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PALO VERDE NUCLEAR GENERATING STATION

UNITS 1, 2 & 3

TMI-2 LESSONS LEARNED IMPLEMENTATION REPORT

POOR ORIGINAL

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ARIZONA PUBLIC SERVICE COMPANY
PROJECT MANAGER AND OPERATING AGENT

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PALO VERDE NUCLEAR GENERATING STATION

UNITS 1, 2 & 3

TMI-2 LESSONS LEARNED IMPLEMENTATION REPORT

1059 14

SEPTEMBER 1979

ARIZONA PUBLIC SERVICE COMPANY

P. O. BOX 21666 · PHOENIX, ARIZONA 85036

September 28, 1979
ANPP-13905 - JMA/DBK

Director of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Re: Palo Verde Nuclear Generating Station
Units 1, 2 and 3
Docket Nos. STN 50-528/529/530

Dear Sir:

As a result of the Three Mile Island accident, Arizona Public Service Company (APS) is in the process of assuring that the lessons learned from TMI-2 are reflected in the Palo Verde Nuclear Generating Station (PVNGS) design and operation.

To accomplish this objective, the APS President and Chief Executive Officer, by a directive issued April 12, 1979, established a Safety Evaluation Task Force "to conduct a complete review and analysis of the Three Mile Island accident and its implications for the Palo Verde Nuclear Generating Station project." This task force is comprised of fourteen (14) members from knowledgeable, high levels of management of the utilities participating in the Palo Verde project, the engineer-constructor for the project, the nuclear steam supply system supplier, and turbine generator supplier to the project and the academic technical consultants to the Arizona State Senate and House legislative committees organized to review and consider the implications of the TMI accident.

The task force is exercising oversight of the on-going independent in-depth reviews, initiated since the TMI accident, of the Palo Verde design, planned administrative and operational procedures, training and qualification programs, and emergency response plans. In this connection, it is closely following the work and recommendations of the many industry and governmental organizations currently conducting similar, but more generic investigations related to the TMI accident, including particularly the NRC's Lessons Learned Task Force.

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Upon receipt of the TMI-2 Lessons Learned Task Force Status Report and Short Term Recommendations (NUREG-0578), modified as set forth in the memorandum dated August 20, 1979 from Harold R. Denton, Director, Office of Nuclear Reactor Regulation to the NRC Commissioners, the recommendations related to the design and operation of the Palo Verde project were evaluated. The results of this evaluation are incorporated in the report entitled TMI-2 Lessons Learned Implementation Report submitted herewith. The report identifies those NUREG-0578 recommendations which are incorporated into the present Palo Verde design, identifies recommendations that are not applicable to the Palo Verde design and sets forth APS' commitments to develop the analyses, design modifications, operational procedures and other measures necessary to meet such recommendations in accordance with the schedule specified in NUREG-0578. The report will be amended as appropriate to incorporate the manner in which such commitments are implemented.

Five (5) copies of this report are enclosed for your review. It is hoped that this report, in conjunction with the PVNGS Final Safety Analysis Report to be submitted in October, 1979, will assist in your timely review of the implications of TMI on PVNGS.

Respectfully submitted,

ARIZONA PUBLIC SERVICE COMPANY

By: Edwin E. Van Brunt

Edwin E. Van Brunt, Jr.
Vice President

STATE OF ARIZONA)
) ss.
County of Maricopa)

On its own behalf and as agent for all other joint participants.

Subscribed and sworn to before me this 27 day of Sept, 1979.

John M. Allen
Notary Public

My Commission expires:
My Commission Expires Jan. 23, 1983

INTRODUCTION

The Palo Verde Nuclear Generating Station NUREG-0578 Evaluation addresses the "TMI-2 Lessons Learned Task Force Status Report And Short Term Recommendations" (NUREG-0578), July 1979, as supplemented by letter dated August 20, 1979 from Harold R. Denton, Director, Office of Nuclear Reactor Regulation to the NRC Commissioners.

Each recommendation is addressed, as it applies to the Palo Verde Nuclear Generating Station (PVNGS) design and operation, in the following manner:

(1) Descriptions of the existing PVNGS design correspond with information presented in the PVNGS Final Safety Analysis Report (FSAR).

(2) Proposed modifications, evaluations and analyses of the PVNGS design, as discussed herein, will be summarized in revisions to this report.

(3) Recommendation numbering corresponds to that used in NUREG-0578.

This report will be referenced by subsequent PVNGS FSAR amendments, as appropriate, to address in the FSAR, required modifications and analyses as discussed herein.

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2.1.1 EMERGENCY POWER SUPPLY REQUIREMENTS FOR THE PRESSURIZER HEATERS, POWER OPERATED RELIEF VALVES AND BLOCK VALVES, AND PRESSURIZER LEVEL INDICATORS IN PWRs

2.1.1.a PRESSURIZER HEATER POWER SUPPLY

NUREG-0578 Position

1. The pressurizer heater power supply design shall provide the capability to supply, 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 with the emergency buses shall be accomplished through devices that have been qualified in accordance with safety-grade requirements.

PVNGS Evaluation

The PVNGS pressurizer heaters are powered as follows:

1. The pressurizer heaters (36 elements rated at 50 KW each, connected in 12 groups of 3 delta-connected elements each) have a total capacity of 1800 KW. The heaters are supplied 480 volt power from the non-Class IE and Class IE distribution systems as follows:

	Number of Heaters	Capacity (KW)	480V Bus	IE Bus
A.	5-3 element groups	750	NGN-L11	No
B.	5-3 element groups	750	NGN-L12	No
C.	1-3 element group	150	PGA-L33	Yes
D.	1-3 element group	150	PGB-L32	Yes

Maintenance of natural circulation at hot standby conditions requires 150 KW of heaters. These heaters are immediately available from either of two redundant Class IE buses, fed independently by Safety Trains A and B. Controls for these heaters are likewise independent. As the Class IE buses are a part of the ESF distribution system, they are fed either from the offsite power system or the diesel generators when offsite power is not available.

Heaters supplied from the non-Class IE power system are fed from the offsite power system and are available for manual interconnection to the diesel generators when offsite power is not available.

2. Heaters fed from the Class IE power system (150KW) are automatically shed upon receipt of a Loss of Offsite Power (LOP) or a Safety Injection Activation Signal (SIAS). They may subsequently be manually reconnected to the ESF buses without shedding of any loads from the Class IE buses.

Heaters fed from non-Class IE buses can also be manually connected to the Class IE buses for powering by either the offsite power system or the diesel generators through the ESF transformers. These heaters are available in 150 KW blocks (0-150 KW in the case of the two proportional groups). Sufficient margin exists in the diesel generators for at least an additional 300 KW of heaters to be backfed from each Class IE bus without the shedding of selected emergency loads.

3. If an SIAS or LOP signal are received during an accident or severe transient (shedding the heaters connected to the Class IE buses), the heaters are available for immediate reconnection to the Class IE bus from the control room.

4. The redundant pressurizer heaters required for maintenance of natural circulation at hot standby are fed from Class IE buses via Class IE load side breakers qualified in accordance with safety-grade requirements.

The PVNGS pressurizer heaters, as presently designed, comply with the recommendations of NUREG-0578, July 1979.

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2.1.1 EMERGENCY POWER SUPPLY REQUIREMENTS FOR THE PRESSURIZER HEATERS, POWER OPERATED RELIEF VALVES AND BLOCK VALVES, AND PRESSURIZER LEVEL INDICATORS IN PWRs

2.1.1.b POWER SUPPLY FOR PRESSURIZER RELIEF AND BLOCK VALVES AND PRESSURIZER LEVEL INDICATORS

NUREG-0578 Position

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. These 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.

PVNGS Evaluation

1. PVNGS does not use PORVs and block valves.

2. PVNGS does not use PORVs and block valves.

3. PVNGS does not use PORVs and block valves.

4. Two channels of Class IE level instrumentation are provided for PVNGS. Pressurizer level channels L-110X and L-110Y (refer to Figure 2.1.11-2) are indicated in the control room. Channel L-110X is also recorded in the control room.

   The pressurizer level instrumentation is powered from 120V AC Class IE instrument buses E-PNA-D25 and E-PNB-D26 (refer to Figure 2.1.1-1). These buses are normally powered through inverters from Class IE batteries. The Class IE battery chargers are powered from offsite power or from the diesel generators when offsite power is not available.

The pressurizer level indicators, as presently designed, comply with the recommendations of NUREG-0578, July 1979.

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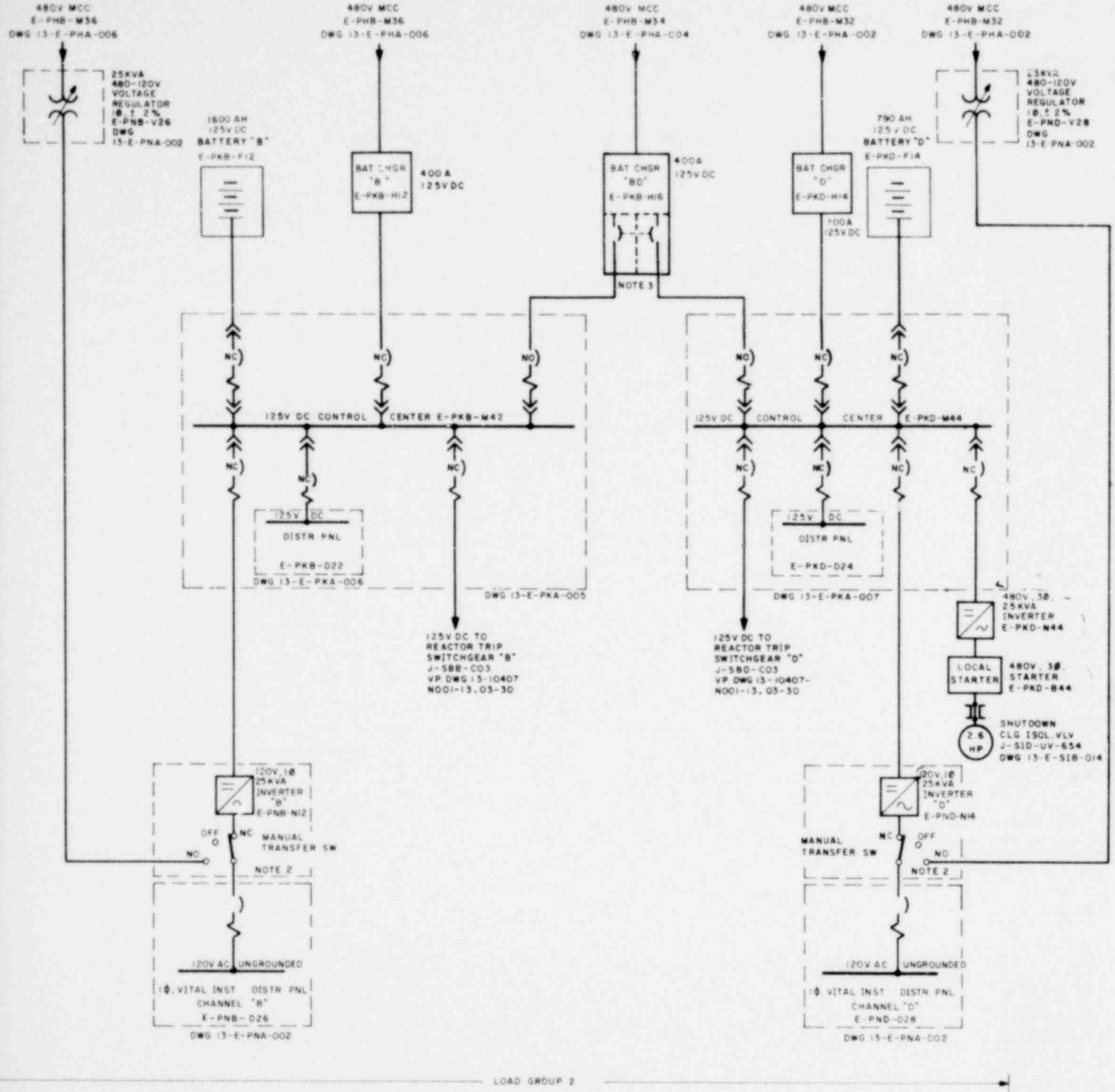

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13-E-PKA-001 REV 1

Palo Verde Nuclear Generating Station

SINGLE LINE DIAGRAM
DC POWER SYSTEM

Figure 2.1.1-1

NOTES:

1. FOR LEGEND & GENERAL NOTES REFER TO DWGS 13-E-ZZB-001 THROUGH 007
2. THIS INDICATES MANUAL TWO POLE THREE POSITION TRANSFER SWITCH
3. OUTPUT SWITCHES IN BATTERY CHARGERS "AC" & "BD" ARE MECHANICALLY INTERLOCKED TO OFFER THE POSSIBILITY OF EITHER BOTH OPEN OR ONE CLOSED AND ONE OPEN AT ANY TIME

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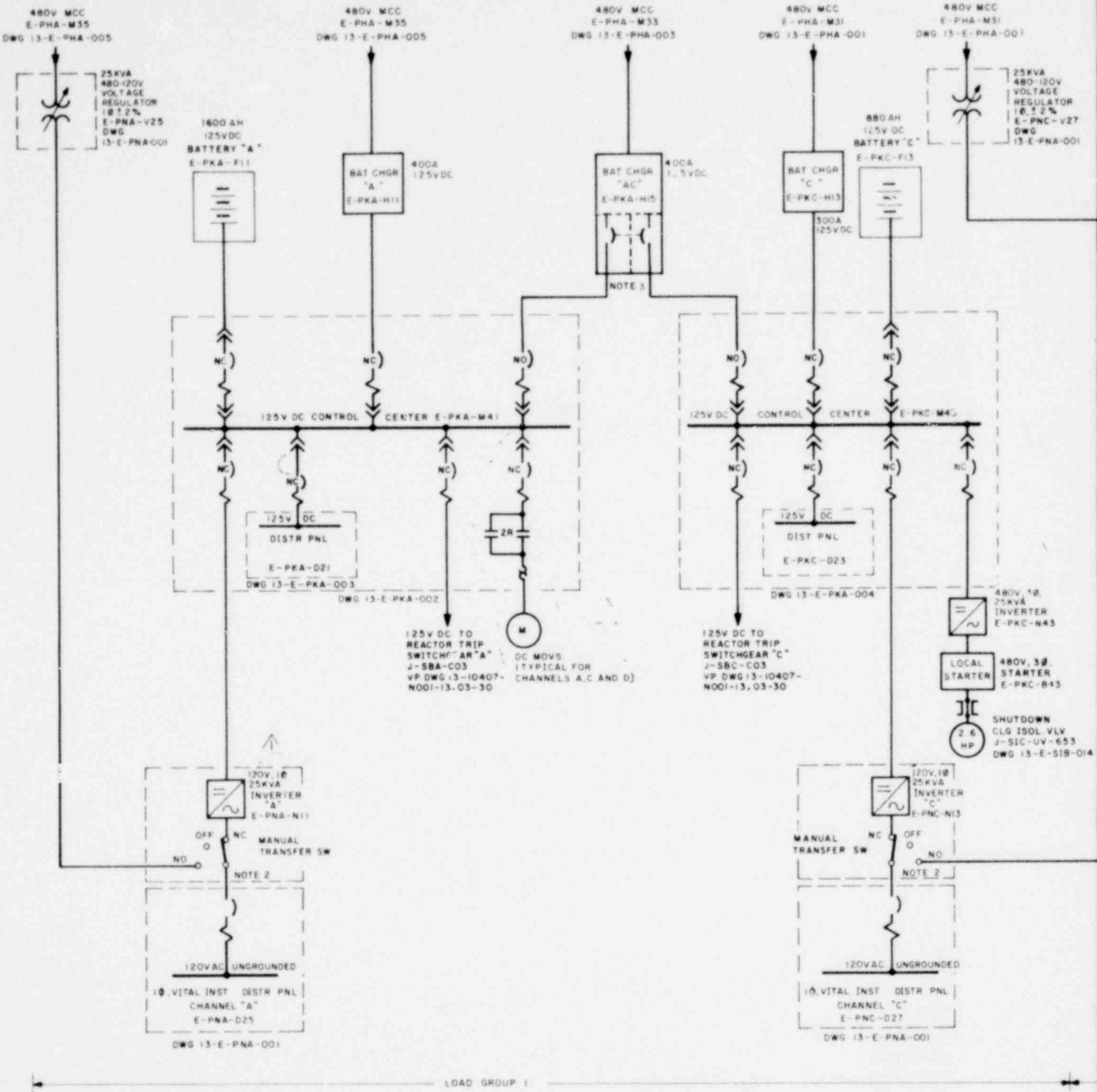

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2.1.2

2.1.2 PERFORMANCE TESTING FOR BWR AND PWR RELIEF AND SAFETY VALVES

NUREG-0578 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. The licensees and applicants shall determine the expected valve operating conditions through the use of analyses of accidents and anticipated operational occurrences referenced in Regulatory Guide 1.70, Revision 2. The single failures applied to these analyses shall be chosen so that the dynamic forces on the safety and relief valves are maximized. Test pressures shall be the highest predicted by conventional safety analysis procedures. Reactor coolant system relief and safety valve qualification shall include qualification of associated control circuitry piping and supports, as well as the valves themselves.

PVNGS Evaluation

PVNGS does not utilize pressure operated relief valves.

An analysis will be performed to determine the expected code safety valve operating conditions based upon the accidents and anticipated operational occurrences described in the PVNGS FSAR.

When this analysis is completed, generic code safety valve qualification tests will be performed, as necessary, to qualify the valve for expected operating conditions following an accident or anticipated operational occurrence. In addition, the discharge piping and supports will be analyzed for their expected operating conditions. Performance of qualification tests is dependent on the availability of test facilities capable of simulating the expected operating conditions.

The results of these analyses and tests will be provided prior to PVNGS Unit 1 Operating License.

2.1.3

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2.1.3.a

Direct Indication of Power-Operated Relief Valve and Safety Valve Position for PWR's and BWR's

2.1.3 INFORMATION TO AID OPERATORS IN ACCIDENT DIAGNOSIS AND CONTROL

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

NUREG-0578 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.

PVNGS Evaluation

PVNGS does not utilize power operated relief valves.

The PVNGS pressurizer code safety valves are headered into the reactor drain tank (RDT), inside containment. Upstream of the common header each code safety valve is monitored for seat leakage by an in-line RTD (refer to Figure 2.1.11-2). Indirect indication of code safety valve operation is provided by RDT pressure, pressurizer pressure and pressurizer level instrumentation.

An evaluation will be performed to determine if the multiple indirect indications of code safety valve status provided in the PVNGS design provide unambiguous indication that a small break area LOCA is in progress.

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If the evaluation determines that direct valve status indication in the control room is required (e.g. by acoustic techniques, flow measurement or valve position indication), it will be incorporated in the PVNGS design prior to Unit 1 Operating License.

Results of this evaluation will be provided by January 1980.

2.1.3.b

Instrumentation for Detection of Inadequate Core Cooling in PWR's and BWR's

2.1.3 INFORMATION TO AID OPERATORS IN DIAGNOSIS AND CONTROL

2.1.3.b INSTRUMENTATION FOR DETECTION OF INADEQUATE CORE COOLING IN PWRs AND BWRs.

NUREG-0578 Position

1. 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 analysis 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".

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.

2. 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.

PVNGS Evaluation

Information to diagnose inadequate core cooling is as follows:

1. Prior to PVNGS Unit 1 Operating License, procedures will be developed to aid the operator in recognition of inadequate core cooling. These procedures will utilize existing redundant safety related instrumentation such as:

A.	Pressurizer Pressure	1500-2500 psia
B.	Pressurizer Pressure	0-3000 psia
C.	Pressurizer Level	0-100%
D.	Coolant Temperature	
	- Hot Leg	375-675F
	- Cold Leg	465-615F

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These procedures will be based upon analyses performed as discussed in Section 2.1.9. Redundant subcooling meters will be installed prior to PVNGS Unit 1 Operating License to provide additional indication of primary coolant saturation condition. The procedures discussed in Item 1 will incorporate the use of primary coolant saturation indication.

2. Upon completion of the analyses described in Section 2.1.9, an evaluation will be performed to determine the need, if any, for additional instrumentation, such as reactor vessel level indication. Any instrumentation determined to be necessary will be installed prior to PVNGS Unit 1 Operating License.

2.1.4

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2.1.4 CONTAINMENT ISOLATION PROVISIONS FOR PWRs AND BWRs

NUREG-0578 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.
2. All plants shall give careful reconsideration to the definition of essential and non-essential systems, shall identify each system determined to be essential, shall identify each system determined to be non-essential, shall describe the basis for selection of each essential system, shall modify their containment isolation designs accordingly, and shall report the results of the re-evaluation to the NRC.
3. All non-essential 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 re-opening of containment isolation valves. Reopening of containment isolation valves shall require deliberate operator action.

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PVNGS Evaluation

The PVNGS containment isolation system is designed as follows:

1. As required by SRP 6.2.4, a containment isolation signal is diversely generated by either a high containment pressure signal (5 psig) or a low pressurizer pressure signal (1600 psig). These parameters additionally generate a safety injection actuation signal. The power access purge is additionally isolated by high containment purge radioactivity.

2. A review of the PVNGS containment isolation system will be performed to determine that no essential systems are automatically isolated by a containment isolation actuation signal(CIAS). Essential systems will be selectively isolated by the operator after their use is no longer required.

   The results of this review will be provided to the NRC staff by January 1980. Plant modifications, if any, will be performed prior to PVNGS Unit 1 Operating License.

3. All non-essential systems penetrating the PVNGS containment are isolated by CIAS.

4. Resetting of a CIAS does not result in the automatic opening of containment isolation valves. Re-opening requires operator action for each valve.

2.1.5

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2.1.5.a

Dedicated Penetrations for External Recombiners or Post-Accident Purge Systems

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2.1.5 POST ACCIDENT HYDROGEN CONTROL SYSTEMS FOR PWR AND BWR CONTAINMENTS

2.1.5.a DEDICATED PENETRATIONS FOR EXTERNAL RECOMBINER OR POST-ACCIDENT EXTERNAL PURGE SYSTEM

NUREG-0578 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 meet the redundancy and single failure requirements of General Design Criteria 54 and 56 of Appendix A to 10 CFR Part 50, and that are sized to satisfy the flow requirements of the recombiner or purge system.

PVNGS Evaluation

Combustible gas concentration in the PVNGS containment building is maintained by the Containment Hydrogen Control System.

The PVNGS Containment Hydrogen Control System (refer to Figure 2.1.5-1) shares no equipment or penetrations with the normal containment purge system and includes:

1. Two redundant and independent hydrogen gas analyzers with separate piping, valves, controls and instrumentations (refer to Section 2.1.10).

2. Redundant and independent mobile, hydrogen recombiners (two for the PVNGS site) with separate supply and return lines, utilizing separate penetrations for each line.

3. Redundant and independent supply and return line isolation valving (one valve inside containment and one outside containment) that meets the requirements of 10 CFR 50, Appendix A, Criterion 54 and 56.

4. Dedicated containment penetrations sized for maximum recombiner flow requirements.

The PVNGS Containment Hydrogen Control System, as currently designed, provides dedicated penetrations and isolation systems that meet redundancy and single failure requirements as recommended by NUREG-0578, July 1979.

2.1.5.b

Inerting BWR Containments

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2.1.5 POST ACCIDENT HYDROGEN CONTROL SYSTEMS FOR PWR AND BWR CONTAINMENTS

2.1.5.b INERTING BWR CONTAINMENTS

NUREG-0578 Position

It shall be required that the Vermont Yankee and Hatch 2 Mark I BWR containments be inerted in a manner similar to other operating BWR plants. Inerting shall also be required for near term OL licensing of Mark I and Mark II BWRs.

PVNGS Evaluation

PVNGS utilizes Combustion Engineering, Inc. System 80^{tm} PWRs with a containment with 2.6×10^6 ft^3 net free volume. Inerting of the PVNGS containment is not required.

2.1.5.c

Capability to Install Hydrogen Recombiner at each Light Water Nuclear Power Plant

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2.1.5. <u>POST ACCIDENT HYDROGEN CONTROL SYSTEMS FOR PWR AND BWR CONTAINMENTS</u>

2.1.5.c CAPABILITY TO INSTALL HYDROGEN RECOMBINER AT EACH LIGHT WATER NUCLEAR POWER PLANT

<u>NUREG-0578 Position</u> (Minority View)

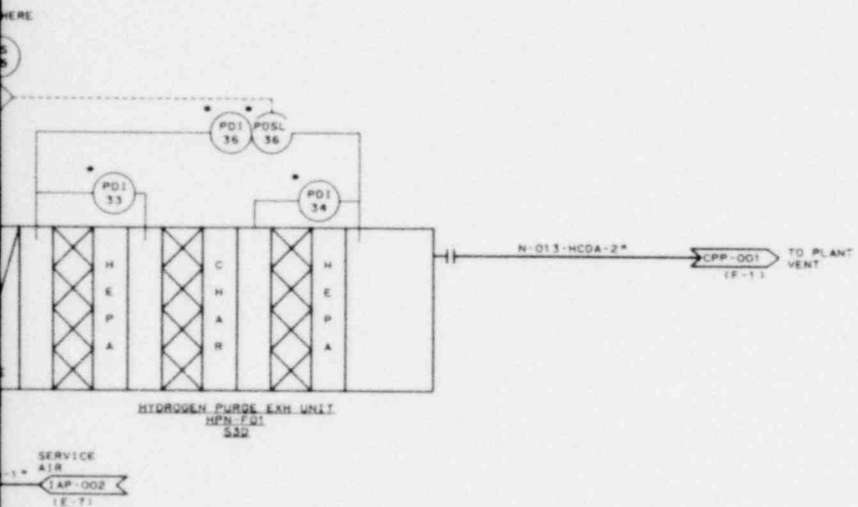
1. All licensees of light water reactor plants shall have the capability to obtain and install recombiners in their plants within a few days following an accident if containment access is impaired and if such a system is needed for long-term post-accident combustible gas control.
2. The procedures and 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.

<u>PVNGS Evaluation</u>

The PVNGS hydrogen recombiners are designed as follows:

1. PVNGS will have available, onsite, two mobile hydrogen recombiners designed to limit the concentration of hydrogen in containment. These hydrogen recombiners are capable of being installed in the affected unit within 24 hours following an accident.

2. Installation procedures for the hydrogen recombiners will be developed and shielding requirements re-analyzed prior to PVNGS Unit 1 Operating License.



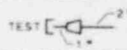
3. THE SYSTEM DESIGNATOR HP IS TO PRECEDE ALL LINE, VALVE, AND INSTRUMENT NUMBERS SHOWN ON THIS DRAWING UNLESS OTHERWISE INDICATED.

4. ELECTRICAL HEAT TRACING SHALL BE POWERED FROM IE SOURCE OF POWER.

NOTES:

1. INSTALL VALVES WITH FLOW DIRECTION ARROW IN DIRECTION NOTED TO CLOSE AGAINST CONTAINMENT DESIGN PRESSURE.

2. PIPE SHALL BE THREADED TO ACCEPT A 2" X 1" CAPPED TEST REDUCER AS SHOWN BELOW.



DURING NORMAL OPERATION WHEN END OF PIPE IS OPEN, PROVIDE SCREW COUPLINGS FOR PROTECTION OF THE THREAD.

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13-M-HPP-001 REV 3

Palo Verde Nuclear Generating Station

COMBUSTIBLE GAS CONTROL SYSTEM

Figure 2.1.5-1

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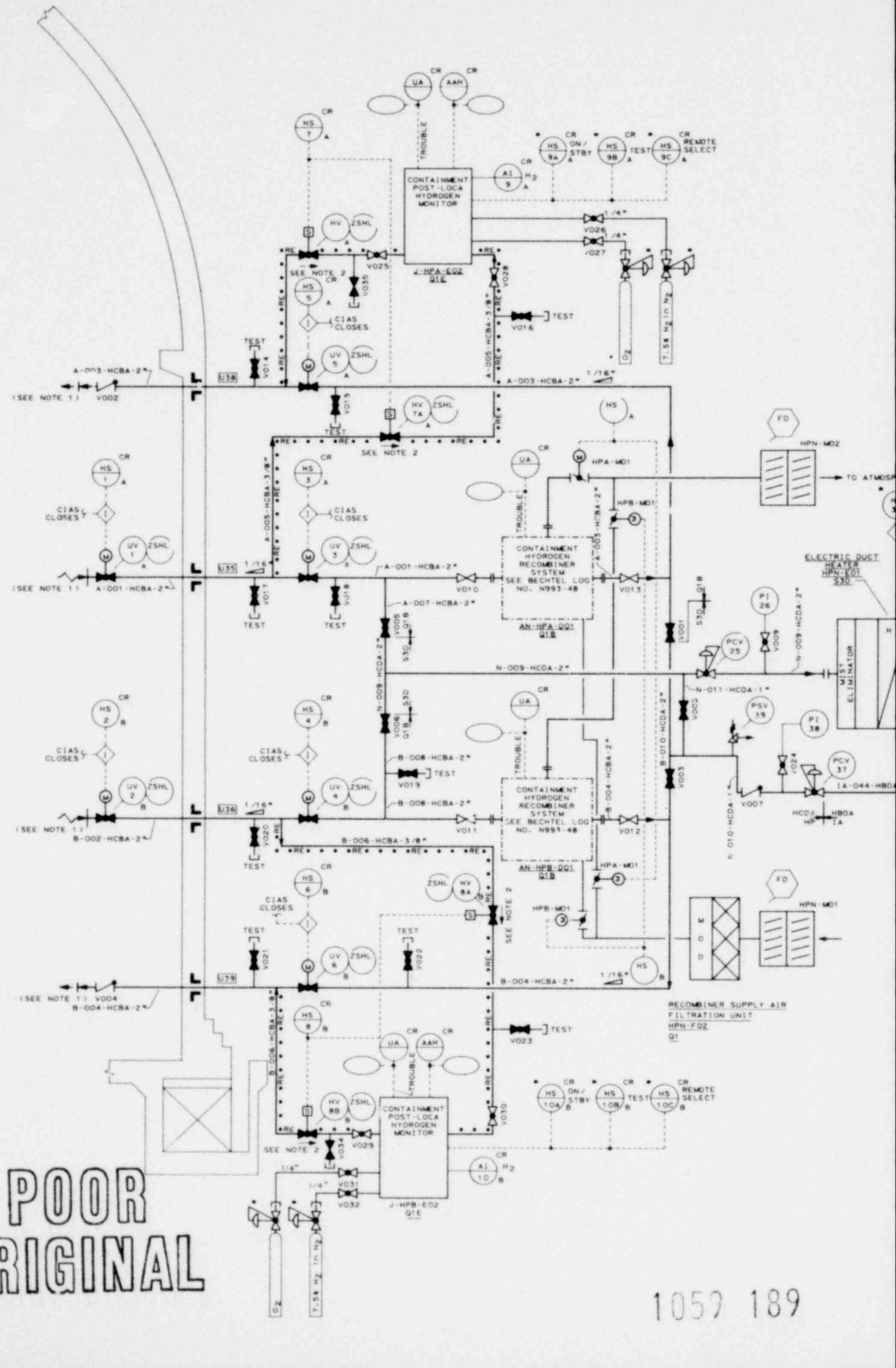

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2.1.6

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2.1.6.a

Integrity of Systems Outside Containment Likely to Contain Radioactive Materials (Engineered Safety Systems and Auxiliary Systems) for PWR's and BWR's

2.1.6 POST ACCIDENT CONTROL OF RADIATION IN SYSTEMS OUTSIDE CONTAINMENT OF PWRs AND BWRs

2.1.6.a INTEGRITY OF SYSTEMS OUTSIDE CONTAINMENT LIKELY TO CONTAIN RADIOACTIVE MATERIALS (ENGINEERED SAFETY SYSTEMS AND AUXILIARY SYSTEMS) FOR PWRs AND BWRs

NUREG-0578 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:

1. Immediate Leak Reduction

    a. Implement all practical leak reduction measures for all systems that could carry radioactive fluids outside of containment.

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

2. Continuing Leak Reduction

    Establish and implement a program of preventive maintenance to reduce leakage to as-low-as-practical levels in systems outside containment likely to contain highly radioactive fluid during a serious transient or accident.

PVNGS Evaluation

The PVNGS program of preventive maintenance will reduce leakage to as low as practical. The program will include:

1. In-service inspection of ASME Class 2 and 3 systems in accordance with AMSE Boiler and Pressure Vessel Code Section XI, including pressure tests.

2. Alarming of abnormal airborne radioactivity levels by the radiation monitoring system for gaseous radwaste system leakage.

3. Routine monitoring of plant components.

Undesirable leakage identified during tests or inspections will be reduced to a level as low as is practical.

2.1.6.b

Design Review of Plant Shielding & Environmental Qualification of Equipment for Spaces/Systems Which May be Used in Post-Accident Operations

2.1.6 POST ACCIDENT CONTROL OF RADIATION IN SYSTEMS OUTSIDE CONTAINMENT OF PWRs AND BWRs

2.1.6.b DESIGN REVIEW OF PLANT SHIELDING AND ENVIRONMENTAL QUALIFICATION OF EQUIPMENT FOR SPACES/SYSTEMS WHICH MAY BE USED IN OPERATIONS

NUREG-0578 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 and 100% of the core noble gas inventory 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.

PVNGS Evaluation

As part of the PVNGS design process, shielding in areas requiring personnel access for accident mitigation is designed such that access is not unduly limited.
In addition, safety related equipment is qualified for the maximum expected equipment life dose.

An analysis of the PVNGS shielding design criteria, now based on ALARA, will be performed to determine if the results of TMI invalidate a y of the PVNGS shielding criteria. If, as a result of this analysis, the shielding design criteria in a plant area requires modification, the shielding in that area will be redesigned based on revised criteria. Any increase in permanent or temporary shielding, redesign or procedural control determined to be necessary will be completed prior to PVNGS Unit 1 Operating License. Refer to Section 2.1.8.a for a discussion of sampling system shielding.

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2.1.7

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2.1.7.a

Automatic Initiation of the Auxiliary Feedwater System for PWR's

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2.1.7 IMPROVED AUXILIARY FEEDWATER SYSTEM RELIABILITY FOR PWRs

2.1.7.a AUTOMATIC INITIATION OF THE AUXILIARY FEEDWATER SYSTEM FOR PWRs

NUREG-0578 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 axuiliary 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.

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6. The AC motor-driven pumps and valves in the auxiliary feedwater system shall be included in the automatic actuation (simultaneous and/or sequential) of the loads to 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.

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

PVNGS Evaluation

1. The Auxiliary Feedwater System shown in Figure 2.1.7-1 consists of two Seismic Category I pumps and their associated valves and one non-Seismic Category I pump and its associated valves.

   The Seismic Category I portion of the auxiliary feedwater system is provided to automatically initiate residual heat removal capability during emergency conditions, such as a steam line rupture, loss of normal feedwater, or loss of offsite and normal onsite power.

The Seismic Category I portion of the auxiliary feedwater system is automatically actuated by an Auxiliary Feedwater Actuation Signal (AFAS) from the Engineered Safety Features Actuation System (ESFAS). The AFAS is initiated for each steam generator by a low steam generator level coincident with a "not ruptured" calculated signal for that steam generator (refer to Figure 2.1.7-2).

The AFAS logic determines whether a steam generator is not intact in the event of a secondary system break by sensing:

- The steam generator has initiated a low water level trip.
- The steam generator pressure is less than the other by a predetermined value.
- The other steam generator has been calculated as not being ruptured.

The non-Seismic Category I portion of the auxiliary feedwater system is provided for normal non-emergency operation during startup, cooldown, and hot standby.

2. The Seismic Category I portion of the auxiliary feedwater system is composed of components in two separate and diverse load groups (i.e., load group 1 and load group 2. Each of the four auxiliary feedwater valves associated with each steam generator is automatically actuated in such a manner that no single failure can prevent either the supply of auxiliary feedwater to an intact steam generator or the

1059 201

isolation of auxiliary feedwater from a ruptured steam generator. Load group 2 powers the Seismic Category I motor-driven auxiliary feedwater pump and its associated valves and controls. Load group 1 supplies dc power to the steam-driven turbine controls and the valves associated with the turbine-driven auxiliary feedwater pump. No ac power is required for support of the turbine-driven auxiliary feedwater train. The instrumentation and controls of the components and equipment in load group 1 are physically and electrically separate and independent of the instrumentation and controls of the components and equipment in load group 2. This separation is maintained such that both trains are not terminated on common logic circuits.

3. Provisions are made to permit periodic testing of the auxiliary feedwater initiation signals and circuitry. These tests cover the trip actions from sensor input through the protection system and actuation devices. The system test does not interfere with the system protective function. The testing system meets the criteria of IEEE-338-1971, "IEEE Trial-Use Criteria for the Periodic Testing of Nuclear Power Generating Station Protective Systems," and Regulatory Guide 1.22,

"Periodic Testing of Protection System Actuator Functions." Testing is performed in accordance with the surveillance test requirements of the PVNGS Technical Specifications.

4. AFAS circuits are a part of the Engineered Safety Features Systems. The initiating signals and circuits are powered from Class IE buses in separate load groups as discussed in Item 2. The initiating sensors are powered from separate and redundant Class IE nuclear instrumentation and control panels, each of which is supplied by either offsite power or the diesel generators when offsite power is not available and is backed up by Class IE batteries.

5. Manual initiation capability for each auxiliary feedwater train exists in the control room. Control of individual auxiliary feedwater system components is also available in the control room. No single failure in the manual initiation portion of the circuit can result in the loss of auxiliary feedwater system function.

6. The Seismic Category I AC motor-driven pump and power operated valves in its train are automatically and sequentially loaded on the diesel generator bus upon loss of offsite power.

7. Failure of the AFAS will not result in the loss of manual capability to control auxiliary feedwater system components from the control room.

8. AFAS circuits are Class IE as presently designed.

The PVNGS auxiliary feedwater system, as presently designed, complies with the recommendations of NUREG-0578, July 1979.

1059 204

2.1.7.b

Auxiliary Feedwater Flow Indication to Steam Generators for PWR's

POOR ORIGINAL

1059 205

2.1.7 IMPROVED AUXILIARY FEEDWATER SYSTEM RELIABILITY FOR PWRs

2.1.7.b AUXILIARY FEEDWATER FLOW INDICATION TO STEAM GENERATOR FOR PWRs

NUREG-0578 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.

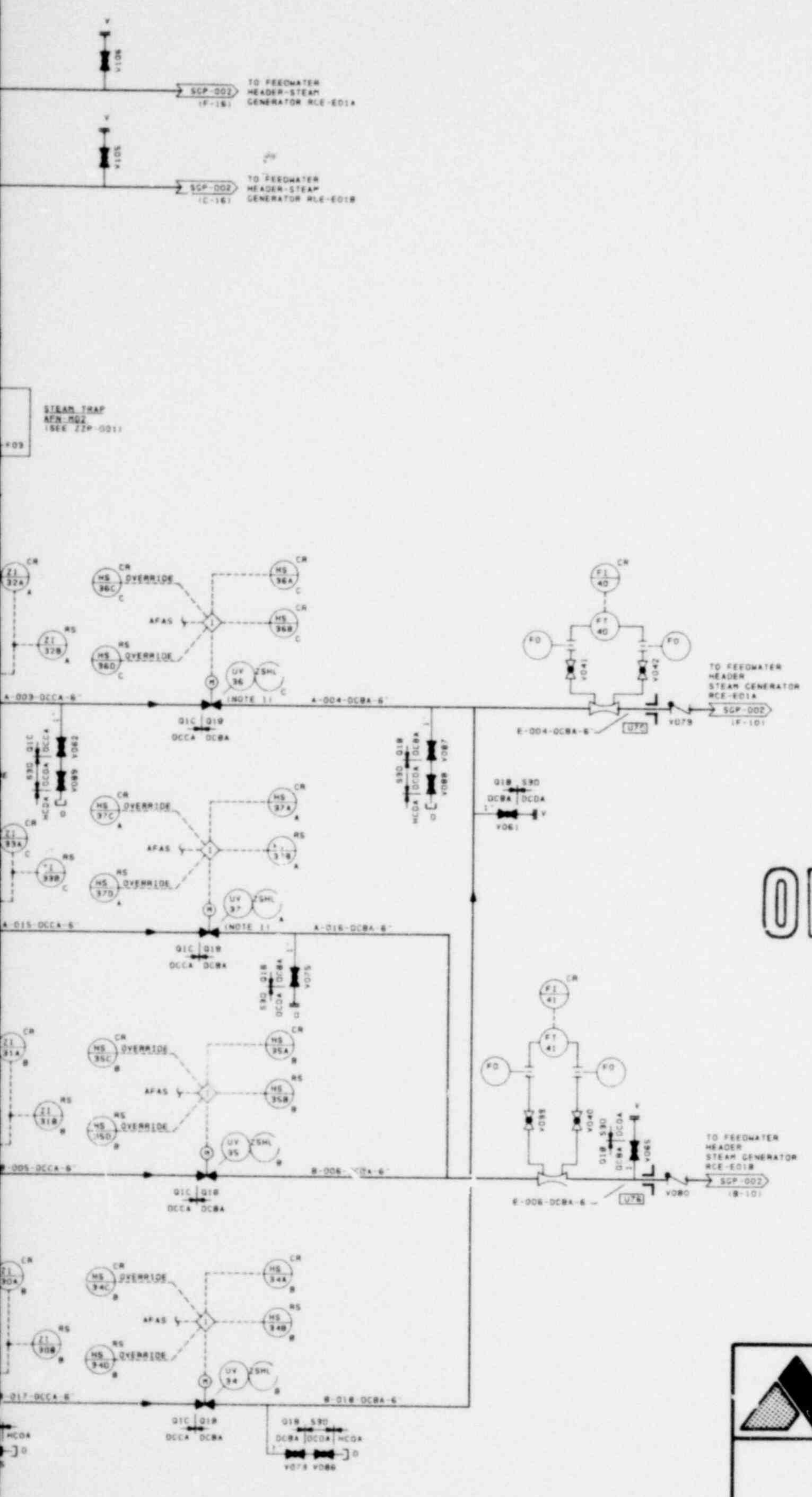
2. The auxiliary feedwater flow instrument channels shall be powered from the emergency buses consistent with satisfying the emergency power diversity requirements of the auxiliary feedwater system set forth in Auxiliary Systems Branch Technical Position 10-1 of the Standard Review Plan, Section 10.4.9.

PVNGS Evaluation

1. Monitoring of both trains of auxiliary feedwater flow is provided at PVNGS. These flow indicator channels are made from highly reliable non-Class IE components and are displayed on the

main control boards. Class IE (safety grade) pressure indicators located upstream of the manual block valves are also provided. Pressure indicator PI-18A and flow indicator FI-40 monitor flow from the train A turbine-driven auxiliary feedwater pump (refer to Figure 2.1.7-1). Pressure indicator PI-17A and flow indicator FI-41 monitor the flow from the train B motor-driven auxiliary feedwater pump.

2. The safety grade pressure indication channels are powered from their associated trains redundant Class IE buses. The instrument bus powering the auxiliary feedwater flow instrumentation is normally powered from a non-Class IE 480 V motor control center (MCC). Upon loss of this MCC the instrument bus is automatically transferred to a Class IE MCC via a Class IE isolation transformer. This Class IE MCC is powered from offsite power or from its diesel generator when offsite power is not available.



POOR ORIGINAL

NOTES:

1. VALVE HAS 125 VDC MOTOR ACTUATOR.

2. LINE IS Q1C TO FIRST FIELD WELD OUTSIDE MAIN STEAM SUPPORT STRUCTURE.

THE S A DESIGNATOR AF IS TO PRECEDE ALL LI K VALVE AND INSTRUMENT NUMBERS SHOWN ON THIS DRAWING UNLESS OTHERWISE INDICATED.

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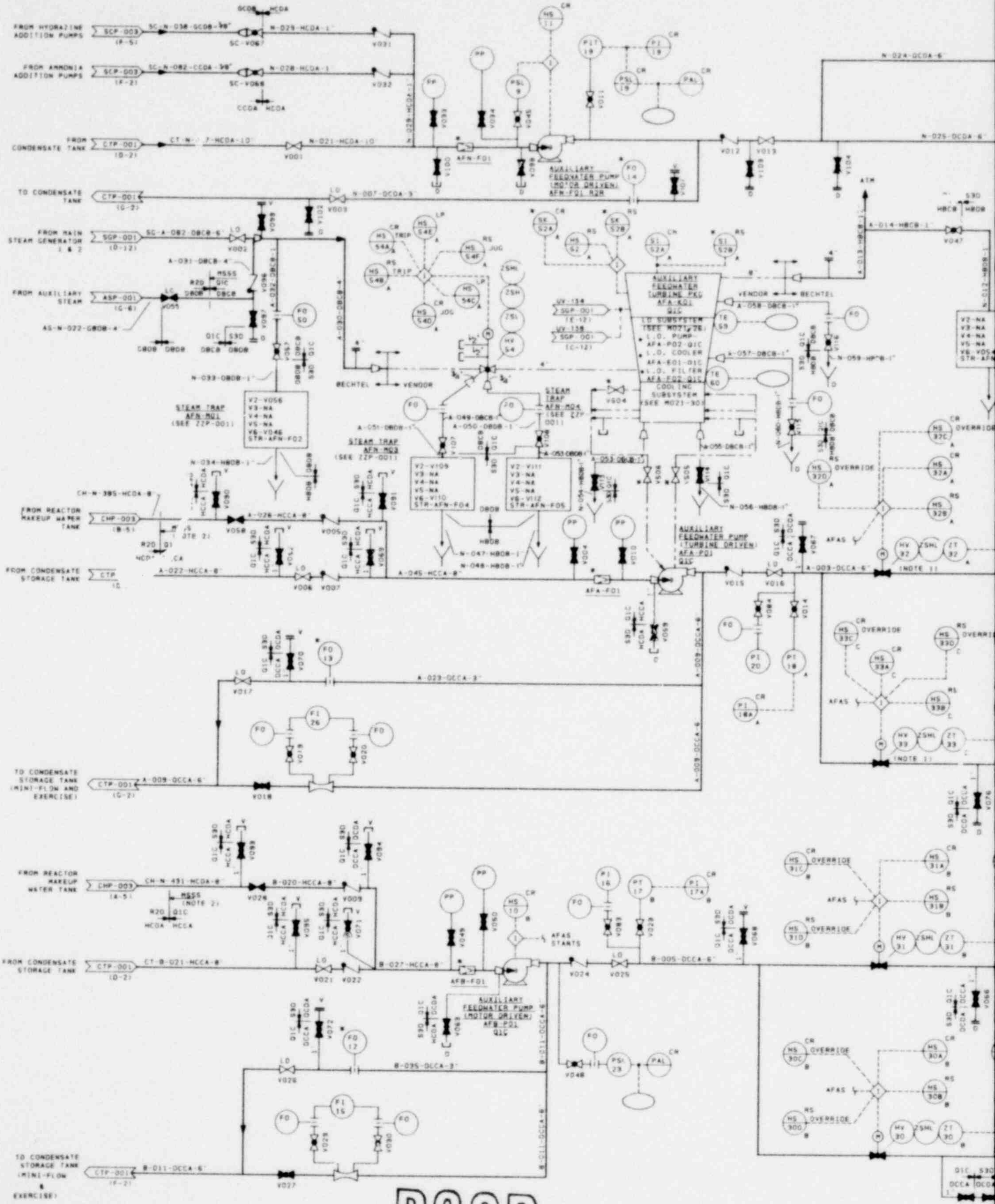
Palo Verde Nuclear Generating Station

AUXILIARY FEEDWATER SYSTEM

Figure 2.1.7-1

1059 208



POOR ORIGINAL

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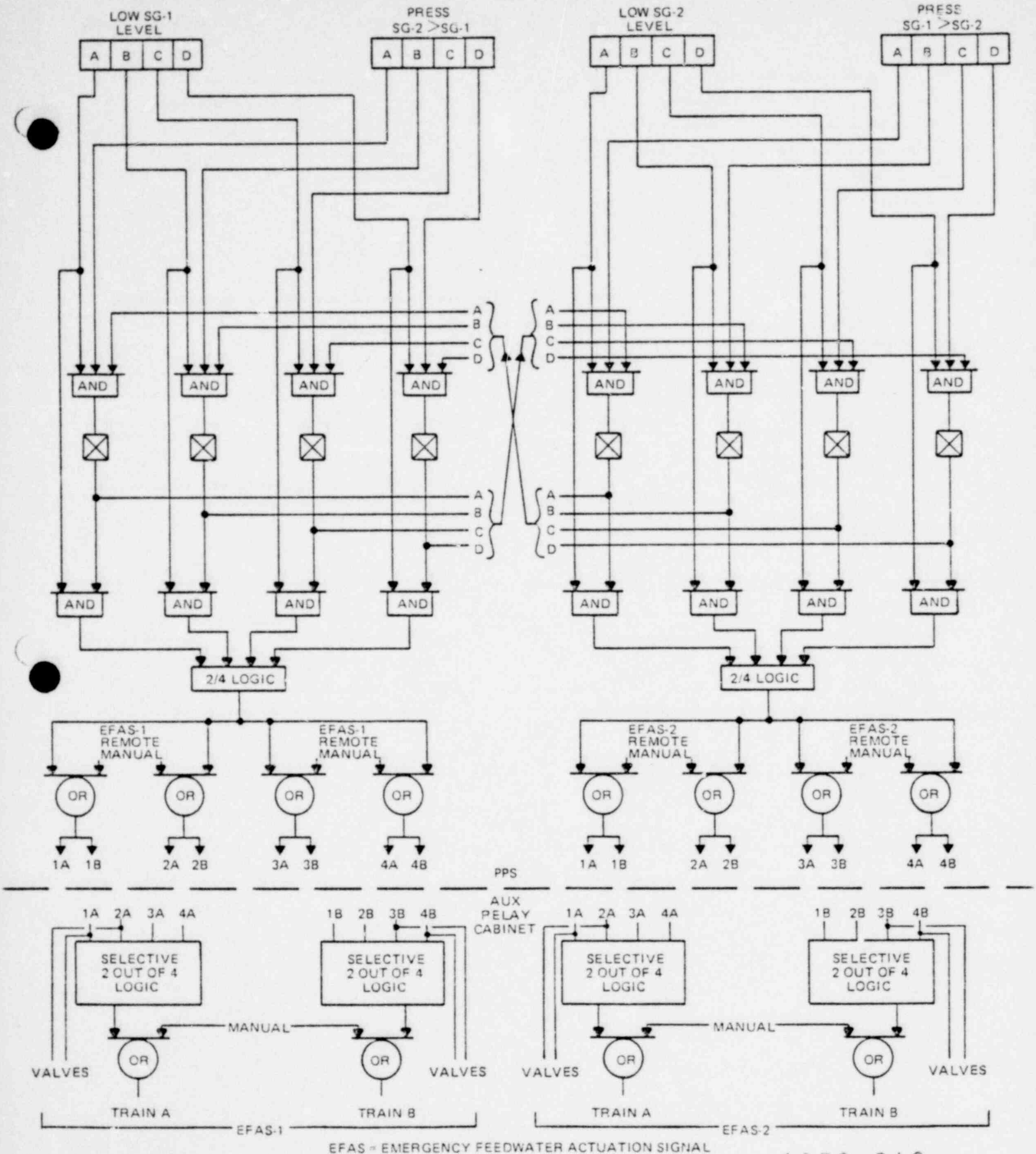

1059 210

C - E SYSTEM 80	ESFAS SIGNAL LOGIC (EFAS 1, EFAS 2)	Figure 2.1.7-2

2.1.8

2.1.8.a

Improved Post-Accident Sampling Capability

POOR ORIGINAL

1059 212

2.1.8 INSTRUCTIONS TO FOLLOW T'E COURSE OF AN ACCIDENT

2.1.8.a IMPROVED POST-ACCIDENT SAMPLING CAPABILITY

NUREG-0578 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 that 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 buildi.g 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.

PVNGS Evaluation

An analysis of the design and operation of the reactor coolant and containment atmosphere sampling systems will be performed to ensure that samples can be obtained promptly (less than 1 hour) during accident conditions without exposure to any individual to greater than 3 and 18-3/4 Rems to the whole body or extremities, respectively. Any design or procedural changes will be implemented prior to PVNGS Unit 1 Operating License.

1059 214

An analysis of PVNGS spectrum analysis facilities will be performed to ensure that the design allows prompt quantification (less than 2 hours) of previously prepared radioisotope samples that may be indicative of core damage. The assumed reactor coolant spectrum will correspond to a Regulatory Guide 1.3 or 1.4 release.

PVNGS chemical analysis facilities will be designed to allow prompt analysis of boron concentration (less than 1 hour) and chloride concentration (less than 8 hours) assuming a radioactive sample corresponding to a Regulatory Guide 1.3 or 1.4 release.

2.1.8.b

Increased Range of Radiation Monitors

POOR ORIGINAL

2.1.8 INSTRUMENTATION TO FOLLOW THE COURSE OF AN ACCIDENT

2.1.8.b INCREASED RANGE OF RADIATION MONITORS

NUREG-0578 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.

1059 217

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.

PVNGS Evaluation

The PVNGS radiation monitoring system is a follows;

1. Noble gas effluent monitoring of the plant vent is presently designed to have an upper limit of 10^{-1} μCi/cc (Xe-133). Additional monitors will be added with an upper range of 10^4 μCi/cc (Xe-133). The range of the plant vent monitors shall overlap by at least a factor of ten.

2. Post-accident effluent monitoring of radioiodines is provided for by sampling of plant vent iodine monitor adsorption media [illegible] y onsite laboratory analysis. Procedures for this sampling will be developed prior to Unit 1 Operating License.

1059 218

3. The PVNGS power access purge monitors (two redundant, ESF monitors) are located around the power access purge duct adjacent to the containment. These monitors are capable of monitoring the purge duct during normal operation and the containment (through the containment wall) subsequent to an accident. When monitoring the containment radiation level these monitors will have a maximum range of at least 10^6 rad/hr. The monitors will be qualified to post-accident condition. The design of the monitors will be evaluated further to determine if a maximum range of 10^8 rad/hr is achievable. This evaluation will be completed by March, 1981.

2.1.8.c

Improved In-Plant Iodine Instrumentation

POOR ORIGINAL

1059 220

2.1.8 INSTRUMENTATION TO FOLLOW THE COURSE OF AN ACCIDENT

2.1.8.c IMPROVED IN-PLANT IODINE INSTRUMENTATION

NUREG-0578 Position

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

PVNGS Evaluation

Prior to PVNGS Unit 1 Operating License, procedures will be implemented for determination of airborne iodine concentration by sampling iodine monitor cartridges for spectrum analysis. These procedures will be used when the plant radiation monitoring system indicates iodine levels requiring the use of respiratory equipment.

Spectral analysis equipment is available at each PVNGS unit.

2.1.9

1059 222

2.1.9 ANALYSIS OF DESIGN AND OFF-NORMAL TRANSIENTS AND ACCIDENTS

NUREG-0578 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.

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 of 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.

1059 225

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.

PVNGS Evaluation

Arizona Public Service Company will work with Combustion Engineering to determine the generic analyses necessary for the System 80 design. The analyses will be performed on a schedule consistent with development of procedures, training of operator's installation of hardware (refer to Section 2.1.3) prior to PVNGS Unit 1 Operating License.

1059 226

2.1.10

2.1.10 INSTRUMENTATION TO MONITOR CONTAINMENT CONDITIONS DURING THE COURSE OF AN ACCIDENT

NUREG-0578 Position

Consistent with satisfying the requirements set forth in General Design Criterion 13 to provide the capability in the control room to ascertain containment conditions during the course of an accident, the following requirements shall be implemented:

1. A continuous indication of containment pressure shall 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 minum five psig for all containments.
2. 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.
3. 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. Also for PWRs, a wide range instrument shall be provided and cover the range

from the bottom of the containment to the elevation equivalent to a 500,000 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.

The containment pressure, hydrogen concentration and wide range containment water level measurements shall meet the design and qualification provisions of Regulatory Guide 1.97, including qualification, redundancy, and testibility. The narrow range containment water level measurement instrumentations shall be qualified to meet the requirements of Regulatory Guide 1.89 and shall be capable of being periodically tested.

PVNGS Evaluation

PVNGS containment monitoring instrumentation is as follows:

1. Eight, Class IE containment pressure indicators are provided in the control room. Four indicators have a -4 to 20 psig range. The other four indicators have a -4 to 85 psig range.

   An analysis performed for the containment maximum negative pressure (refer to FSAR Section 6.2.1.1.3.6) demonstrates a maximum credible negative pressure of -2.6 psig. The existing pressure indicator range (to -4 psig) is, therefore, considered adequate to measure containment negative pressure under any postulated conditions.

The containment is designed for 60 psig internal pressure and tested to 69 psig. The maximum calculated containment pressure under accident conditions is 49.2 psig. The existing pressure indicator range (to 85 psig) provides a 40% margin above the containment design pressure and a 70% margin above the conservatively calculated peak pressure. Therefore, a containment pressure range greater than the existing 85 psig maximum is not considered to be necessary.

The addition of monitors with a range to 180 psig would add unnecessary confusion to the available containment pressure instrumentation by:

a. Displaying at only 25% of span during the worst postulated accident.

b. Displaying containment pressure with an accuracy of only ± 3.6 psi ($\pm 2\%$).

2. A continuous indication of hydrogen concentration is available in the control room. Two Class IE redundant monitors are presently provided with a range of 0-10%. The hydrogen monitors are manually initiated following a LOCA. Once initiated, they provide a continuous measurement of hydrogen concentration (refer to Figure 2.1.5-1).

1059 230

3. Continuous indication of containment water level will be provided for PVNGS. An evaluation will be performed to determine the design requirements for this instrumentation. The results of this evaluation will be presented to the NRC staff by March, 1981.

1059 231

2.1.11

2.1.11 INSTALLATION OF REMOTELY OPERATED HIGH POINT VENTS IN THE REACTOR COOLANT SYSTEM

NUREG-0578 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:

1. 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 50.46.

2. Analyses demonstrating that the direct venting of non-condensable 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 Plan Section 6.2.5.

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

PVNGS Evaluation

PVNGS will utilize existing penetrations on the pressurizer and reactor vessel head provide high point venting of the Reactor Coolant System (RCS) Refer to Figure 2.1.11-1 for a schematic of the RCS. The pressurizer vent (refer to Figure 2.1.11-2) penetrates the top of the pressurizer and is currently utilized for steam space sampling. The reactor vessel head vent (refer to Figure 2.1.11-2) penetrates the top of the reactor vessel head and is currently piped to the Reactor Drain Tank (RDT), within containment, via a manual valve. Piping and valves will be provided to allow remote operation of the pressurizer and reactor vessel vents from the control room. In order to ensure a low probability of inadvertent action, valves and controls will satisfy the single failure criterion and IEEE-279-1971.

The following information will be supplied as the RCS Vent System design is finalized (prior to Unit 1 Operating License):

1. A description of the vents, piping, valves and controls utilized for the RCS Vent System. An analysis demonstrating the acceptability of a vent piping break in accordance with 10 CFR 50.46.

2. An analysis demonstrating that the venting of RCS non-condensable gases via the RCS Venting System does not result in containment hydrogen concentrations greater than those allowed by 10 CFR 50.44, Regulatory Guide 1.7, Revision 0 and Standard Review Plan 6.2.5.

3. Operating procedures for use of the RCS Vent System.

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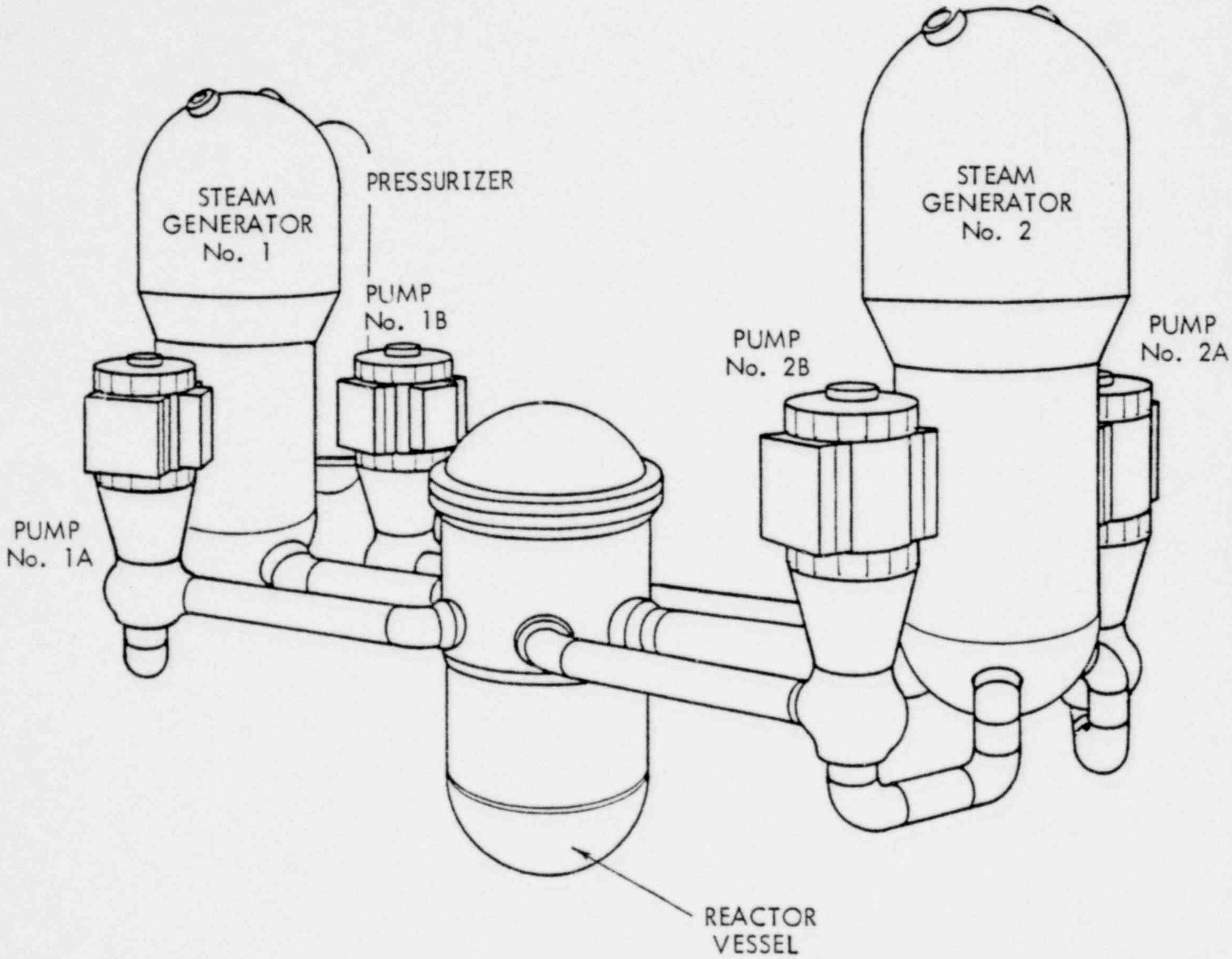

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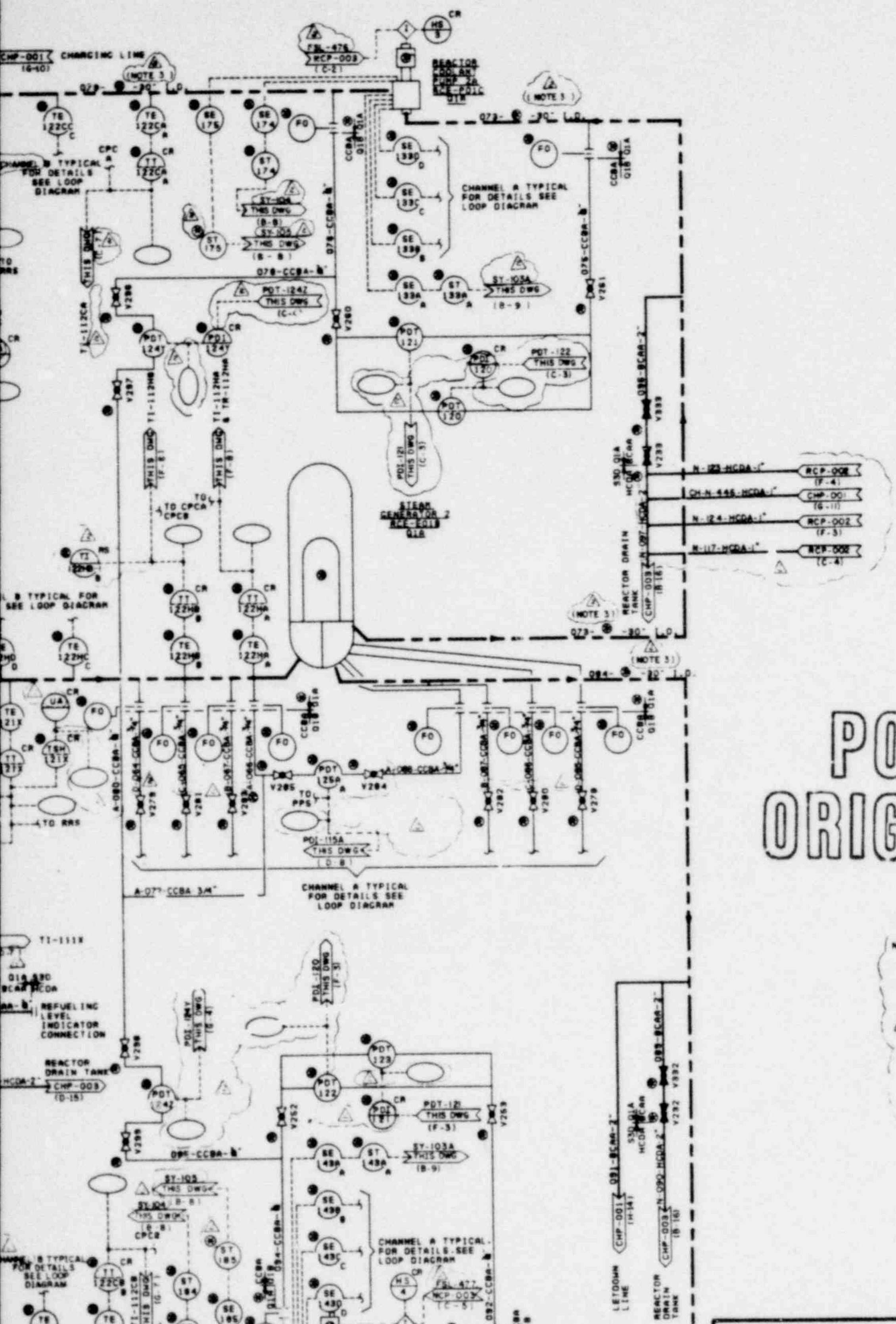
C - E SYSTEM 80	ISOMETRIC VIEW OF THE REACTOR COOLANT SYSTEM	Figure 2.1.11-1



POOR ORIGINAL

NOTE

1. ALL GLOBE VALVES SHOWN IN THIS DRAWING HAVE FLOW UNDER THE SEAT
2. THE SAFETY TRAIN DESIGNATOR E IS TO PRECEDE ALL LINE NUMBERS SHOWN IN THIS DRAWING UNLESS OTHERWISE INDICATED
3. CE SUPPLIED PIPING DESIGNED FOR 2500 PSIA AND 650°, AND PER CODE REQUIREMENTS OF ASME SECTION III, CLASS 1
4. THIS P&ID IS REDRAWN FROM THE CE SUPPLIED P&ID PVNGS DRAWING E-14273-310-110 REVISION NO.02

THE SYSTEM DESIGNATOR RC IS TO PRECEDE ALL LINE, VALVE, AND INSTRUMENT NUMBERS SHOWN ON THIS DRAWING UNLESS OTHERWISE INDICATED

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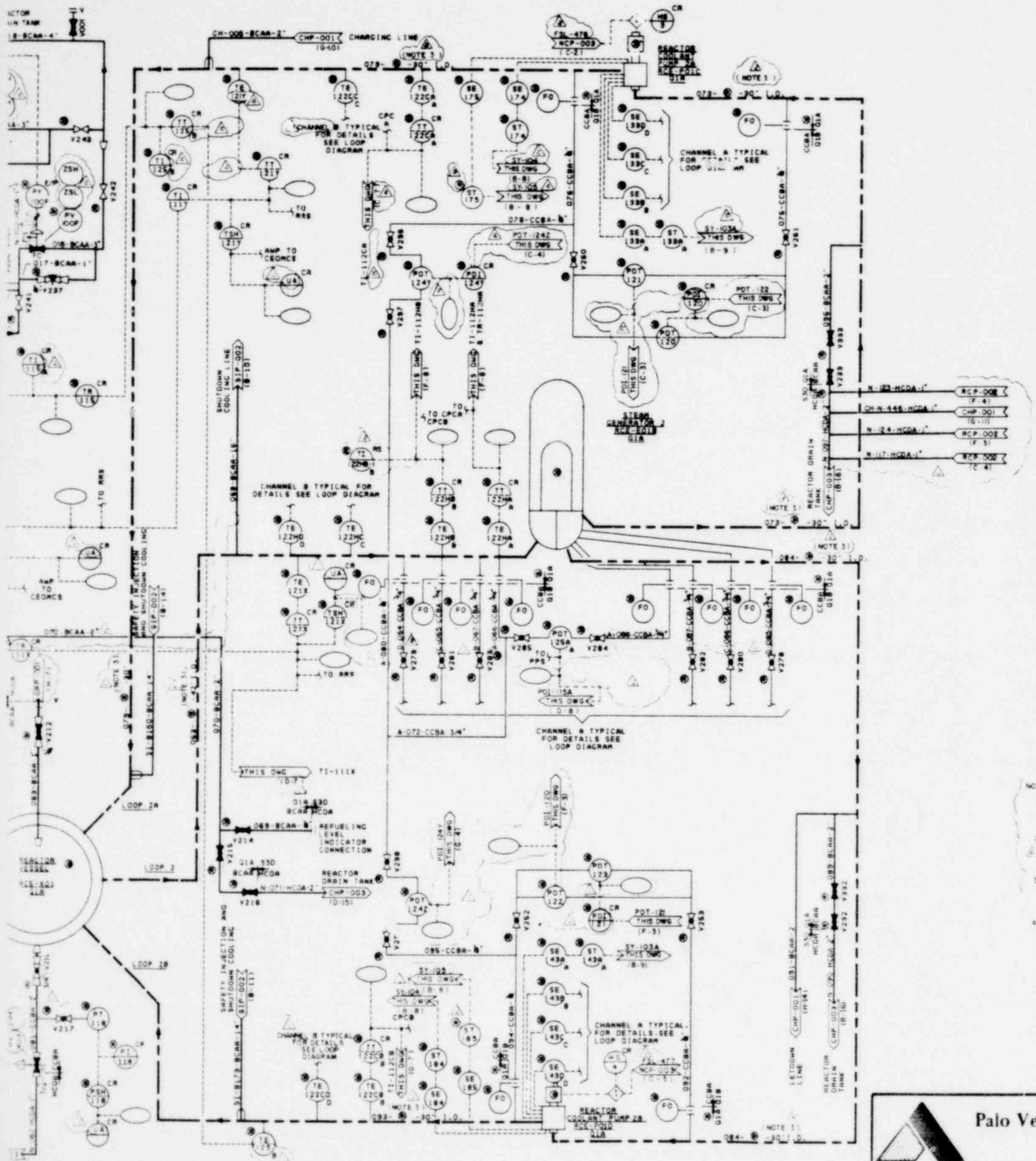
Palo Verde Nuclear Generating Station

REACTOR COOLANT SYSTEM

Figure 2.1.11-2

1059 237



POOR ORIGINAL

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Palo Verde

REACTOR C

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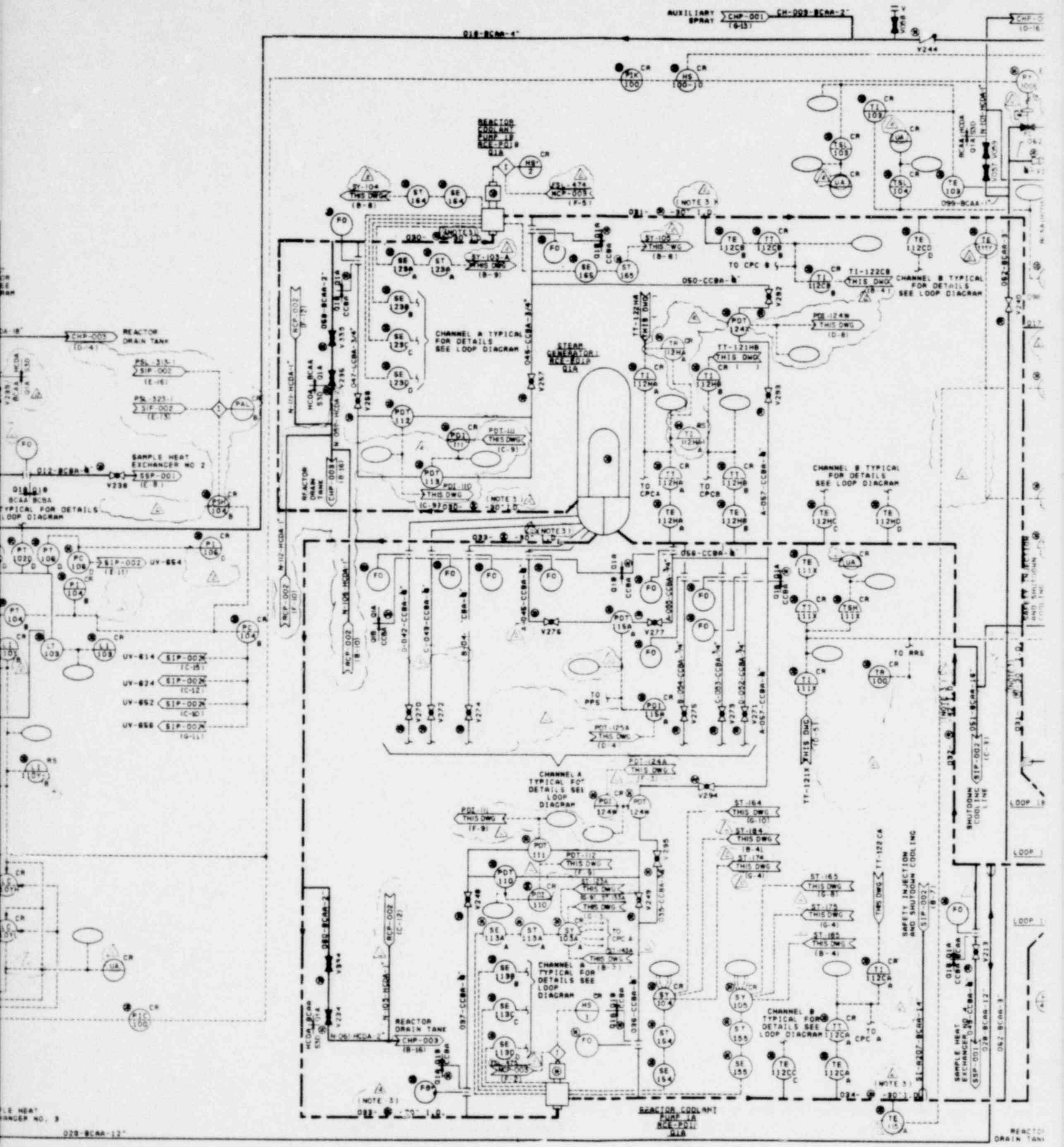

POOR ORIGINAL

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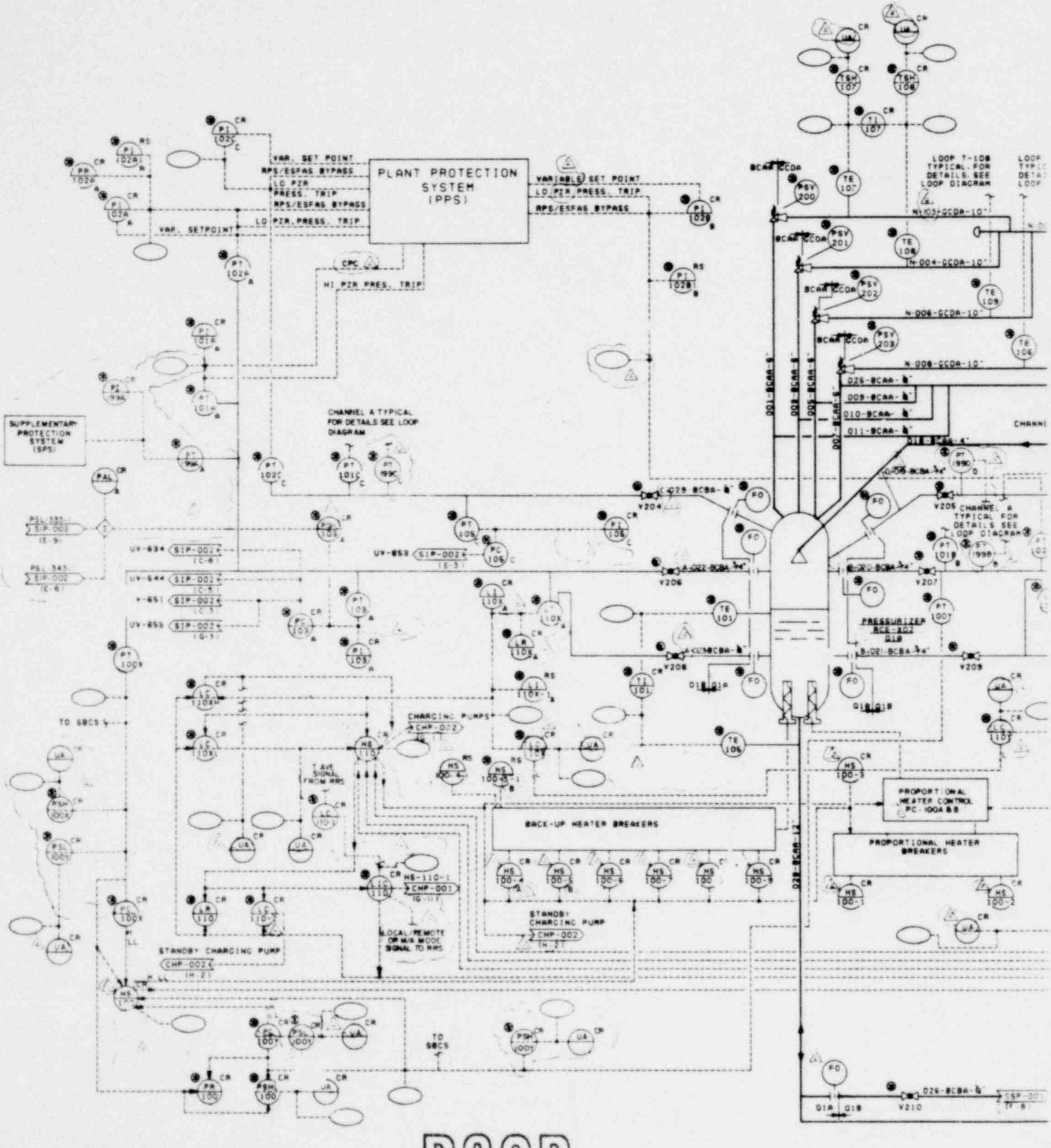

POOR ORIGINAL

1059 240

OPERATIONS

1052 241

2.2.1

2.2.1.a

Shift Supervisor's Responsibilities

2.2.1 IMPROVED REACTOR OPERATIONS COMMAND FUNCTION

2.2.1.a SHIFT SUPERVISORS RESPONSIBILITIES

NUREG-0578 Position

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.

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.

PVNGS Evaluation

The responsibilities and authority of the Shift Supervisor is delineated in FSAR Chapter 13.1. In addition, the Station Manual and the Operating Department Program also prescribe responsibilities and authority of the Shift Supervisor.

The shift organization at PVNGS is such that the Shift Foreman will perform many of the administrative functions that are normally completed during a shift.

2.2.1.b

Shift Technical Advisor

2.2.1 IMPROVED REACTOR OPERATIONS COMMAND FUNCTION

2.2.1.b SHIFT TECHNICAL ADVISOR

NUREG-0578 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 tne 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 spycific 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.

PVNGS Evaluation

There is sufficient time prior to PVNGS Unit 1 Operating License to assess fully the alternatives for providing

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additional technical and analytical capability for diagnosis of off-normal events. Two alternatives considered by the Lessons Learned Task Force have merit for PVNGS:

1. Upgrade the requirements for operators to include more engineering and plant response training.
2. Improve plant response diagnosis capabilities by backfit of computer and plant status display innovations.

The PVNGS Operator Training Program will be revised and upgraded to include more engineering and plant response training. The PVNGS Training Simulator will be used extensively to accomplish this objective.

The PVNGS design already has incorporated some advanced informational displays. These include the Safety Equipment Status Panels, computer-generated graphics displays and the various information displays which can be generated by the Core Monitoring Computer and the Plant Monitoring Computer.

2.2.1.c

Shift and Relief Turnover Procedures

POOR ORIGINAL

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2.2.1 IMPROVED REACTOR OPERATIONS COMMAND FUNCTION

2.2.1.c SHIFT RELIEF AND TURNOVER PROCEDURE

NUREG-0578 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 oncoming auxiliary operators and technicians. Such checklists or logs shall include any equipment under maintenance of test that by themselves could degrade a system critical to the p e-vention 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); ar

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

PVNGS Evaluation

PVNGS will have a procedure for shift relief which will include the use of checklists and the review of operating

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logs. The relieving operator or supervisor will be required to sign that he has read the appropriate logs and completed the watch turnover as specified in the procedure.

Provisions will be made for periodic verification of the relief and turnover procedure.

The shift relief and turnover procedure and the provisions for its periodic verification will be prepared and implemented prior to PVNGS Unit 1 Operating License.

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2.2.2

2.2.2.a

Control Room Access

POOR ORIGINAL

2.2.2 IMPROVED IN-PLANT EMERGENCY PROCEDURES AND PREPARATIONS

2.2.2.a CONTROL ROOM ACCESS

NUREG-0578 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.

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.

PVNGS Evaluation

Prior to PVNGS Unit 1 Operating License, administrative procedures will be develcped to limit personnel access to the unit control room during operational transient and accident conditions. These procedures will delineate:

1. That the shift supervisor has the authority to restrict control room access to personnel responsible for plant operations and technical advisory support to operations.

2. Clear lines of communications, authority, and responsibilities in the event of an emergency. Only senior licensed operators will direct the licensed operators.

1059 257

2.2.2.b

Onsite Technical Support Center

POOR ORIGINAL

1059 258

2.2.2 IMPROVED IN-PLANT EMERGENCY PROCEDURES AND PREPARATIONS

2.2.2.b ONSITE TECHNICAL SUPPORT CENTER

NUREG-0578 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 stored and filed at the site and accessible to the technical support center under emergency conditions. Examples of such records include a system description, general arrangement drawings, piping and instrumentation diagrams, piping system isometrics, electrical schematics, wire and cable lists, and single line electrical diagrams. It is not intended that all records described in ANSI 45.2.9-1974 be stored and filed at the site

and accessible to the technical support center under emergency conditions; however, as stated in that standard, storage systems shall provide for accurate retrieval of all information without undue delay.

PVNGS Evaluation

An Onsite Technical Support Center for Palo Verde Nuclear Generating Station will be established prior to PVNGS Unit 1 Operating License. This center will be located within the site boundary and will provide a work area for supervisory and technical personnel from Arizona Public Service Company and the NRC. It will have communications with the control room and the other response centers. An analysis will be performed to determine equipment to be incorporated into the center to monitor and display the status of the affected unit. The center will be habitable to permit occupancy following a loss of coolant accident. As-built drawings and other appropriate records shall be available in readily retrievable form at the site and shall be accessible to the Onsite Technical Support Center.

The Palo Verde Nuclear Generating Station Emergency Plan will be revised to describe the Onsite Technical Support Center and its function.

1059 260

2.2.2.c

Onsite Operational Support Center

POOR ORIGINAL

1059 261

2.2.2 IMPROVED IN-PLANT EMERGENCY PROCEDURES AND PREPARATIONS

2.2.2.c ONSITE OPERATIONAL SUPPORT CENTER

NUREG-0578 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.

PVNGS Evaluation

Prior to PVNGS Unit 1 Operating License, an area separate from the unit control rooms will be designated as the PVNGS Onsite Operational Support Center prior to receipt of operating license. This will be an assembly area where plant operations support personnel will report in an emergency situation for further orders or assignment. The Operational Support Center will have communication with the control room and the Emergency Control Center (refer to the PVNGS Emergency Plan).

The Emergency Plan will reflect the existence of the Operational Support Center and will establish the methods and lines of communication and management.

2.2.3

1059 265

2.2.3 REVISED LIMITING CONDITIONS FOR OPERATION OF NUCLEAR POWER PLANTS BASED UPON SAFETY SYSTEM AVAILABILITY

NUREG-0578 Position

All NRC nuclear power plant licensees shall provide information to define a limiting operational condition based on a threshold of complete loss of safety function. Identification of a human or operational error that prevents or could prevent the accomplishment of a safety function required by NRC regulations and analyzed in the license application shall require placement of the plant in a hot shutdown condition within 8 hours and in a cold shutdown condition within 24 hours.

The loss of operability of a safety function shall include consideration of the necessary instrumentation, controls, emergency electrical power sources, cooling or seal water, lubrication, operating procedures, maintenance procedures, test procedures and operator interface with the system, which must also be capable of performing their auxiliary or supporting functions. The limiting conditions for operation shall define the minimum safety functions for modes 1, 2, 3, 4, and 5 of operation.

The limiting conditions of operation shall require the following:

1. If the plant is critical, restore the safety function (if possible) and place the plant in a hot shutdown condition within 8 hours.

2. Within 24 hours, bring the plant to cold shutdown.

3. Determine the cause of the loss of operability of the safety function. Organizational accountability for the loss of operability of the safety system shall be established.

4. Determine corrective actions and measures to prevent recurrence of the specific loss of operability for the particular safety function and generally for any safety function.

5. Report the event within 24 hours by telephone and confirm by telegraph, mailgram, or facsimile transmission to the Director of the Regional Office, or his designee.

6. Prepare and deliver a Special Report to the NRC's Director of Nuclear Reactor Regulation and to the Director of the appropriate regional office of the Office of Inspection and Enforcement. The report shall contain the results of Steps 3 and 4 above,

along with a basis for allowing the plant to return to power operation. The senior corporate executive of the licensee responsible and accountable for safe plant operation shall deliver and discuss the contents of the report in a public meeting with the Office of Nuclear Reactor Regulation and the Office of Inspection and Enforcement at a location to be chosen by the Director of Nuclear Reactor Regulation.

7. A finding of adequacy of the licensee's Special Report by the Director of Nuclear Reactor Regulation will be required before the licensee returns the plant to power.

PVNGS Evaluation

In his August 20, 1979 memorandum to the NRC Commissioners, Harold R. Denton, Director, Office of Nuclear Reactor Regulatory, announced his intent to initiate a rulemaking proposal on limiting conditions of operations, which would include consideration of the recommendations of the Lessons Learned Task Force and alternative approaches. APS will follow this proposal with great interest and will participate in any rulemaking proceedings that may be instituted.