib house and planned for temporary installation in case of failure of the Dresden Lock and Dam has been installed in the line permanently, thus eliminating this step from the operator actions necessary in the event of such failure. Further, an evaluation is being made to determine the feasibility of physically storing the stop logs inside the crib house as a means of speeding their installation in case of a failure of this dam. In any instance the stop logs will be stored in a location which will minimize the time required for their installation by station operators should a dam failure occur.

A preoperational test will be performed which will verify the ability of the system to accomplish its intended purpose. The test will also verify that the required manual operations can be performed in a reasonable time.

14. DESIGN OF MAIN STEAM LINE FLOW RESTRICTORS TO CODE

The main steam line flow restrictors are engineered devices, the design of which is not specifically delineated in the code. A code case interpretation has been requested which will be transmitted to the Commission upon receipt.

15. NON-DESTRUCTIVE TESTING ON SAFETY AND RELIEF VALVES

The following is to clarify a response to Question B.16 included in Amendment 14/15 and presents a synopsis of non-destructive testing accomplished on Dresden safety and relief valves.

A) Safety Valves - The following non-destructive tests were accomplished:

1. The seat, bushing (integral sleeve), and disc of each valve have been ultrasonically inspected per SP-20* and liquid penetrant inspected per SP-23.*

2. Each valve flange which connects to the pipe has been magnetic particle inspected per ASME Section VIII, and the connecting bolts have been magnetic particle inspected per SP-21.*

These tests meet the requirements of the drafts "ASME Standard Code for Pumps and Valves for Nuclear Power." Safety Valve environmental testing has been described in a response to Question 5.23 contained in Amendment 7/8.

* SP-20,21,23, and OS-88 are Dresser internal shop practices, and are equivalent to ASME B&PV Code, Section III, method and acceptance level standards.

In addition, all safety valve body castings are being radiographed at the following areas:
Inlet flange
Lower vertical section (portion of body into which internal sleeve screws)
Lug and 2 in. envelope of body casting around lug
These areas meet ASTM E71-Class III. The remainder of the casting will not be radiographed.

B) Relief Valves - Electromatic relief valves supplied to Dresden are similar to the valves as furnished for the Oyster Creek and Nine Mile Point Power Plants.

The following non-destructive tests were accomplished:

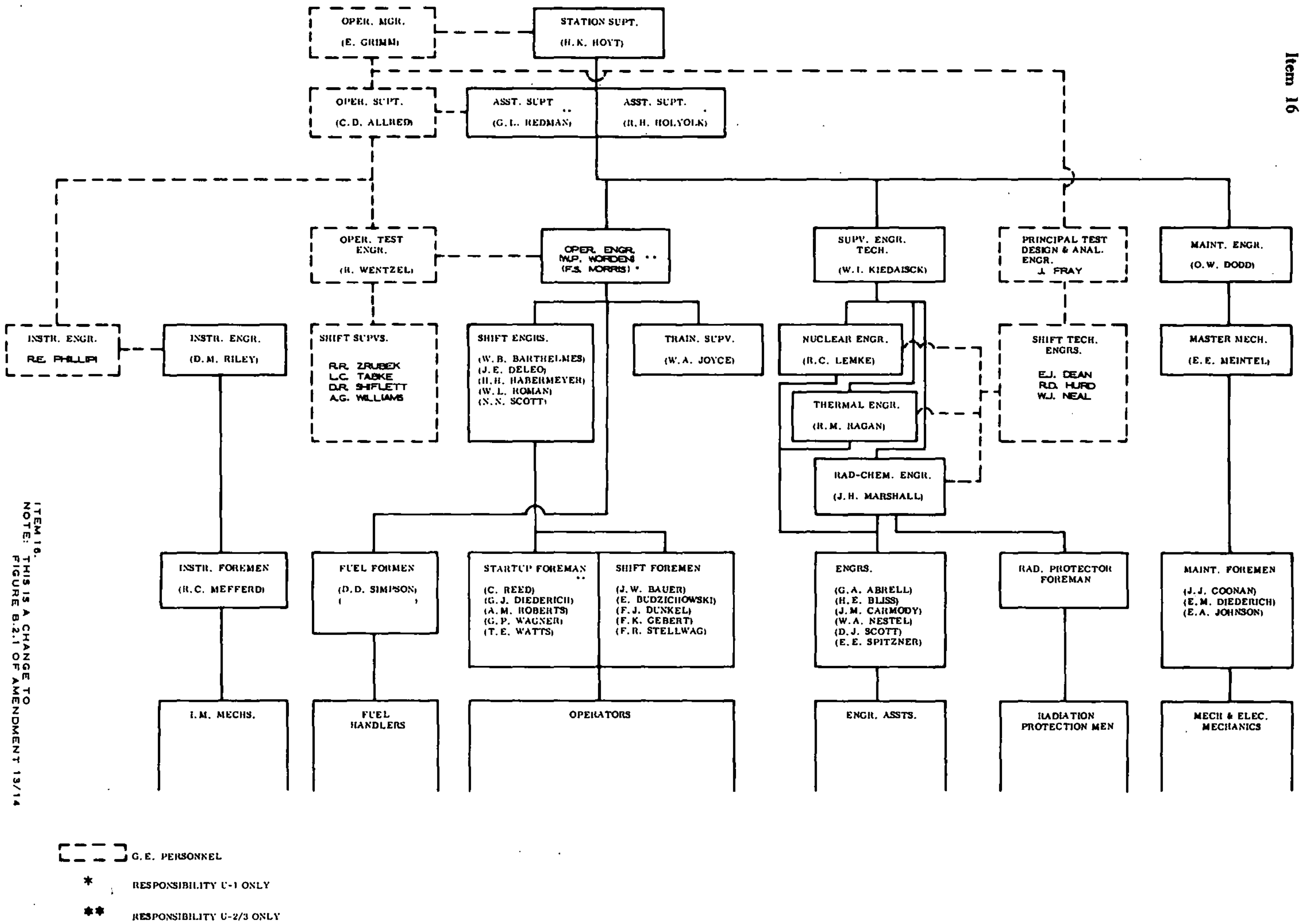
1. Valve bases (body) were UT inspected per SP-20* and magnetic particle inspected per SP-21.* In addition, welds were radiographed in accordance with Paragraph UW-51 of ASME Section VIII. Welds will also be dye penetrant checked to ASME B&PV Code Section III equivalent.

2. Valve discs were UT inspected per SP-20.*

3. Valve seat rings were UT inspected per SP-20* and the hard-face sealing surfaces were dye penetrant checked per OS-88.*

* SP-20,21,23, and OS-88 are Dresser internal shop practices, and are equivalent to ASME B&PV Code, Section III, method and acceptance level standards.

Item 16



ITEM 16.
NOTE: THIS IS A CHANGE TO FIGURE B.2.1 OF AMENDMENT 13/14

FIGURE B.2.1. DRESDEN UNIT 2 STARTUP ORGANIZATION

D 2,3

53.7

make evaluations and reports as required by regulatory agencies and good power plant practice.

3.1 Startup Organization Chart

The organization chart for Dresden Station personnel shown in the SAR Figure 13.1.1 has been expanded, Figure B.2.1, to show the specific areas of responsibility for the startup test in operation. Positions held by personnel of the General Electric startup organization are also shown in connection with the Commonwealth Edison personnel with whom close liaison will be maintained.

Commonwealth Edison has assigned an Assistant Station Superintendent (G. L. Redman) and five startup engineers (who will act as shift foreman on Unit 2) to devote full time (i.e., no responsibility for Unit 1) to the startup work activities. Except as noted in the following sentences, the entire organization shown in the Startup Organization Chart (Fig. B.2.1), will be available by assignment to assist in the startup work activities. These assignments will not reduce the Unit 1 manning below previously agreed to levels. One Assistant Station Superintendent [R.H. Holyoak] and Operating Engineer [F.S. Morris] will devote full time to Unit 1. The other Operating Engineer [W. P. Worden] will be primarily assigned as Operating Engineer for Unit 2 and secondarily assigned to Unit 1 for electrical operating matters. After Unit 2 turnover to Commonwealth Edison, the operation of Units 1 and 2 will be integrated.

3.2 Role of Licensed Personnel

The plant operations will be conducted by supervisors and operators having appropriate SO and O licenses. After proper approvals from the station superintendent and other authorization by regulatory boards, the implementation of startup tests will be by order of the assistant superintendent and/or the operating engineer electrical to the shift engineer responsible for all station operations on his shift. A shift foreman or startup engineer will have the responsibility of direct supervision of the work activities of operators assigned to Unit 2 operations in order to accomplish the startup test and operations. The

* This is a change to page B.2-8 to Amendment 13/14.

Name:	Edwin A. Grimm	Citizenship: Age:	U.S.A. 48

Employed by: General Electric Company

Job Title: Plant Test Engineer

Highest Formal Education: M.S. Physical Chemistry
University of Idaho, 1951

Experience:

June 1969 to Present
General Electric Company, Atomic Power Equipment Department, Morris, Illinois. Assigned to Dresden Nuclear Power Station as Operations Manager.

October 1966 to June 1969
Atomic Power Equipment Department, San Jose, as Plant Test Engineer. Defined and prepared pre-operational tests for Tarapur BWR. In July, 1967, transferred to IGE (India) to direct pre-op testing at Tarapur, and participate in initial nuclear operation through 100% power testing.

January 1965 to October 1966
Senior Engineer, Systems Analysis, N Reactor Plant. Controlled construction tie-in activities to modify NPR for power generation, and directed plant tests associated with turbine-generator startup. Performed controls analyses and developed modifications to improve reliability and performance.

January 1961 to January 1965
Senior Operations Engineer, N Reactor Plant Section. Reviewed instrument and electrical design for NPR, developed technical operating manuals prepared and directed pre-startup tests, developed plant operator section and training programs. Directed pre-startup testing and operating procedures during plant startup and power test program. Served on Department Manager's Startup Council and made presentations to ACRS.

August 1959 to January 1961
Reactor Operations Engineer, N Reactor Project Section. Reviewed detail design of NPR for practicability and operability. Participated in detail design of portions of reactor systems.

September 1951 to August 1959
Joined General Electric Company, Hanford Atomic Products Department, Richland, Washington on September 15, 1951, as Supervisor, Processing Operations, Irradiation Processing Department. Major responsibility was supervising the operation of a production nuclear reactor.

1. a. Name: **Ronald E. Phillipi** Citizenship: U.S.A.

Home Address: Route 1, Morris, Illinois Age: 30

b. Employed By: General Electric Company

Job Title: Field Engineer

2. a. Highest Formal Education: BSEE,
U.C.L.A., December 1967

b. Experience:

February 1969 to Present
General Electric Co., Atomic Power Equipment Department, Morris, Illinois. Preoperational testing and nuclear instrumentation calibration.

January 1968 to January 1969
General Electric Company, Nuclear Instrument Department, San Jose, California – Equal time was spent writing preoperational test procedures for Dresden and writing panel and systems test procedures for Quality Control.

January 1965 to February 1966
Hughes Aircraft Co., 11940 W. Jefferson Blvd., Culver City, California Aerospace Group - Research and Development Division. Position - Apprentice Engineer Support Technician. Develop and test prototype electronic missile circuits and systems.

February 1964 to January 1965
Teledyne Systems Corporation, 12964 Panama St., Los Angeles 66, California. Test and trouble-shoot R-390A/URR General Communication Receivers (military version), using standard test equipment.

September 1962 to March 1963
Santa Monica College, 1815 Pearl Street, Santa Monica, California. (Part-time work.) Check out tools and electronic equipment to students.

June 1962 to September 1962

Packard Bell Electronics, 12333 West Olympic Blvd., Los Angeles 64, California. (Part-time job). Check and trouble shoot commercial television receivers.

August 1957 to July 1961
U.S.Navy – Maintain, repair, and operate submarine electronic equipment, such as: SS-1 (surface search radar), AN/BPQ-1 (surface search and missile guidance radar), DAS-4 (loran), AN/BQS (active/passive sonar), also: passive sonar, fathometer, IFF, and test equipment; signal generators, syncropes, VTVM's, LCP bridge, VOM echo boxes, dummy antennas, and megers.

(All periods of time unaccounted for were spent as a student.)

Name: **Joseph Fray**

Birthdate: December 10, 1925

Education:

Degree	Institution	Year
AMCT - Elec. Engin.	Faculty of Technology Manchester Univ. (England)	1945
HNC - Physics	Harris College, Preston (England)	1963

Professional Qualifications:

Chartered Eng. M.I.E.E. — Associate Inst. of Physics

Experience:

1945 - 1948 Junior positions held with several electricity supply authorities.

1948 - 1958 Central Electricity Generating Board (England).
Progressed from General Assistant Engineer to Regional Technical Engineer with the Central Electricity Generating Board (England). This work was concerned with the field testing of generators, motors, switchgear and other primary plant in power stations together with all associated protective and control equipment.

1958 - 1960 Chapelcross Nuclear Power Station, Annan, Scotland.
Field testing of all site electrical equipment and participating in plant startup of the four nuclear reactors built by the United Kingdom Atomic Energy Authority.

1960 - 1968 United Kingdom Atomic Energy Authority, Reactor Group Headquarters at Risley.
Nuclear Engineer concerned with the design of civil nuclear power reactors with reference to plant control. Study of core statics and kinetics problems; e.g., reactivity balance calculations, local criticality hazards and limiting transients. During this period, six months spent in the field on the startup of the Advanced Gas Cooled Reactor at Windscale, involving control rod calibration, reactor physics measurements and flow tests.

1968 - Present APED - General Electric, San Jose, California.
Licensing and Safeguards Unit – Preparation and revision of the Technical Specifications for the Tarapur reactors and for TVA reactors at Browns Ferry.
Responsible for studying and revising the Startup Test Procedures for the Dresden-2 plant. Also participated in the third Test Design and Analysis training course at the GE-BWR Training Center (Simulator).

Name:	**Eric J. Dean**		
Birthdate:	March 27, 1941		
Education:	**Degree**	**Institution**	**Years**
	BS - Engin. Phy	Cornell	1963

Experience

1963 - 1967	Westinghouse (Astronuclear Lab) 1 Design and analysis of NRX nuclear reactor (1963-65 Pittsburgh, Pennsylvania). 2 Planning tests, instrumentation, predictions, analysis and reporting of results. (1965-67) (Jackson, Flats, Nevada).
January 1, 1968 to January 1969	General Electric - APED. Performed nuclear and thermal hydraulic analyses for startup testing of GE-BWR's. Prepared selected startup test procedures for Dresden-2 and Millstone (1968).
January 1969 to present	Assigned to Tarapur site near Bombay, India, as Test Design and Analysis engineer for startup of twin reactor plant. (As of July 25, 1969 both reactors had been tested to maximum power.)

Name: **William J. Neal**

Birthdate: November 25, 1929

Education:

Degree	Institution	Year
BS - Physics	Washburn University	1956

Experience:

1956-1969

Phillips Petroleum Company, Atomic Energy Division, Idaho Falls, Idaho

SPERT Projects

Reactor operator and Group Leader. SPERT I and SPERT III reactors.

Responsible for preparation of test proposals and procedures, assembly and disassembly of experimental equipment, core loading and unloading calculation of test parameters, preliminary analysis of test results. Operated and supervised the operation of the reactors during experiments.

STEP Supervisor of Test Operations

SNAPTRAN Program

Responsible for the operation and maintenance of test facility and reactor, preparation of all operation and maintenance procedure. Supervised reactor operation and coordinated all activities associated with conduct of SNAPTRAN tests.

Semi-scale Blowdown Test Program

Responsible for operation and maintenance of test facility and preparation of operating procedures.

LOFT Program

Responsible for review of LOFT reactor and facility design for operability and maintainability, coordinated preparation of System Design Descriptions.

1969-Present

APED - General Electric Company, San Jose, California

Fukushima Unit 1

Responsible for preparation of Fukushima Unit 1 Final Safety Analysis Report.

Dresden Unit 2

Contributed to review and revision of Dresden Unit 2 Startup Test Procedures.

Name: **R. O. Hurd**

Birthdate: September 30, 1940

Education:

Degree	Institution	Year
BS Physics	Washington State University	1963
MSNE	Columbia	1967

Experience:

1963 – 1967 Burns and Roe, Systems Analysis Department of the Nuclear Division. Participated in analog computer simulation of PM-3A during design analysis, similar work on the Standard Nuclear Power plant for the U.S. Army nuclear power program. Performed shield design calculations for AARR and did site study for proposed plant.

July, 1967 – Present **GE-APED**

Responsible for signal selection and specification of signal conditioning for Oyster Creek transient and stability tests. Performed thermal-hydraulics analyses for Nine Mile Point Startup Test Instructions. Responsible for Millstone Startup Test Specification and Instructions including first application of select rod insert feature. Contributed to preparation of Startup Test Procedures for Nine Mile Point, Dresden-2 and Millstone. Participated in first training course for TD&A and GE-BWR Simulator. Performed extensive error analysis on BWR in-core instrumentation systems. Shift Test Engineer during Oyster Creek fuel loading and open vessel testing.

Item 17

3.1.2 Ground Floor, FSAR Figure 12.1.31, El. 517' 6":

A. Install an additional door to achieve an air lock entry with double interlocked doors into Unit 2 from Unit 3. The air lock will be fabricated in the same manner as described in 3.1.6A, below.

B. The two normal personnel accesses and equipment lock will operate as originally designed.

C. All outside doors will be locked.

3.1.3 FSAR Figure 12.1.20, El. 545' 6". Lock door between Units 2 and 3.

3.1.4 FSAR Figure 12.1.29, El. 570' 0":

A. Lock door between Units 2 and 3.

B. Maintain normal personnel interlocked access between turbine building and reactor building.

3.1.5 Isolation Condenser Floor, FSAR Figure 12.1.5, El. 589' 0". Lock door between Units 2 and 3.

3.1.6 Operating Floor Level, FSAR Figure 12.1.4, El, 613' 0":

A. Install an air lock on the stairwell on the northeast side of the operating floor. This interlocked personnel access will be fabricated using concrete block and precast concrete roof slabs. This type of construction will meet secondary containment leak rate testing requirements.

B. Seal the stairwell at the N-44 column lines with the same material as above.

* This is a change to page B.3-4 to Amendment 13/14.

C. Seal the elevator shaft at northeast side of operating floor.

D. The equipment hatch and the hatch on the east side of the fuel storage pool will be sealed to prevent air leakage.

E. The shielding blocks over the reactor, between the reactor well and dryer separator storage area, and the shield blocks between the reactor well and fuel storage pool will be sealed as required to meet the secondary containment leak rate specifications. Material similar to that used for other secondary containment will be used for this purpose.

NOTE: ALL OF THE PRECEDING FROM 3.1.1 THROUGH 3.1.6e IS NECESSARY TO OBTAIN SECONDARY CONTAINMENT IN THE REACTOR BUILDING REQUIRED FOR THE POWER TEST PROGRAM AND FULL POWER OPERATION.

F. An 20-foot high solid personnel barrier and a suitable dust barrier will be erected across the operating floor at the column 44 line. A locked personnel door will be provided in this barrier.

3.2 Turbine and Radwaste Buildings

3.2.1 Ground Floor, FSAR Figure 12.1.31, El. 517' 6":

An 8-foot chain link fence with one personnel locked gate will run from the west side of the entry to radwaste 16 feet south, east to column 44, south to column G, west past the CO_2 system and south to column H wall. A locked personnel gate will be provided in this fence.

* This is a change to page B.3-5 to Amendment 13/14.

3.4 Outside Area

3.4.1 The tank area will be fenced by extending present north-south fence on the east side to column H wall, and extending other fences to include the two new condensate tanks. This will exclude construction from the tank area.

3.4.2 Install a temporary fence on the north side of the turbine buildings to include radwaste building, heating boiler house discharge structure, and crib house for Units 2 and 3. Access to this area will be limited to operating personnel.

4.0 Equipment Separation

Key features inherent in the plant layout and design of Units 2 and 3 which minimize the possibilities for interactions are the provision of duplicate systems for each unit and physical separation of the equipment and controls in all but a few cases as discussed below. Where possible, completion of installation and testing of these shared systems will be accomplished before Unit 2 startup. Table 4.1 in the back of this section summarizes the equipment discussed in this section.

4.1 Water Supply

The water supplied to Unit 3 for flushing and testing must be free of radioactive contamination. The source for such water is the two condensate storage tanks which could become slightly contaminated by processed water from the radwaste facility and from the Unit 2 hotwell. The piping will be modified and appropriate valves will be locked closed to isolate one of the two condensate storage tanks and maintain a source of uncontaminated water for Unit 3.

4.2 Intertied Systems

In some cases, complete, separated systems have been provided for Units 2 and 3 with interconnecting piping.

---

* This is a change to page B.3-7 to Amendment 13/14.

18. FUEL SURVEILLANCE PROGRAM

General Electric will perform a surveillance program on Boiling Water Reactor fuel, either production or developmental, which operates beyond current production fuel experience as it becomes available for inspection. The schedule of inspections will be contingent on the availability of the fuel as influenced by plant operating and facility requirements. The program will include surveillance of reactor plant off-gas activity, relevant plant operating data and fuel inspection.

The lead full length fuel rods with respect to exposure, linear heat generation rate, and the combination will be inspected. Inspection techniques to be used are:

1. Leak detection tests, such as "sipping."

2. Visual inspection with various aids such as binoculars, borescope, and/or underwater TV with a photographic record of observations as appropriate.

3. Non-destructive testing of selected fuel rods by ultrasonic test techniques.

4. Diameter measurements of selected fuel rods.

Unexpected conditions or abnormalities, such as distortions, clad perforation, or surface disturbances will be analyzed. Resolution of specific technical questions indicated by site examinations may require examination of selected fuel rods in RML facilities.

The results of the program will be used to evaluate the Boiling Water Reactor Fuel design methods and criteria used by General Electric.

The results of the surveillance program will be reviewed with the Division of Reactor Licensing at appropriate intervals.

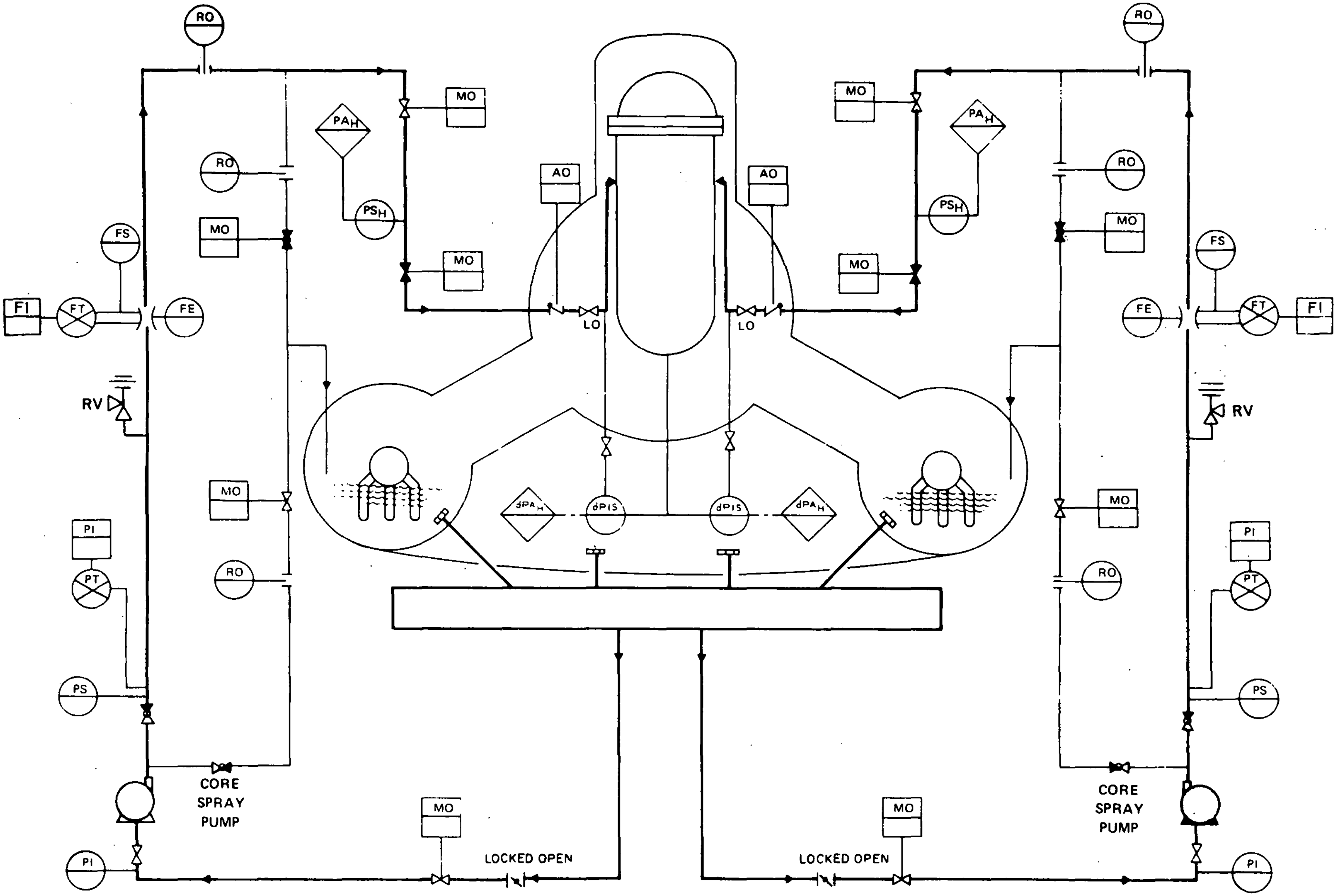


FIGURE 6.2.2 CORE SPRAY COOLING SYSTEM P&ID

Revised 8/7/69

53, 17

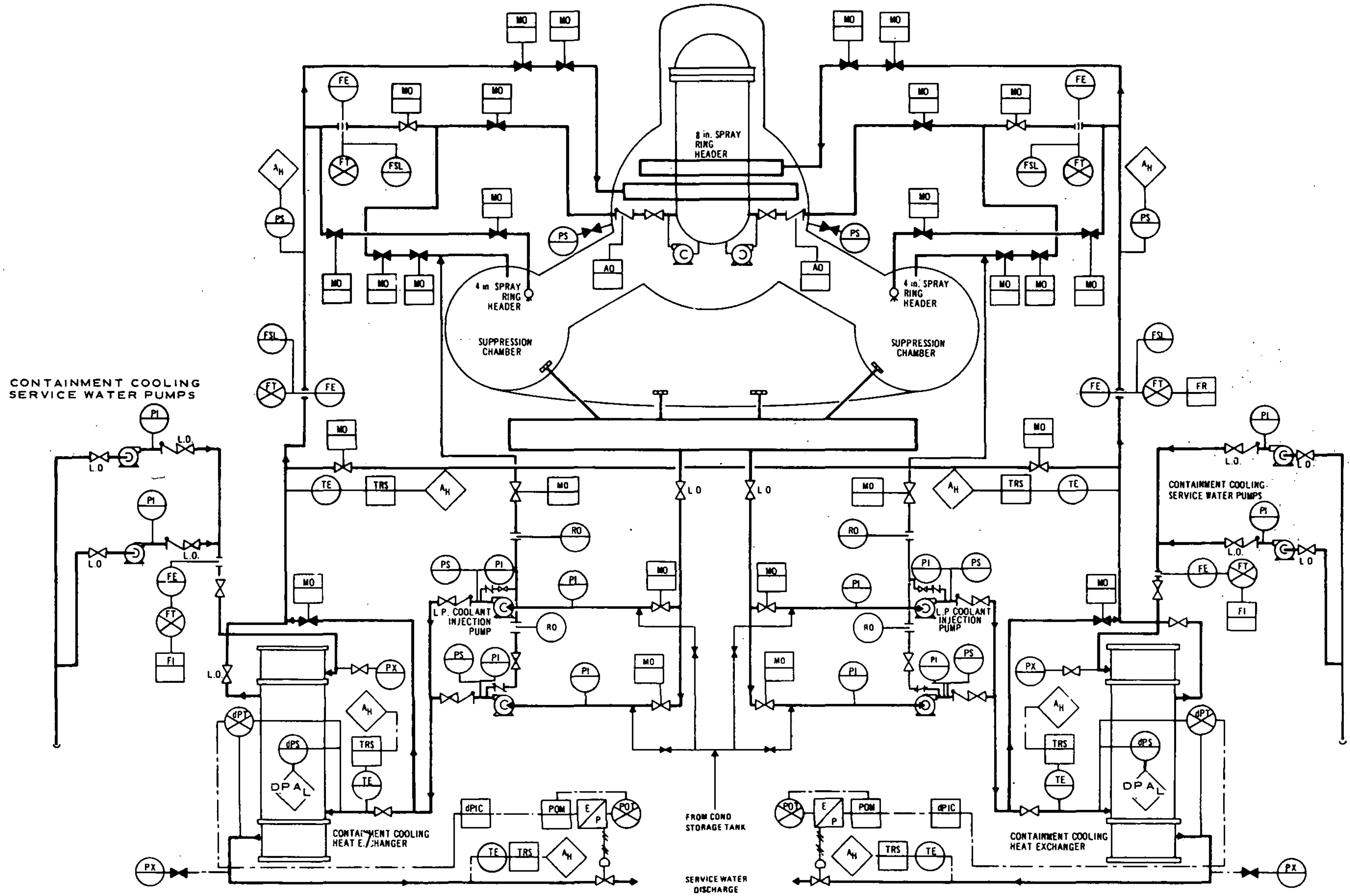


Figure 6.2.7 Low Pressure Coolant Injection/Containment Cooling System P&ID

Revised 8/7/69

53, 20

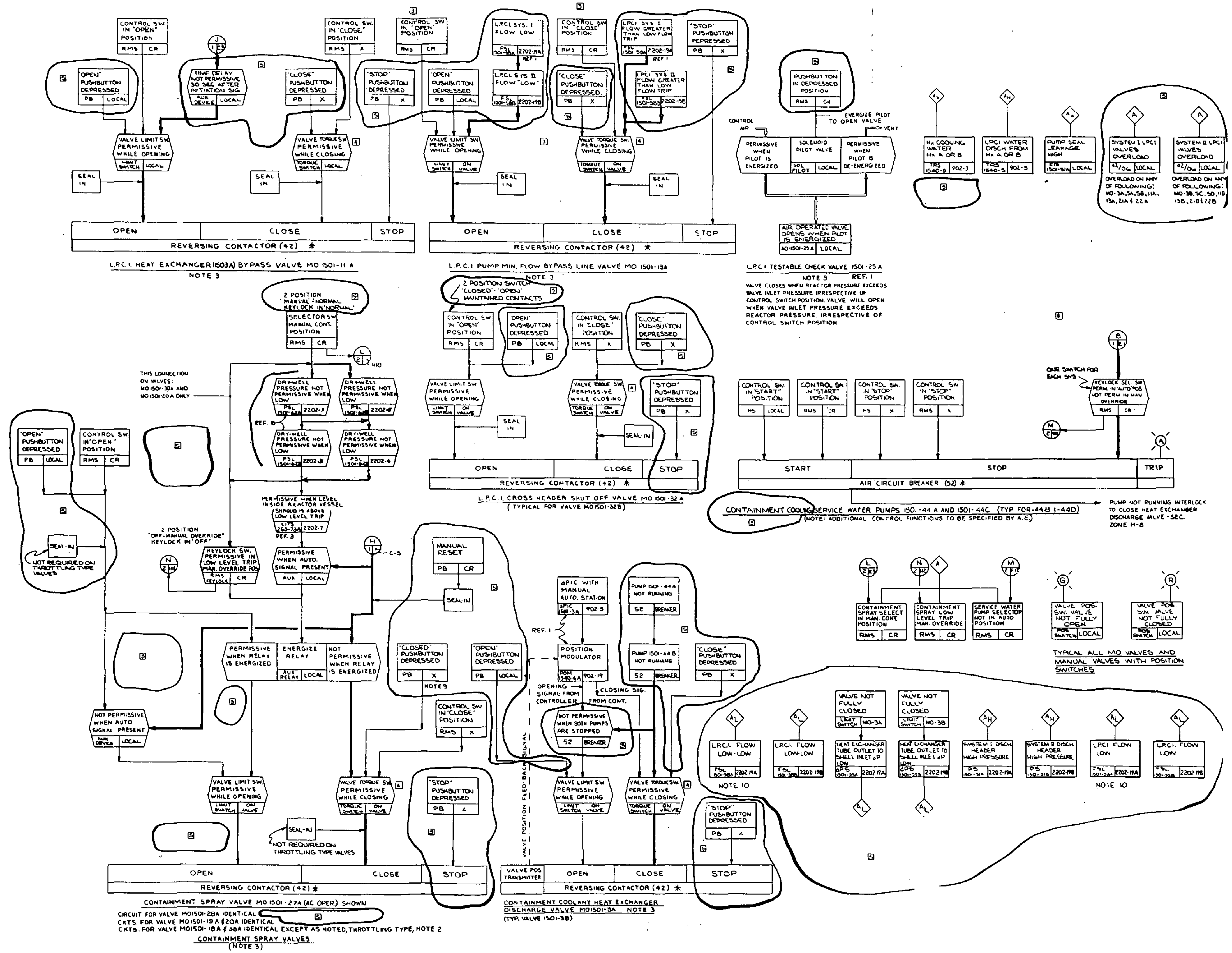


Figure 6.2.9b Low Pressure Coolant Injection/Containment Cooling System Functional Block Diagram

Revised
8/7/69

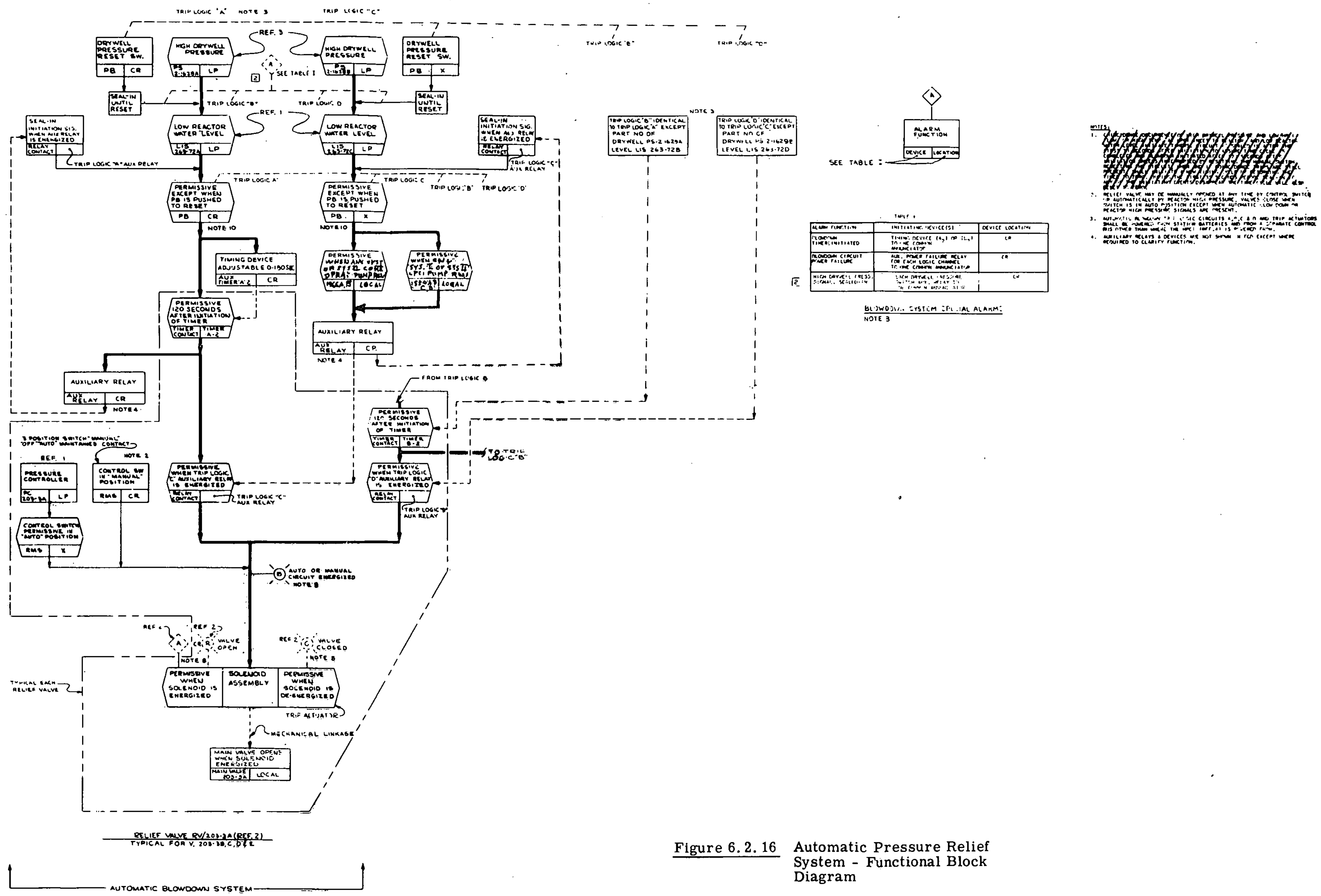


Figure 6.2.16 Automatic Pressure Relief System - Functional Block Diagram