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ARIZONA PUBLIC SERVICE COMPANY

P. O. BOX 21666 · PHOENIX, ARIZONA 85036

June 19, 1980
ANPP-15665-EEVBJr

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

Re: Joint Application for Operating Licenses
Palo Verde Nuclear Generating Station
Units 1, 2 and 3
NRC Docket Nos. STN-50-528/529/530

Reference: Letter dated June 18, 1980 to E. E. Van Brunt, Jr., Vice President, Nuclear Projects, Arizona Public Service Company from Darrell Eisenhut, Director, Division of Licensing, USNRC

In accordance with the reference letter, 10 CFR 2.101 and 10 CFR 50.30, provided herewith are (1) fifteen copies of the General Information required by 10 CRF 50.33, (2) forty copies of the Final Safety Analysis Report for the Palo Verde Nuclear Generating Station, (3) forty-one copies of the Environmental Report - Operating License Stage for the Palo Verde Nuclear Generating Station, and (4) forty copies of the TMI-2 Lessons Learned Implementation Report for the Palo Verde Nuclear Generating Station.

Proof of Service for the distribution required by the reference letter and 10 CFR 2.101 will be provided under separate cover within ten days of docketing.

Respectfully submitted,

ARIZONA PUBLIC SERVICE COMPANY

BY: Edwin E. Van Brunt
Edwin E. Van Brunt, Jr.
APS Vice President,
Nuclear Projects
ANPP Project Director

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

State of Arizona)
) ss:
County of Maricopa)

Subscribed and sworn to before me this

19th day of June, 1980

John M. Allen

My Commission Expires Jan. 23, 1983

GEN INFO
~~L004~~ L004 S 1/15
FSAR S 1/40
TMI & L005
E.R
L006 S 1/4

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PALO VERDE NUCLEAR GENERATING STATION

TMI-2 LESSONS LEARNED IMPLEMENTATION REPORT

ARIZONA PUBLIC SERVICE COMPANY
PROJECT MANAGER AND OPERATING AGENT

INTRODUCTION

The Palo Verde Nuclear Generating Station Lessons Learned Implementation Report (LLIR) addresses those TMI-related items approved for implementation by the Commission at this time as published in the "Clarification of TMI Action Plan Requirements," NUREG-0737, November 1980.

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 existing PVNGS design correspond with information presented in the PVNGS Final Safety Analysis Report (FSAR) as amended and in Combustion Engineering, Inc. (C-E) Standard Safety Analysis Report - Final Safety Analysis Report (CESSAR) as amended

(2) Future work performed and modifications installed will be summarized as revisions to this report

(3) This report includes only requirements from NUREG-0737 related to PVNGS

(4) Where appropriate, responses will be incorporated into the FSAR in future FSAR amendments

(5) The NRC positions presented herein are from NUREG-0737 unless otherwise noted.

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I. OPERATIONAL SAFETY

I.A OPERATING PERSONNEL

I.A.1.1 SHIFT TECHNICAL ADVISOR

Position

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

The STA shall have a bachelor's degree or equivalent in a scientific or engineering discipline and have received specific training in the response and analysis of the plant for transients and accidents. The STA 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 STAs that pertain to the engineering aspects of assuring safe operations of the plant, including the review and evaluation of operating experience.

PVNGS Evaluation

PVNGS plans to accomplish the function of enhanced accident assessment by upgrading the qualifications of shift supervisors and assistant shift supervisors, and upgrading the man-machine interface. Until these actions are accomplished, a shift technical advisor shall be present on shift, commencing at fuel load for Unit 1, whenever one or more units are operating in modes 1 through 4 as defined in the Technical Specifications.

The following information will be submitted by December 1, 1981:

1. Description of current STA training program and demonstration of conformance with November 9, 1979, NRC letter, D. B. Vassallo to All Pending Construction Permit Applicants.

2. Plans for STA requalification training.

3. Description of long-term STA program, including qualifications, selection criteria, training plans, and plans, if any, for the eventual phaseout of the STA program.

I.A.1.2 SHIFT SUPERVISOR ADMINISTRATIVE DUTIES

Position (NUREG-0694)

Review the administrative duties of the shift supervisor and delegate functions that detract from or are subordinate to the management responsibility for assuring safe operation of the plant to other personnel not on duty in the control room.

PVNGS Evaluation

The responsibility and authority of the shift supervisor is delineated in FSAR Section 13.1. The administrative duties of the shift supervisor will be defined in the PVNGS Station Manual and will be in accordance with the guidance in the November 9, 1979 letter from D. B. Vassallo to All Pending Construction Permit Applicants. Administrative functions which detract from or are subordinate to plant operational safety will be assigned to other personnel who do not direct operational functions.

Additionally, the PVNGS Operations organization includes a senior licensed assistant shift supervisor for each unit, in addition to the assigned shift supervisor who will perform many of the administrative functions which typically would have been performed by the shift supervisor.

The PVNGS Operations organization also includes personnel (including the licensed day shift supervisor for each unit) available during day shift on week days whose function is to assume administrative functions that otherwise might detract from the shift organizations' ability to devote full attention to the operation of the plant.

I.A.1.3 SHIFT MANNING

Position

(1) Limit Overtime

Administrative procedures shall be established to limit maximum work hours of all personnel performing a safety-related function. 2

(2) Minimum Shift Crew

The minimum shift crew for a unit shall include three operators, plus an additional three operators when the unit is operating. Shift staffing may be adjusted at multi-unit stations to allow credit for operators holding licenses on more than one unit.

In each control room, including common control rooms for multiple units, there shall be at all times a licensed reactor operator for each reactor loaded with fuel and a senior reactor operator licensed for each reactor that is operating. There shall also be onsite at all times, an additional relief operator licensed for each reactor, a licensed senior reactor operator who is designated as shift supervisor, and any other licensed senior reactor operators required so that their total number is at least one more than the number of control rooms from which a reactor is being operated.

PVNGS Evaluation

1. Limit Overtime

PVNGS administrative procedures shall, by fuel load, provide provisions limiting maximum hours worked by personnel performing a safety related function to no more than 12 hours of continuous duty exclusive of travel time with at least 12 hours between work periods, no more than 72 hours in any 7-day period, and no more than 14 consecutive
2 tive days of work without at least two consecutive days off. Only the Manager of Nuclear Operations shall have authority to waive these limits. The personnel effected by this requirement will be senior reactor operators, reactor operators, radiation protection technicians, auxiliary operators, I & C technicians and key maintenance personnel.

2. Minimum Shift Crew

Shift crew composition may be less than the minimum requirements for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew
2 member, provided immediate action is taken to restore the shift crew composition to within the minimum requirements of the following table.

The minimum shift organization as described in FSAR Section 13.1.2.3 will be updated to require, commencing at fuel load, for each unit:

License Category	Applicable Modes (a) 1, 2, 3 and 4(d)	5 and 6
SOP (SRO)	2(b)	1(b,c)
OP (RO)	2	1
Non-Licensed	2	1

(a) Operational modes are as defined in the Technical Specifications. (CESSAR FSAR Chapter 16). 2

(b) At least one of the unit senior reactor operators on site and on shift will be a shift supervisor. At least one senior reactor operator will be in the unit control room during mode 1 through 4 operations for each unit, however, this requirement may be waived for a brief time to allow response to in-plant conditions. 2

(c) Does not include the licensed senior reactor operator or senior reactor operator limited to fuel handling, supervising core alterations. A licensed senior operator is required to directly supervise any core alteration activity.

(d) A shift technical advisor (STA) will be on site and on shift for a period of 24 hours at a time. The STA will be asleep at times during this period, however he will be available in the control room within ten minutes whenever one or more units are operating in modes one through four. 2

PVNGS administrative procedures will by fuel load provide
2 provisions governing required shift staffing.

I.A.2.1 IMMEDIATE UPGRADING OF REACTOR OPERATOR AND SENIOR REACTOR OPERATOR TRAINING AND QUALIFICATIONS

Position (March 28, 1980 NRC letter, H. R. Denton to All Power Reactor Applicants and Licensees.)

(1) SRO Experience

Applicants for senior operator licenses shall have 4 years of responsible power plant experience. Responsible power plant experience should be that obtained as a control room operator (fossil or nuclear) or as a power plant staff engineer involved in the day-to-day activities of the facility, commencing with the final year of construction. A maximum of 2 years power plant experience may be fulfilled by academic or related technical training, on a one-for-one time basis. Two years shall be nuclear power plant experience. At least 6 months of the nuclear power plant experience shall be at the plant for which he seeks a license.

Effective date: Applications received on or after May 1, 1980.

(2) SRO's be RO's, 1 year

Applicants for senior operator licenses shall have held an operator's license for 1 year.

Effective date: Applications received after December 1, 1980.

(3) Three Month Training On-Shift

(a) Senior operator*: Applicants shall have 3 months of shift training as an extra man on shift.

(b) Control room operator*: Applicants shall have 3 months training on shift as an extra person in the control room.

Effective date: Applications received after August 1, 1980.

(4) Modify Training

Training programs shall be modified as necessary, to provide:

(a) Training in heat transfer, fluid flow and thermodynamics.

(b) Training in the use of installed plant systems to control or mitigate an accident in which the core is severely damaged.

(c) Increased emphasis on reactor and plant transients.

Effective date: Present programs have been modified in response to Bulletins and Orders. Revised programs should be submitted for OLB review by August 1, 1980.

*Precritical applicants will be required to meet unique qualifications designed to accommodate the fact that their facility has not yet been in operation.

(5) Facility Certification

Certifications completed pursuant to Sections 55.10(a)(6) and 55.33a(4) and (5) of 10 CFR Part 55 shall be signed by the highest level of corporate management for plant operation (for example, Vice President for Operations).

Effective date: Applications received on or after May 1, 1980.

<u>PVNGS Evaluation</u>

1. SRO Experience

SRO applicants will be required to have 4 years of responsible power plant experience, of which at least 2 years shall be nuclear power plant experience (including 6 months at PVNGS) with no more than 2 years being academic or related technical training.

2. SRO's be RO's, 1 year.

Applicants for a SRO hot license will have been a licensed operator for 1 year commencing 18 months following Unit 1 fuel load. The requirement for 1 year of licensed operator experience may be satisfied by:

A. Holding an NRC operator or senior operator license at a nuclear power plant (including PVNGS) for at least 1 year.

B. Serving in a position equivalent to a licensed operator or senior operator at a military propulsion reactor for at least 1 year.

C. Persons who possess a degree in engineering or applicable sciences who:

(1) Have 4 years of responsible power plant experience. Responsible power plant experience should be that obtained as a control room operator (fossil or nuclear) or as a power plant staff engineer involved in the day-to-day activities of the facility, commencing with the final year of construction. A maximum of 2 years power plant experience may be fulfilled by academic or related technical training, on a one-for-one time basis. Two years shall be nuclear power plant experience. At least 6 months of the nuclear power plant experience shall be at the plant for which he seeks a license.

(2) Have 3 months shift training as an extra person on shift

(3) Training in heat transfer, fluid flow and thermodynamics

(4) Training in the use of installed plant systems to control or mitigate an accident in which the core is severely damaged

(5) Training with emphasis on reactor and plant transients.

3. Three Month Training On-Shift

The training program described in FSAR Section 13.2.1 provides a combination of experience and training in the control room and on a simulator that provides license candidates appropriate control room experience. 2

4. Modify Training

Heat transfer, fluid flow and thermodynamics will be included in Phase I training. Our response to item II.B.4 addresses training for mitigating core damage. Plant transients are currently covered in Phase IV training on the simulator.

5. Facility Certification

The PVNGS operating organization is described in PVNGS FSAR Section 13.1.2. The Vice President of Electric Operations for Arizona Public Service Company is the highest level of corporate management for plant operation and, as such, shall sign certifications pursuant to Sections 55.10(a)(6) and 55.33a(4), and (5) of 10 CFR Part 55.

I.A.2.3 ADMINISTRATION OF TRAINING PROGRAMS

Position

Pending accreditation of training institutions, licensees and applicants for operating licenses will assure that training center and facility instructors who teach systems, integrated responses, transient, and simulator courses demonstrate senior reactor operator (SRO) qualifications and be enrolled in appropriate requalification programs.

PVNGS Evaluation

Commencing at Unit 1 fuel load, permanent members of the training staff who instruct licensed operator candidates in systems, integrated transient responses, and control room simulator courses shall have 4 years of power plant experience, of which 2 years shall be nuclear power plant experience (including 6 months at PVNGS) with no more than 2 years being academic or related technical training. They shall successfully complete an examination equivalent to an SRO examination. They shall take part in the requalification training to maintain current on plant operating history, problems and changes to procedures and administrative limitations. If licensed as an SRO they shall participate in the requalification training program with the exception stated in PVNGS FSAR Section 13.2.2. 2

I.A.3.1 REVISE SCOPE AND CRITERIA FOR LICENSING EXAMINATIONS

Position

(1) Increase Scope (NRC letter, H. R. Denton to All Power Reactor Applicants and Licensees, dated March 28, 1980)

(a) A new category shall be added to the operator written examination entitled, "Principles of Heat Transfer and Fluid Mechanics."

(b) A new category shall be added to the senior operator written examination entitled, "Theory of Fluids and Thermodynamics."

(c) Time limits shall be imposed for completion of the written examinations:

1. Operator: 9 hours
2. Senior Operator: 7 hours

(d) All applicants for senior operator licenses shall be required to be administered an operating test as well as the written examination.

(e) Applicants will grant permission to NRC to inform their facility management regarding the results of the examinations for purposes of enrollment in requalification programs.

(2) Increase Passing Grade (NRC letter, H. R. Denton to All Power Reactor Applicants and Licensees, dated March 28, 1980)

The passing grade for the written examination shall be 80% overall and 70% in each category.

(3) Simulator Exams (NUREG 0737)

Simulator examinations will be included as part of the licensing examinations.

PVNGS Evaluation

1. Increase Scope

Refer to the response in item I.A.2.1. PVNGS shall request that each applicant for an operator license grant permission to the NRC to inform PVNGS management regarding the result of their examination.

2. Increase Passing Grade

All license candidates which are recommended for license examinations are expected to have the ability to complete the examination with a satisfactory score in each category. Candidates will be evaluated on a basis of a passing grade of 80% overall and 70% in each category.

3. Simulator Exams

2 The PVNGS plant specific simulator will be made available to NRC examiners for examining candidates for reactor operator and senior reactor operator licenses prior to fuel load, including cold examinations.

I.B OVERALL ORGANIZATION

I.B.1.2 INDEPENDENT SAFETY ENGINEERING GROUP

Position

Each applicant for an operating license shall establish an onsite independent safety engineering group (ISEG) to perform independent reviews of plant operations.

The principal function of the ISEG is to examine plant operating characteristics, NRC issuances, Licensing Information Service advisories, and other appropriate sources of plant design and operating experience information that may indicate areas for improving plant safety. The ISEG is to perform independent review and audits of plant activities, including maintenance, modifications, operational problems, and operational analysis, and aid in the establishment of programmatic requirements for plant activities. Where useful improvements can be achieved, it is expected that this group will develop and present detailed recommendations to corporate management for such things as revised procedures or equipment modifications.

Another function of the ISEG is to maintain surveillance of plant operations and maintenance activities to provide independent verification that these activities are performed correctly and that human errors are reduced as far as practicable. ISEG will then be in a position to advise utility management on the overall quality and safety of operations. ISEG need not perform detailed audits of plant operations and shall not be responsible for sign-off functions such that it becomes involved in the operating organization.

PVNGS Evaluation

Arizona Public Service Company will establish a group that is independent of the plant staff, but is assigned onsite, to perform independent reviews of plant operations and which may have the capability for evaluation of operating experiences at nuclear power plants. The establishment and staffing of this group will be implemented prior to the fuel load of PVNGS Unit 1.

I.C OPERATING PROCEDURES

I.C.1 GUIDANCE FOR THE EVALUATION AND DEVELOPMENT OF PROCEDURES FOR TRANSIENTS AND ACCIDENTS

Position

In letters of September 13 and 27, October 10 and 30, and November 9, 1979, the Office of Nuclear Reactor Regulation required licensees of operating plants, applicants for operating licenses and licensees of plants under construction to perform analyses of transients and accidents, prepare emergency procedure guidelines, upgrade emergency procedures, including procedures for operating with natural circulation conditions, and to conduct operator retraining (see also item I.A.2.1). Emergency procedures are required to be consistent with the actions necessary to cope with the transients and accidents analyzed. Analyses of transients and accidents were to be completed in early 1980 and implementation of procedures and retraining were to be completed 3 months after emergency procedure guidelines were established; however, some difficulty in completing these requirements has been experienced. Clarification of the scope of the task and appropriate schedule revisions are being developed. In the course of review of these matters on Babcock and Wilcox (B&W)-designed plants, the staff will follow up on the bulletin and orders matters relating to analysis methods and results, as listed in NUREG-0660, Appendix C (see

Table C.1, Items 3, 4, 16, 18, 24, 25, 26, 27; Table C.2, Items 4, 12, 17, 18, 19, 20; and Table C.3, Items 6, 35, 37, 38, 39, 41, 47, 55, 57).

PVNGS Evaluation

Arizona Public Service Company (APS) has participated in
C-E Owners Group activities conducted since the Three Mile
Island accident to develop improved emergency procedure guide-
lines and associated supporting analyses. The C-E Owners Group
has completed numerous documents which have been submitted to
1 the NRC for review. A summary of results obtained to date and
current activities is provided below.

The initial C-E Owners Group analysis of Inadequate Core Cooling (ICC) is documented in Report CEN-117, "Inadequate Core Cooling - A Response to NRC IE Bulletin 79-06C, Item 6 for Combustion Engineering Nuclear Steam Supply Systems." This report was submitted to the NRC staff for review on October 31, 1979, by the C-E Owners Group.

"Operational Guidance for Inadequate Core Cooling" was prepared by the C-E Owners Group based on the analyses in Report CEN-117. This operational guidance was distributed to members of the C-E Owners Group for their use in review and possible revision of plant emergency procedures in December, 1979. A copy of this operational guidance was submitted to the NRC staff for review by the C-E Owners Group on December 10, 1980.

Since early 1980, the C-E Owners Group has sponsored an extensive study of instrumentation response characteristics under ICC conditions. This study was described to the NRC staff at a meeting in Bethesda, Maryland on May 28, 1980. This study was completed in December, 1980, and its results have been distributed to members of the C-E Owners Group for their use. APS is currently evaluating the results of this study for use in possible revisions to plant emergency procedures. Such revisions would be based upon determination of the usefulness of specific instrumentation for detection of ICC. This evaluation and subsequent revision of plant emergency procedures, as required, is expected to be completed by fuel load.

The initial C-E Owners Group analyses of transients and accidents (non-LOCA) are documented in Report CEN-128, "Response of Combustion Engineering Nuclear Steam Supply System to Transients and Accidents." This report was submitted to the NRC staff for review on April 1, 1980. The results in this report show how a typical C-E-designed plant would most likely respond to various event initiators and shows what systems are actuated following each event. The report includes results of plant simulation analyses with digital computer codes to determine transient behavior of pertinent plant process parameters, components, and systems and results of sequence of events analyses performed to identify component and system functions and alternate means to accomplish specified safety functions.

The analyses contained in Report CEN-128 consider single active failure for each system called upon to function for a particular event. Passive failures and multiple system failures are not considered. The sequence of events analyses (SEA) show the various paths through an event without probabilistic considerations. Each SEA demonstrates how specified safety functions are satisfied. Sequence of events diagrams (SED) are used to show how these functions are accomplished and include single active failures in each responding system and operator failure to perform manual actions. Consequential failures are considered in the SED for the steam line break.

Since early 1980, the C-E Owners Group has conducted a program to develop analyses of transients and accidents involving multiple failures. These analyses were outlined to the NRC staff in a meeting held in Bethesda, Maryland on January 31, 1980. These analyses are currently scheduled to be completed in the first quarter of 1981. The results of these analyses will provide one basis for possible revision of emergency procedure guidelines.

The initial C-E Owners Group development of emergency procedure guidelines was completed in the first quarter of 1980. These emergency procedure guidelines are documented in Report CEN-128. This report was submitted to the NRC staff for review on April 1, 1980.

The emergency procedure guidelines contained in Report CEN-128 were prepared based on extensive reviews of existing emergency procedures, past safety and design analyses, the plant simulation and sequence of events analyses in CEN-128, and interviews with operations personnel at plants with operating C-E reactors. These emergency procedure guidelines were prepared to be used as a basis for reviewing and revising, if necessary, existing plant emergency procedures. APS is using these guidelines to assist in developing PVNGS emergency procedures where appropriate. 2

The NRC staff in a letter dated July 17, 1980, sent questions to the C-E Owners Group concerning the emergency procedure guidelines documented in Report CEN-128. A meeting was held with the NRC staff in Bethesda, Maryland on September 11, 1980, to discuss these questions and answers to them. The C-E Owners Group is presently preparing answers to these questions and revisions to the emergency procedure guidelines in Report CEN-128 as are appropriate. A preliminary response to these questions was submitted by the C-E Owners Group to the NRC staff in a letter dated December 10, 1980. The remaining responses were submitted to the NRC staff by the C-E Owners Group in a meeting on January 30, 1981.

Since early 1980, the C-E Owners Group has conducted an extensive evaluation of specific technical characteristics of emergency procedure guidelines. These include (1) the diagnostic guidance to be provided in emergency procedure guidelines, (2) the need for a separate guideline for inadequate core cooling, and (3) the format for presentation of emergency guidance.
1 This evaluation was completed in the first quarter of 1981. The results of this evaluation serve as one basis for possible revision of emergency procedure guidelines contained in Report CEN-128.

The C-E Owners Group agreed on December 3, 1980, to conduct a series of workshops concerning emergency procedure guidelines in early 1981. The first such workshops were conducted in
2 January, February, March, and April 1981. These workshops provided a formal process by which the emergency procedure guidelines documented in Report CEN-128 will be revised to
1 account for multiple failure considerations. Input to these workshops was provided by the analysis and emergency procedure guidelines studies which have been conducted by the C-E Owners Group since early 1980. The workshops were attended by staff
1 personnel from C-E and from utilities which own C-E reactors. These workshops also provided the opportunity to explore multiple-failure scenarios beyond those which have been currently identified in the C-E Owners Group analyses of transients and accidents. LOCA was also considered in these workshops.

The C-E Owners Group held a meeting with the NRC Division of Systems Integration and Human Factors Safety on January 30, 1981, to discuss the process being used for revision of the emergency 1
procedure guidelines. The revised emergency procedure guidelines were submitted to the staff on June 30, 1981 as CEN-152 in addition to the submittal of CEN-156 titled Emergency Procedure Guideline Development.

The NRC provided comments on these documents in a July 24, 1981 2
meeting and in a letter to C-E Owners Group dated September 15, 1981. The C-E Owners Group is in the process of reviewing these guidelines in light of the NRC comments.

I.C.2 SHIFT RELIEF AND TURNOVER PROCEDURES

Position (NUREG-0694)

Revise plant procedures for shift relief and turnover to require signed checklists and logs to assure that the operating staff (including auxiliary operators and maintenance personnel) possess adequate knowledge of critical plant parameter status, system status, availability and alignment.

PVNGS Evaluation

PVNGS Operations will have detailed administrative procedures available 60 days prior to fuel load that meet the guidance of the November 9, 1979, NRC letter from D. B. Vassallo to all Pending Construction Permit Applicants for shift relief and turnovers to ensure that current plant conditions and system status is conveyed to the oncoming shift. 2

These procedures will include the use of checklists and logs to ensure that there is a proper turnover of command functions and current operating conditions. Turnover and relief will include a review of tagouts, abnormal conditions, jumpers/ bypasses, surveillance testing, and conditions affecting Technical Specifications. Annunciator panels, CRT's, and key operating parameters will also be monitored to verify system status and equipment condition.

I.C.3 SHIFT SUPERVISOR RESPONSIBILITIES

<u>Position (NUREG-0694)</u> (NRC Letter, D. B. Vassallo to All Pending Construction Permit Applicants, dated November 9, 1979)

(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.

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

PVNGS Evaluation

The duties and authorities of the shift supervisor will be defined to emphasize that he has primary onshift responsibility for safe operation of the plant.

The administrative duties and authority of the shift supervisor and his subordinates will be defined in the PVNGS Station Manual. Administrative functions or duties which detract from the shift supervisor's control of the plant are discussed in the response to item I.A.1.2. Lines of command will be clearly established to enable the shift supervisor to fulfill the responsibility for safe operation of the plant.

The Vice President of Electric Operations shall issue a management directive, prior to PVNGS Unit 1 fuel load, that emphasizes the primary management responsibility of the shift supervisor for safe operation of the plant on his shift.

I.C.4 CONTROL ROOM ACCESS

<u>Position</u> (NRC Letter, D. B. Vassallo to All Pending Construction Permit Applicants, dated November 9, 1979)

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

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

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

PVNGS Evaluation

Administrative procedures will limit access to the control room. The shift supervisor will have the responsibility and the authority to control access to those personnel who are required, or are requested, to support the operation of the plant. These procedures will establish clear lines of authority and communication during all plant conditions, including startups, normal, off-normal, and emergency conditions. These procedures will clearly establish the line of succession for the individual in charge of the control room. The line of authority and responsibility in the control room is discussed in our response to item I.C.3.

I.C.5 PROCEDURES FOR FEEDBACK OF OPERATING EXPERIENCE TO PLANT STAFF

Position

In accordance with Task Action Plan I.C.5, Procedures for Feedback of Operating Experience to Plant Staff (NUREG-0660), each applicant for an operating license shall prepare procedures to assure that operating information pertinent to plant safety originating both within and outside the utility organization is continually supplied to operators and other personnel and is incorporated into training and retraining programs. These procedures shall:

(1) Clearly identify organizational responsibilities for review of operating experience, the feedback of pertinent information to operators and other personnel, and the incorporation of such information into training and retraining programs;

(2) Identify the administrative and technical review steps necessary in translating recommendations by the operating experience assessment group into plant actions (e.g., changes to procedures; operating orders);

(3) Identify the recipients of various categories of information from operating experience (i.e., supervisory personnel, shift technical advisors, operators, maintenance personnel, health physics technicians)

or otherwise provide means through which such information can be readily related to the job functions of the recipients;

(4) Provide means to assure that affected personnel become aware of and understand information of sufficient importance that should not wait for emphasis through routine training and retraining programs;

(5) Assure that plant personnel do not routinely receive extraneous and unimportant information on operating experience in such volume that it would obscure priority information or otherwise detract from overall job performance and proficiency;

(6) Provide suitable checks to assure that conflicting or contradictory information is not conveyed to operators and other personnel until resolution is reached; and,

(7) Provide periodic internal audit to assure that the feedback program functions effectively at all levels.

PVNGS Evaluation

An operating experience review program is being developed for PVNGS which will establish the responsibilities and methodologies for reviewing the operating experience of PVNGS and other nuclear plants. The program and implementing procedures will be developed in accordance with this requirement and will be in effect prior to Unit 1 fuel load.

I.C.6 GUIDANCE ON PROCEDURES FOR VERIFYING CORRECT PERFORMANCE OF OPERATING ACTIVITIES

Position

It is required (from NUREG-0660) that licensees' procedures be reviewed and revised, as necessary, to assure that an effective system of verifying the correct performance of operating activities is provided as a means of reducing human errors and improving the quality of normal operations. This will reduce the frequency of occurrence of situations that could result in or contribute to accidents. Such a verification system may include automatic system status monitoring, human verification of operations and maintenance activities independent of the people performing the activity (see NUREG-0585, Recommendation 5), or both.

Implementation of automatic status monitoring if required will reduce the extent of human verification of operations and maintenance activities but will not eliminate the need for such verification in all instances. The procedures adopted by the licensees may consist of two phases -- one before and one after installation of automatic status monitoring equipment, if required, in accordance with item I.D.3.

PVNGS Evaluation

FSAR Section 7.5.1.1.6 describes the Safety Equipment Status Panel, designed to implement Regulatory Guide 1.47, which displays the availability of selected equipment important to nuclear safety. The presence of this system will reduce the extent of human verification of operations and maintenance activities. However, the following additional requirements will be implemented prior to Unit 1 fuel load to insure safety system reliability:

A. Permission to release equipment or systems for maintenance shall be granted by designated operating personnel holding a senior operator license. Prior to granting permission, such operating personnel shall verify that the equipment or system can be released, and determine how long it may be out of service. Granting of such permission shall be documented. Attention shall be given to the potentially degraded degree of protection when one subsystem of a redundant safety system has been removed for maintenance.

B. After permission has been granted to remove the system from service, it shall be made safe to work on. Measures shall provide for protection of equipment and workers. Equipment and systems in a controlled status shall be clearly identified. Strict control measures for such equipment shall be enforced.

C. Conditions to be considered in preparing equipment for maintenance, in addition to requirements of the Technical Specifications, include, for example: shutdown margin; method of emergency core cooling; establishment of a path for decay heat removal; temperature and pressure of the system; valves between work and hazardous material; venting, draining and flushing; entry into closed vessels; hazardous atmospheres; handling hazardous materials; and electrical hazards. When entry into a closed system is required, control measures shall be established to prevent entry of extraneous material and to assure that foreign material is removed before the system is reclosed.

D. Procedures shall be provided for control of equipment, as necessary, to maintain personnel and reactor safety and to avoid unauthorized operation of equipment. These procedures shall require control measures such as locking or tagging to secure and identify equipment in a controlled status. The procedures shall require independent verifications, where appropriate, to ensure that necessary measures, such as tagging equipment, have been implemented correctly.

E. Temporary modifications, such as temporary bypass lines, electrical jumpers, lifted electrical leads, and temporary trip point settings, shall be controlled by approved procedures which shall include a requirement for independent verification by either a second person or by a functional test which conclusively proves the proper installation or removal of the temporary modification. A log, or other documented evidence, shall be maintained of the current status of such temporary modifications.

F. When equipment is ready to be returned to service, operating personnel shall place the equipment in operation and verify and document its functional acceptability. Attention shall be given to restoration of normal conditions, such as removal of jumpers or signals used in maintenance or testing to such as returning valves, breakers or switches or proper start-up or operating positions from "test" or "manual" positions, and assuring that all alarms which are indicative of inoperative status are extinguished. For safety-related equipment, proper alignment shall be independently verified by a second person unless alignment is proven by functional testing.

I.C.7 NSSS VENDOR REVIEW OF PROCEDURES

Position (NUREG-0694)

(1) Low Power Test Program

Obtain nuclear steam supply system (NSSS) vendor review of low-power testing procedures to further verify their adequacy.

(2) Power Ascension and Emergency Procedures

Obtain NSSS vendor review of power-ascension test and emergency procedures to further verify their adequacy.

PVNGS Evaluation

(1) and (2) Low Power Test Program and Power Ascension and Emergency Procedures

FSAR Section 14.2.2.8 describes the responsibilities of the NSSS vendor representative on the Test Working Group which includes review of test procedures pertaining to or interfacing with their supplied systems.

APS will obtain the NSSS vendor review of emergency operating procedures that involve the NSSS vendor scope of supply.

I.C.8 PILOT MONITORING OF SELECTED EMERGENCY PROCEDURES FOR NEAR-TERM OPERATING LICENSE APPLICANTS

Position (NUREG-0694)

Correct emergency procedures, as necessary, based on the NRC audit of selected plant emergency operating procedures (e.g., small-break LOCA, loss of feedwater, restart of engineered safety features following a loss of AC power, steam-line break, or steam-generator tube rupture).

PVNGS Evaluation

Arizona Public Service Company will review and work with the NRC to coordinate improvements that can be made to emergency procedures identified as a result of the described audit.

I.D. CONTROL ROOM DESIGN

I.D.1 CONTROL-ROOM DESIGN REVIEWS

Position

In accordance with Task Action Plan I.D.1, Control Room Design Reviews (NUREG-0660), all licensees and applicants for operating licenses will be required to conduct a detailed control-room design review to identify and correct design deficiencies. This detailed control-room design review is expected to take about a year. Therefore, the Office of Nuclear Reactor Regulation (NRR) requires that those applicants for operating licenses who are unable to complete this review prior to issuance of a license make preliminary assessments of their control rooms to identify significant human factors and instrumentation problems and establish a schedule approved by NRC for correcting deficiencies. These applicants will be required to complete the more detailed control room reviews on the same schedule as licensees with operating plants.

PVNGS Evaluation

Arizona Public Service (APS) formed a Control Room Design
Review (CRDR) Management Team (Team) in June, 1980 and performed 1
a preliminary design assessment (PDA) of the PVNGS control room. The Team was comprised of APS engineering, APS operations, Bechtel Power Corporation (BPC) engineering, and BPC consultant, Torrey Pines Technology (TPT) of San Diego, California.

This PDA was performed in compliance with NUREG-0660 and 0737 (Item I.D.1) to identify, prioritize and correct discrepancies using NUREG/CR-1580 guidelines as deemed appropriate.

The program was conducted in three phases:

1. <u>Phase I - Indoctrination and checklist development</u>

   The indoctrination process included development of a program plan used to direct and coordinate the overall study. In addition, this phase was used by the consultant to secure plant documents and receive instructions for their usage, to determine administrative procedures, to establish a reference library, and to familiarize appropriate personnel at the plant site. Simultaneous with indoctrination was the development of guidelines and criteria for developing checklists (thirteen) in the areas of human factors, operator preparedness and system factors. This phase started in August of 1980 and was completed in December of 1980.

2. <u>Phase II - Site reviews and data evaluation</u>

   During this phase all checklists were revised as required and utilized to obtain data at the plant simulator. The systems factor checklist was formed into a working committee obtaining data by discussions with personnel involving Bechtel, C-E and ANPP engineering, plant operators and the consultant (T.P.T.). This phase also included evaluation of the data obtained by the data collection teams and the results review committee. The checklists discrepancies determined to be Human Engineering Discrepancies (HED) were forwarded to the program management team for concurrance and implementation. This phase began in December of 1980 and was completed in February of 1981. Two separate subcommittees were formed as a result of HED findings, namely the demarcation study and the alarm prioritization study. These efforts were conducted after completion of Phase II.

3. <u>Phase III - Report preparation</u>

   This phase consisted of collecting material from Phases I & II and preparing the final report for review and publication. This report will be available in two parts, namely, an executive summary and a technical report. In addition this phase included preparing material for NRC reviews and audits. Efforts for the final report began in January 1981 and remain open for continued efforts as may be deemed necessary by the study's management team.

The program consisted of three parallel tasks:

1. <u>Human Factors</u> - an assessment of control room and environment in terms of accepted human engineering principles.

2. <u>Operator Preparedness</u> - an assessment of operating environment and of operator training and selection

practices in terms of obtaining and maintaining a competent operating staff.

3. System Factors an assessment of plant systems and control board configurations in terms of required system functions and operator functions.

The data collected by each task were evaluated for discrepancies from the established criteria. These discrepancies were reviewed by:

1. A Results Review Committee who categorized them and made disposition recommendations to the Management Team.

1 2. The Management Team who made final corrective action recommendations to APS management for implementation scheduling.

The review concluded the following:

- 139 deviations from the human engineering guidelines of which:
  - 90 are categorized as Human Engineering Discrepancies and corrective actions are mandatory.
  - 49 are categorized as minor and corrective actions are not mandatory.

This review included TMI-required control board changes.

The major Human Engineering Discrepancies (HED) are:

1. Foxboro Displays/controllers - Inadequate labeling, scale readibility and functional distinctiveness.

2. Inadequate system demarcation on control boards.

3. Annunciator System - Inadequate alarm priority, displays, window consistency, grouping and demarcation.

Most of the HED corrective actions have been identified and need only be scheduled for implementation. Instrument relocations identified are being factored into the design. A control board 1
demarcation study was conducted in mid-march 1981 as a result of the PDA finding. A design change is currently being prepared to perform the control board color coded demarcation changes. Also an annunciator prioritization study is currently under way to resolve the HED's identified during the PDA. It is planned to have changes completed by April 1, 1982.

Remaining work within the scope of this review consists of walk-through of the plant emergency operating procedures when they 2
are available. This effort is not expected to surface additional discrepancies of any significance.

1
The control room and main control board layout design and implementation are considered to have incorporated good human engineering practice.

The early part of this effort was divided into three phases. Phase I of the study developed the guidelines to be used while conducting the CRDR from NRC control room audit reports of other utility facilities and NUREG/CR-1580.

Phase II of the study consisted of the detailed data taking effort and the identification of human factors deficiencies. The three task areas addressed were human factors, systems factors, and operator preparedness factors. The deficiencies identified were analyzed for proper resolution and assigned priorities to assist in determining a schedule for implementation.

Phase III, which is currently in progress, includes preparation and publication of a preliminary report.

Because of the stage of development of emergency operating procedures (EOPs), that portion of the CRDR involving the
2 walk-through and videotaping of EOPs will start in the second
quarter of 1982. When this later part of the CRDR is completed, a final report for submittal to the NRC will be prepared.
2 The submittal date is targeted for July, 1982.

The review has resulted in APS initiating implementation of the following to date:

- Color demarcation
- Instrument relocation
- Alarm prioritization
- Additional instrumentation

I.D.2 PLANT SAFETY PARAMETER DISPLAY CONSOLE

Position

In accordance with Task Action Plan I.D.2, Plant Safety Parameter Display Console (NUREG-0660), each applicant and licensee shall install a safety parameter display system (SPDS) that will display to operating personnel a minimum set of parameters which define the safety status of the plant. This can be attained through continuous indication of direct and derived variables as necessary to assess plant safety status.

PVNGS Evaluation

PVNGS will establish a Plant Safety Parameter Display System (SPDS) to display and transmit plant status to appropriate individuals in the event of an accident. The applicants will be guided by the staff's positions delineated in NUREG-0696 relative to the emergency response facilities described below. A description of the SPDS and its procedures for implementation is given as is follows: 1

A. Location

The PVNGS Units 1, 2 & 3 SPDS will be located in each of the unit's control room on elevation 140'-0".

B. Activation

The SPDS will be activated at all times in accordance with NUREG-0654. The instrumentation displaying the plant status (described below) is continuously operating and thus will provide plant parameter display capability from accident initiation.

C. Instrumentation

This system receives data inputs from the plant using the existing process instrument loops and certain new wide range monitoring instruments added to meet Regulatory Guide 1.97, Revision 2 and NUREG-0737 requirements. The software for the SPDS provides interactive displays of the critical safety parameters. Identical displays are provided in the TSC, EOF, Satellite TSC and control room. The SPDS also has an alarm/diagnosis heirarchy such that any monitored critical parameter that exceeds a predetermined set point will activate an alarm indication on the display. This will provide the ability to identify the specific malfunctioning equipment. These data shall be stored such that the plant steady-state operating conditions prior to the accident, the transient conditions producing the initiating event, and the plant systems dynamic behavior throughout the course of the accident will be available on the SPDS.

D. Instrumentation Power Supply

The SPDS is a non-IE system and utilizes either non-IE process inputs or IE signals that have been appropriately isolated to prevent any potential degradation of the

IE system. However, the SPDS will be supplied by an uninterruptible power supply capable of sustaining data processing operation for at least 60 minutes with subsequent manual connection to a diesel back-up power supply for long-term SPDS operation. The control room SPDS display and data acquisition multiplexers will be powered from a reliable instrument bus with Class IE backup power.

E. Seismically-Qualified Backup SPDS

In addition to the SPDS described above, a Qualified Safety Parameter Display System (QSPDS) will be provided 1 in the control room to serve as a seismically qualified backup to the SPDS for a severe seismic event. The QSPDS also provides control room display of inadequate core cooling to meet the requirements of NUREG-0737 and additional new wide-range monitoring instruments added to meet Regulatory Guide 1.97, Revision 2 and NUREG-0737. The displayed parameters are a subset of the primary SPDS parameters. The QSPDS is a redundant, safety-grade system powered from redundant Class IE buses.

I.G PREOPERATIONAL AND LOW-POWER TESTING

I.G.1 TRAINING DURING LOW-POWER TESTING

Position NUREG-0694

(1) Propose Tests

Define and commit to a special low-power testing program approved by NRC to be conducted at power levels no greater than 5 percent for the purposes of providing meaningful technical information beyond that obtained in the normal startup test program and to provide supplemental training.

(2) Submit Analysis and Procedures

(3) Training and Results

Supplement operator training by completing the special low-power test program. Tests may be observed by other shifts or repeated on other shifts to provide training to the operators.

PVNGS Evaluation

1. Propose Tests

APS will propose to the NRC a special low-power test program to be conducted at power levels no greater than 5 percent for the purpose of providing meaningful technical information beyond that obtained in the remainder of the startup test program.

2. Submit Analysis and Procedures

   The test shall be proposed and a plan shall be submitted to the NRC prior to Unit 1 fuel load.

3. Training and Results

   Training will be scheduled for each shift crew on selected plant evolutions that will not adversely affect safe conduct of the testing, and that cannot be satisfactorily accomplished on the plant simulator. The results of the training will be submitted prior to operation at full power.

II. SITING AND DESIGN

II.B CONSIDERATION OF DEGRADED OR MELTED CORES IN SAFETY REVIEW

II.B.1 REACTOR COOLANT SYSTEM VENTS

Position

Each applicant and licensee shall install reactor coolant sys-
tem (RCS) and reactor vessel head high point vents remotely
operated from the control room. Although the purpose of the
system is to vent noncondensible gases from the RCS which may
inhibit core cooling during natural circulation, the vents must
not lead to an unacceptable increase in the probability of a
loss-of-coolant accident (LOCA) or a challenge to containment
integrity. Since these vents form a part of the reactor
coolant pressure boundary, the design of the vents shall con- 1
form to the requirements of Appendix A to 10 CFR Part 50,
"General Design Criteria." The vent system shall be designed
with sufficient redundancy that assures a low probability of
inadvertent or irreversible actuation.

Each licensee shall provide the following information concerning the design and operation of the high point vent system:

(1) Submit a description of the design, location, size, and power supply for the vent system along with results of analyses for loss-of-coolant accidents initiated by a break in the vent pipe. The results of the analyses

should demonstrate compliance with the acceptance criteria of 10 CFR 50.46.

(2) Submit procedures and supporting analysis for operator use of the vents that also include the information available to the operator for initiating or terminating vent usage.

PVNGS Evaluation

PVNGS Unit 1, 2 and 3 are in the process of completing the design and installation of a Reactor Coolant Gas Vent System (RCGVS), that will be operable prior to fuel load. A description of the design is as follows:

A. SYSTEM DESIGN BASIS

The RCGVS is designed to be used to remotely vent gases from the reactor vessel head and pressurizer steam space
1 during post-accident situations when large quantities of non-condensible gases may collect in these high points. Although primarily designed to be available during post-accident conditions, the system can also be used to aid in the RCS venting procedures following a maintenance outage.

The design criteria for the vent system are as follows:

1. The system shall permit remote (control room) venting from the reactor vessel head vent or the pressurizer.

2. The vent flow rate capability shall be based upon the following considerations:

   a) The vent rate should be sufficient to preclude the gas accumulation from interfering with core cooling.

   b) Coolant loss through the vent should not exceed makeup capacity.

   c) The vent mass rate should not result in heat loss from the RCS in excess of the pressurizer heater capacity.

3. The vents shall conform to the applicable requirements of 10CFR50, Appendix A, General Design Criteria. In particular, these vents shall be safety grade and satisfy the single failure criterion and the requirements of IEEE-279 in order to ensure a low probability of inadvertent action. 1

4. The design shall minimize modification of currently installed safety class equipment and piping which may be radioactive or require major reactor coolant system hydrostatic testing after installation.

5. The system shall be analyzed to determine effects of pipe breakage and the results should be demonstrated acceptable in accordance with the acceptance criteria of 10CFR 50.46.

6. The vent system shall be capable of selectively venting to either the containment or the reactor
1 drain tank.

7. The vent system shall be designed to vent superheated steam, steam water mixtures, water, fission gases,
2 helium, nitrogen, and hydrogen at pressures and temperatures as high as 2500 psig and 700°F.

8. Control room position indication shall be provided
1 for all power operated valves.

9. The system shall be designed not to interfere with
2 refueling maintenance activities.

The reactor coolant vent system is designed to vent non-condensible gases from the reactor coolant system during post-accident conditions. The purpose of venting is to prevent possible interference with core cooling. Small amounts of gas can be vented to the reactor drain tank and thus not enter the containment atmosphere. Larger volumes
1 will require venting to the containment - either through the ruptured reactor drain tank rupture disk or directly - where the hydrogen concentration will be controlled by the containment hydrogen recombiner. Pressure instrumentation is included in the design to monitor system performance.

Although designed for accident conditions, the system may be used to aid in the refueling venting of the reactor coolant system. Although venting of the CEDMs and RCPs will still be necessary, pressurizer and reactor vessel venting can be accomplished with the system if desired. Vent flow would be directed to the reactor drain tank for this operation to prevent inadvertent release of radioactive fluid to the containment.

B. SYSTEM DESCRIPTION

1. System Parameters

The components in the RCGVS consists of piping, valves, and instrumentation to direct vented gas flow from existing components to an existing tank as shown on figure II.B.1-1. The main design parameters for the vent line are: 1

Design Flow	500 SCFM
Design Temperature	700°F
Design Pressure	2500 psig
Line Size	1"

2. Descriptive Summary

The system permits the main control room operator to remotely vent the pressurizer and reactor vessel head. The pressurizer vent ties into an existing 3/4" vent line. The reactor vessel head vent ties into the existing 3/4" reactor vessel vent and will be flanged to permit head removal for refueling. The vent line is connected to the pressurizer relief valve common discharge header, which terminates in the reactor drain tank, entering below the water level. Existing reactor drain tank penetration is thus utilized. The capability to vent directly to containment is also provided should the operator not desire to allow the rupture disc on the reactor drain tank to rupture when large quantities of gas must be vented. This line terminates at an unobstructed location above the operating deck to promote good mixing within the containment.

The normal vent paths are from either the pressurizer or reactor vessel head to the reactor drain tank.
1 These paths are powered from emergency power sources, seismic category I, and of the appropriate quality class to conform to existing standards. The vent flow is directed into the reactor drain tank below the water level to remove energy from the steam and to cool the gas itself. If large quantities of gas must be vented, the reactor drain tank will pressurize and eventually rupture its rupture disc providing a path to containment to continue the venting process. If the rupture is deemed undesirable by the operators, venting can be remotely shifted directly to the containment upon actuation of the reactor drain tank high pressure alarm, which annunciates in

the main control room. This path is powered from emergency power sources, Seismic Category I, and of the appropriate quality class to conform to existing standards. No single active failure can prevent the RCGVS from performing its design function.

For cases when the vented gas is not highly radioactive and containment isolation has not occurred, the gas delivered to the reactor drain tank can be routed to the gaseous radwaste system surge tank. Although unavailable during accident conditions, this path will allow processing of gas removed by the vent system if the system is used to aid in RCS venting for refueling operations. 1

To preclude inadvertent opening of any one of six solenoid-operated vent valves, they are placed in their locked-closed position via key lock switches in the main control room. Opening of any solenoid-operated vent valve requires deliberate operator action.

3. <u>Operation</u>

The operator can vent non-condensible gases from the reactor coolant by the following flow paths using the remotely operable vent system:

a) Reactor vessel head vent to the reactor drain tank

b) Pressurizer vent to the reactor drain tank

c) Reactor vessel or pressurizer vent to containment directly

d) Reactor vessel head or pressurizer vent to the gaseous radwaste system surge tank via the reactor drain tank.

Accident events with large gas generation may demand rapid response by the operator to remove the gases and to establish stable plant conditions in a controlled manner. Maintaining core water cover establishes and maintains core cooling which is of primary importance. Other events that may not cause the core
1 to uncover permit a slower response.

Operating considerations for each flow path are given below:

a) Reactor Vessel Head Vent to the Reactor Drain Tank

Venting the reactor vessel to the reactor drain tank is accomplished by opening one of two reactor vessel head vent solenoid isolation valves and the solenoid-operated valve to the reactor drain tank. System pressure integrity is confirmed by vent line pressure indication.

Integrity of the reactor drain tank is confirmed by monitoring tank pressure, temperature and level. Continuous venting to the reactor drain tank may eventually result in rupture of the rupture disk, particularly if a substantial amount of noncondensible gases are generated in the reactor vessel. In this instance, direct venting to the containment would be preferable. The operator has the capability to monitor the hydrogen levels in containment and start the hydrogen recombiner.

b) Pressurizer Vent to the Reactor Drain Tank

If the operator desires to degas the reactor coolant system or remove accumulated gases in the pressurizer, the pressurizer vent can be aligned to the reactor drain tank. This is accomplished by opening one of two pressurizer vent solenoid isolation valves and the solenoid-operated valve to the reactor drain tank. System pressure integrity is confirmed by vent line pressure indication. Reactor drain tank conditions are monitored by pressure, level and temperature instruments. Continuous venting to the reactor drain tank may lead to failure of the rupture disc. In this case, the precautions noted in a) above are to be followed. 1

c) Reactor Vessel Head or Pressurizer Vent to Containment Directly

Extended venting from either source to the reactor drain tank may eventually rupture the rupture disc. While this may be of no real consequence during a major accident, there may be times when the operator would desire to maintain the rupture disc intact. In this event, the venting may be directed to the containment directly by opening the solenoid-operated containment vent isolation valve. If this path is
1 used, the precautions with respect to hydrogen in the containment noted in a) above must be followed.

d) Reactor Vessel Head or Pressurizer Vent to the Gaseous Radwaste System (GRS)

This path may be used if the gases are known to be low in activity, (i.e., will not exceed GRS Technical Specification values), normal power is available, low removal rates are acceptable, and adequate storage space is available in the GRS. Alignment of this path is accomplished by initiating vent flow from the pressurizer or reactor vessel head to the reactor drain tank as in a) or b) above. Gases directed to the reactor drain tank can be sent to the GRS and collected in the Gas Surge Tank for eventual storage in the Gas Decay Tanks. This mode of operation is intended for use when the vented gas is not highly radioactive and containment isolation has not occurred.

Consistent with NRC requirements, procedural guidelines for the system will be provided including information relative to initiating or terminating vent system usage.

4. Maintenance

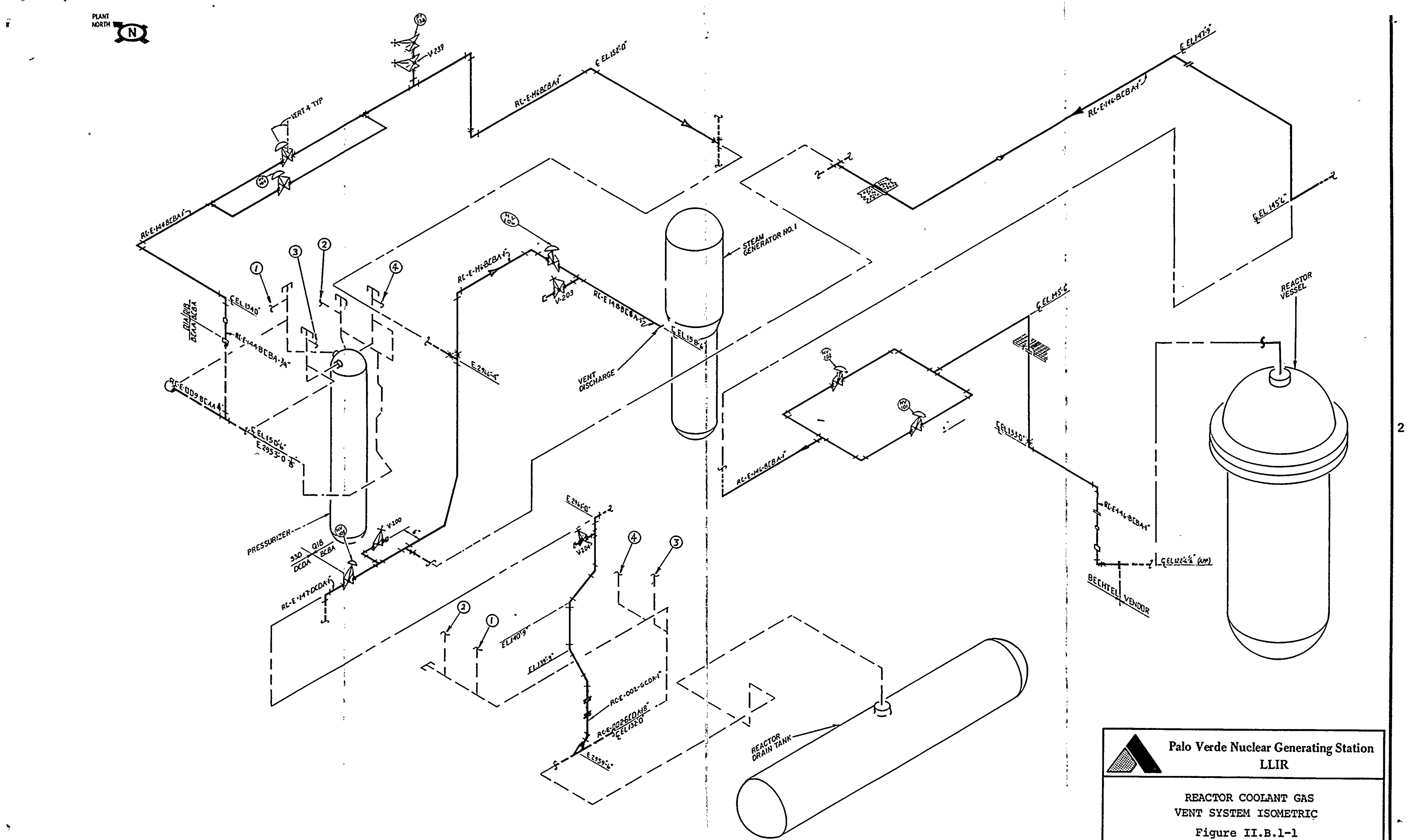
System maintenance is limited to inservice inspection of the system valves. Adequate test connections are
provided in the system design to accomplish the 1
requisite testing. To avoid interferences with refueling operations, specific consideration will be given to the layout and attachment of the vent system piping and valves to the reactor vessel head. The system will be designed not to interfere with the reactor head removal procedures.

C. SYSTEM CONTROLS

The system is designed to be controlled remotely from the main control room. All valves and instrumentation are powered from Class IE power sources. Position indication is provided for all remotely operated valves (open/shut) and displayed in the control room. Pressure instrumentation is also provided to monitor system performance and displayed in the control room. In addition, reactor drain tank pressure, temperature, and level indication is provided in the control room.

D. LOCA ANALYSIS

1 Consistent with NRC requirements, the system design is acceptable in accordance with 10CFR 50.46. Both the reactor vessel head vent and pressurizer vent nozzles are equipped with 7/16 inch orifices which limit flow to less than the LOCA definition in accordance with 10CFR 50.46.



Palo Verde Nuclear Generating Station
LLIR

REACTOR COOLANT GAS
VENT SYSTEM ISOMETRIC

Figure II.B.1-1

II.B.2 DESIGN REVIEW OF PLANT SHIELDING AND ENVIRONMENTAL QUALIFICATION OF EQUIPMENT FOR SPACES/SYSTEMS WHICH MAY BE USED IN POSTACCIDENT OPERATIONS

Position

With the assumption of a postaccident release of radioactivity equivalent to that described in Regulatory Guides 1.3 and 1.4 (i.e., the equivalent of 50% of the core radioiodine, 100% of the core noble gas inventory, and 1% of the core solids are contained in the primary coolant), each licensee shall perform a radiation and shielding-design review of the spaces around systems that may, as a result of an accident, contain highly radioactive materials. The design review should identify the location of vital areas and equipment, such as the control room, radwaste control stations, emergency power supplies, motor control centers, and instrument areas, in which personnel occupancy may be unduly limited or safety equipment may be unduly degraded by the radiation fields during postaccident 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 postaccident 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 before the incident at TMI-2, the shielding of areas that require personnel access for accident mitigation (such as the control room) was reviewed to determine that access would not be unduly limited. For maintenance actions, this review principally considered shielding separation between redundant ESF components to ensure that repairs to a failed component would not be unduly restricted by radiation from an operating component. In addition, as noted in FSAR Section 3.11, safety related equipment is qualified for the maximum expected radiation dose.

1 An analysis of the PVNGS shielding design was performed to
determine if TMI level source strengths would inhibit mainten-
ance access or violate 10 CFR 50, Appendix A, General Design
Criterion 19 (GDC 19). The review demonstrated that personnel
2 radiation exposures in vital areas, during post-accident
activities will meet the criteria of NUREG 0737 and GDC-19
design basis. The analysis also reviewed equipment qualifi-
cation dose limits in accordance with Commission Order and
Memorandum CLI-80-21 and NUREG 0588.

1 A. Source Terms

Initial core releases used in the analyses are equivalent
to those recommended in Regulatory Guides 1.4 and 1.7 and
Standard Review Plan 15.6.5 and considered two LOCA events.
1 The first was a LOCA with recirculation accomplished via
the containment sump. The second was a LOCA with an intact
primary with recirculation accomplished via the shutdown

cooling system. The following core releases were used in the review:

1. Source A: Containment airborne: 100% noble gases, 25% iodines - see table II.B.2-1.

2. Source B: Reactor coolant: 100% noble gases, 50% halogens, 1% solids - see table II.B.2-2.

3. Source C: Containment sump: 50% halogens, 1% solids - see table II.B.2-3.

Volumes used for each source were:

1. Source A: Containment free volume of 2.6×10^6 ft^3.

2. Source B: Reactor coolant system (RCS) volume of 1
   9.15×10^3 ft^3.

3. Source C: The minimum volume of water, 7.76×10^4 ft^3 present at the time of recirculation. (RCS + refueling water tank + safety injection tanks.)

A LOCA with sump recirculation is represented by sources A and C. An intact primary-degraded core LOCA is represented by sources A and B (source A was not reduced even though there is no mechanism to assume noble gases in both sources). The systems assumed to be operating for each event are shown in table II.B.2-4.

Decay curves, normalized to initial time equal to zero, were developed for the sources as an aid in developing

TABLE II.B.2-1

LOCA SOURCE A - CONTAINMENT AIRBORNE
(Curies)

Nuclide	Activity[a]
Kr-85m	3.48(+7)
Kr-85	1.21(+6)
Kr-87	5.91(+7)
Kr-88	8.59(+7)
Kr-89	1.08(+8)
Kr-90	1.16(+8)
Kr-91	8.95(+7)
I-129	4.03(-1)
I-131	2.83(+7)
Xe-131m	6.17(+5)
I-132	2.90(+7)
I-133	5.80(+7)
Xe-133	2.26(+8)
I-134	6.35(+7)
I-135	5.53(+7)
Xe-135m	6.38(+7)
Xe-135	5.31(+7)
I-137	3.60(+7)
Xe-137	2.09(+8)
I.138	2.19(+7)
Xe-138	2.05(+8)
Xe-140	1.16(+8)
Xe-143	2.75(+6)
Xe-144	4.03(+5)

a. Numbers in parenthesis denote powers of ten

TABLE II.B.2-2

LOCA SOURCE B - REACTOR COOLANT

(Curies)

Nuclide	Activity[a]	Nuclide	Activity	Nuclide	Activity
Se-84	2.55(+5)	Sn-129	1.08(+5)	Cs-140	2.00(+6)
Br-84	1.28(+7)	Sb-129	2.08(+5)	Ba-140	2.11(+6)
As-85	5.52(+4)	Te-129m	3.68(+4)	La-140	2.11(+6)
Se-85	3.02(+5)	Te-129	2.03(+5)	Xe-143	2.75(+6)
Br-85	1.61(+7)	I-129	8.05(-1)	Cs-143	4.52(+5)
Kr-85m	3.48(+7)	Sn-131	3.50(+5)	Ba-143	1.47(+6)
Kr-85	1.21(+6)	Sb-131	9.30(+5)	La-143	1.81(+6)
Se-87	3.29(+5)	Te-131m	1.72(+5)	Ce-143	1.83(+6)
Br-87	2.87(+7)	Te-131	1.00(+6)	Pr-143	1.84(+6)
Kr-87	5.91(+7)	I-131	5.65(+7)	Xe-144	4.03(+5)
Br-88	3.87(+7)	Xe-131m	6.17(+5)	Cs-144	1.25(+5)
Kr-88	3.59(+7)	Sn-132	3.08(+5)	Ba-144	8.75(+5)
Rb-88	8.61(+5)	Sb-132	8.24(+5)	La-144	1.61(+6)
Br-89	3.96(+7)	Te-132	1.15(+6)	Ce-144	1.25(+6)
Kr-89	1.08(+8)	I-132	5.80(+7)	Pr-144	1.25(+6)
Rb-89	1.12(+6)	Sn-133	5.37(+4)		
Sr-89	1.13(+6)	Sb-133	1.10(+6)		
Br-90	3.89(+7)	Te-133m	1.42(+6)		
KR-90	1.16(+8)	Te-133	1.01(+6)		
Rb-90	1.31(+6)	I-133	1.16(+8)		
Sr-90	6.60(+4)	Xe-133	2.26(+8)		
Y-90	6.57(+4)	Cs-134	1.28(+4)		
Kr-91	8.95(+7)	Sb-134	5.69(+5)		
Rb-91	1.45(+6)	Te-134	2.14(+6)		
Sr-91	1.49(+6)	I-134	1.27(+8)		
Y-91m	8.80(+5)	Sb-135	1.80(+5)		
Y-91	1.49(+6)	Te-135	1.09(+6)		
Sr-95	1.64(+6)	I-135	1.10(+8)		
Y-95	2.02(+6)	Xe-135m	6.38(+7)		
Zr-95	1.99(+6)	Xe-135	5.31(+7)		
Nb-95	1.95(+6)	Cs-135	2.50(-1)		
Zr-99	2.61(+6)	Cs-136	1.32(+4)		
Nb-99	2.61(+6)	I-137	7.20(+7)		
Mo-99	2.69(+6)	Xe-137	2.09(+8)		
Tc-99m	3.23(+5)	Cs-137	4.84(+4)		
Mo-103	1.46(+6)	Ba-137m	4.52(+4)		
Tc-103	1.53(+6)	I-138	4.39(+7)		
Ru-103	1.53(+6)	Xe-138	2.05(+8)		
Tc-106	6.72(+5)	Cs-138	2.07(+6)		
Ru-106	5.27(+5)	Xe-140	1.16(+8)		

a. Numbers in parenthesis denote powers of ten

TABLE II.B.2-3

LOCA SOURCE C - CONTAINMENT SUMP
(Curies)

Nuclide	Activity[a]	Nuclide	Activity	Nuclide	Activity
Se-84	2.55(+5)	Sn-129	1.08(+5)	Cs-140	2.00(+6)
Br-84	1.28(+7)	Sb-129	2.08(+5)	Ba-140	2.11(+6)
As-85	5.52(+4)	Te-129m	3.68(+4)	La-140	2.11(+6)
Se-85	3.02(+5)	Te-129	2.03(+5)	Cs-143	4.52(+5)
Br-85	1.61(+7)	I-129	8.05(-1)	Ba-143	1.47(+6)
Se-87	3.29(+5)	Sn-131	3.50(+5)	La-143	1.81(+6)
Br-87	2.87(+7)	Sb-131	9.30(+5)	Ce-143	1.83(+6)
Br-88	3.87(+7)	Te-131m	1.72(+5)	Pr-143	1.84(+6)
Rb-88	8.61(+5)	Te-131	1.00(+6)	Cs-144	1.25(+5)
Br-89	3.96(+7)	I-131	5.65(+7)	Ba-144	8.75(+5)
Rb-89	1.12(+6)	Sn-132	3.08(+5)	La-144	1.61(+6)
Sr-89	1.13(+6)	Sb-132	8.24(+5)	Ce-144	1.25(+6)
Br-90	3.89(+7)	Te-132	1.15(+6)	Pr-144	1.25(+6)
Rb-90	1.31(+6)	I-132	5.80(+7)		
Sr-90	6.60(+4)	Sn-133	5.37(+4)		
Y-90	6.57(+4)	Sb-133	1.10(+6)		
Rb-91	1.45(+6)	Te-133m	1.42(+6)		
Sr-91	1.49(+6)	Te-133	1.01(+6)		
Y-91m	8.80(+5)	I-133	1.16(+8)		
Y-91	1.49(+6)	Cs-134	1.28(+4)		
Sr-95	1.64(+6)	Sb-134	5.69(+5)		
Y-95	2.02(+6)	Te-134	2.14(+6)		
Zr-95	1.99(+6)	I-134	1.27(+8)		
Nb-95	1.95(+6)	Sb-135	1.80(+5)		
Zr-99	2.61(+6)	Te-135	1.09(+6)		
Nb-99	2.61(+6)	I-135	1.10(+8)		
Mo-99	2.69(+6)	Cs-135	2.50(-1)		
Tc-99m	3.23(+5)	Cs-136	1.32(+4)		
Mo-103	1.46(+6)	I-137	7.20(+7)		
Tc-103	1.53(+6)	CS-137	4.84(+4)		
Ru-103	1.53(+6)	Ba-137m	4.52(+4)		
Tc-106	6.72(+5)	I-138	4.39(+7)		
Ru-106	5.27(+5)	Cs-138	2.07(+6)		

a. Numbers in parenthesis denote powers of ten

TABLE II.B.2-4

SOURCES USED IN POST-ACCIDENT SHIELDING REVIEW[a][b]

Source Type	LOCA with Sump Recirculation	LOCA - Degraded Core - Intact Primary
A	• Containment Air • Hydrogen control system	• Containment air • Hydrogen control system
B	-	• Safety Injection System • Containment Spray System • Shutdown Cooling System • Post-Accident Sampling System • Letdown System[c]
C	• Safety Injection System • Containment Spray System • Shutdown Cooling System • Post-Accident Sampling System • Letdown System[c]	-

a. Where redundant systems exist, both are assumed in use.
b. Radwaste systems not used post-accident.
c. Portions up to purification filter inlet.

post-accident access plant. These curves are presented as figures II.B.2-1 (source A), II.B.2-2 (source B), and II.B.2-3 (source C).

B. <u>Shield Review</u>

The facility layout will assist in keeping occupational
1 exposures ALARA even after a design basis accident. While
exposures will be significantly higher than during normal
operation, required access is provided to vital areas and
systems without exceeding 5 rem/hr. Zone maps showing
expected dose rates in the event of a LOCA with sump
recirculation are provided as figures II.B.2-4 through
2 II.B.2-23. Zone maps for the hypothetical condition of a
LOCA with an intact primary but with a degraded core are
provided as figures II.B.2-14 through II.B.2-23. The
source terms correspond to those noted in section II.B.2A.
The dose rates projected for these two sets of drawings do
1 not assume decay beyond that corresponding to the onset of
recirculation. Even so, virtually unrestricted access
will be permitted within portions of the upper floor of the
auxiliary building (such as the area of the operational
2 support center) and the lower levels of the control building.
Continuous occupancy will be permitted in the control
1 room, satellite Technical Support Center (TSC), TSC,
diesel generator building and emergency operations facility
(EOF) as dose rates will be 15 mrem/hr or less.

To provide sampling capability with exposures kept ALARA,
PVNGS will incorporate a remote, automated, post-accident
sampling system that meets the requirements of NUREG 0737
and Regulatory Guide 1.97, Revision 2. This system can be
operated from the radiochemistry counting room, the control
room, or the technical support center. Backup grab sample
capability will be provided in the hot lab sample room
using microchemistry techniques to keep the volume of 1
source as small as possible. Refer to section II.B.3 for
further details.

The only other area where access might be required is to
the hydrogen monitors/recombiners. Projected dose rates
at the onset of recirculation are expected to be approxi-
mately 10 to 30 rem/hr (sump recirculation). As the
recombiners do not have to be installed until at least
72 hours after the DBA, (refer to FSAR Figure 6.2.5-2) 2
dose rates will drop due to decay to about 1/10 the doses
noted above. Thus, the installation dose rate (assuming
sump recirculation) will be less than 5 rem/hr. While
the dose rate would be greater than 5 rem/hr for an
intact primary-degraded core event, the recombiners
would not need to be installed since an intact primary 1
would not be consistent with hydrogen generation inside
the containment. If hydrogen generation were pos-
tulated, this would necessitate a break or opening in
the primary. Consequently, sump recirculation would be

available with the concomitant release of noble gases and dilution by the refueling water tank. These consequences would lead to the doses noted above for the sump recirculation mode of cooling (i.e., dose rates less than 5 rem/hr).

ESF grade filtered ventilation is provided for auxiliary building rooms below elevation 100' (refer to FSAR section 9.4). This will reduce airborne sources due to recirculation and/or containment leakage. Non-ESF grade filtered ventilation is available for use to reduce airborne sources above elevation 100' in the auxiliary building (refer to FSAR section 9.4). The use of non-ESF filtration is acceptable since there are no recirculation
1 components above elevation 100'. Thus the only significant source of airborne activity is containment leakage. This leakage has already been accounted for in offsite dose analyses which assumed direct containment leakage to the atmosphere. Secondly, this filter discharges via the plant vent. The plant vent will be monitored in accordance with NUREG 0737 and Regulatory Guide 1.97, Revision 2 to provide notification of decreased filter efficiency. Refer to section II.F.1.

Therefore, considering direct and airborne sources, access can be provided to those vital areas necessary for control of the plant.

C. <u>Meeting NRC Requirements</u>

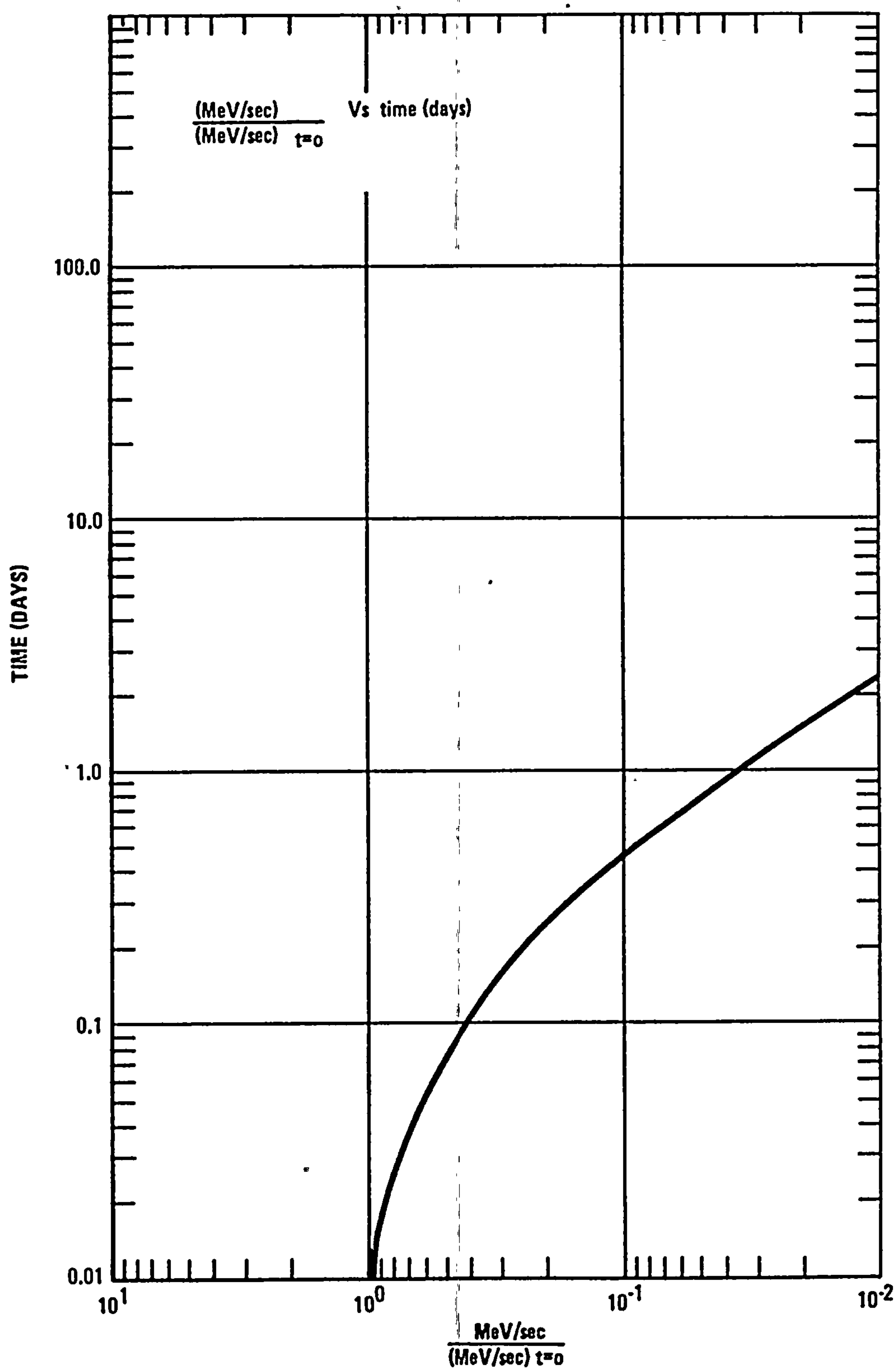
The design review of plant shielding discussed fulfills the NRC requirements outlined in NUREG's 0578 and 0737 as well as Regulatory Guide 1.97, Revision 2.

D. <u>Area Radiation Monitors</u>

A total of 23 intermediate range area radiation monitors per unit are being purchased to monitor a variety of locations in the control, auxiliary, and fuel buildings and the main steam support structure to provide information 1
on actual dose rates post-accident. For further discussion, refer to section II.F.1. These monitors will assist access into areas affected by the radiation from post-accident sources.

E. <u>References</u>

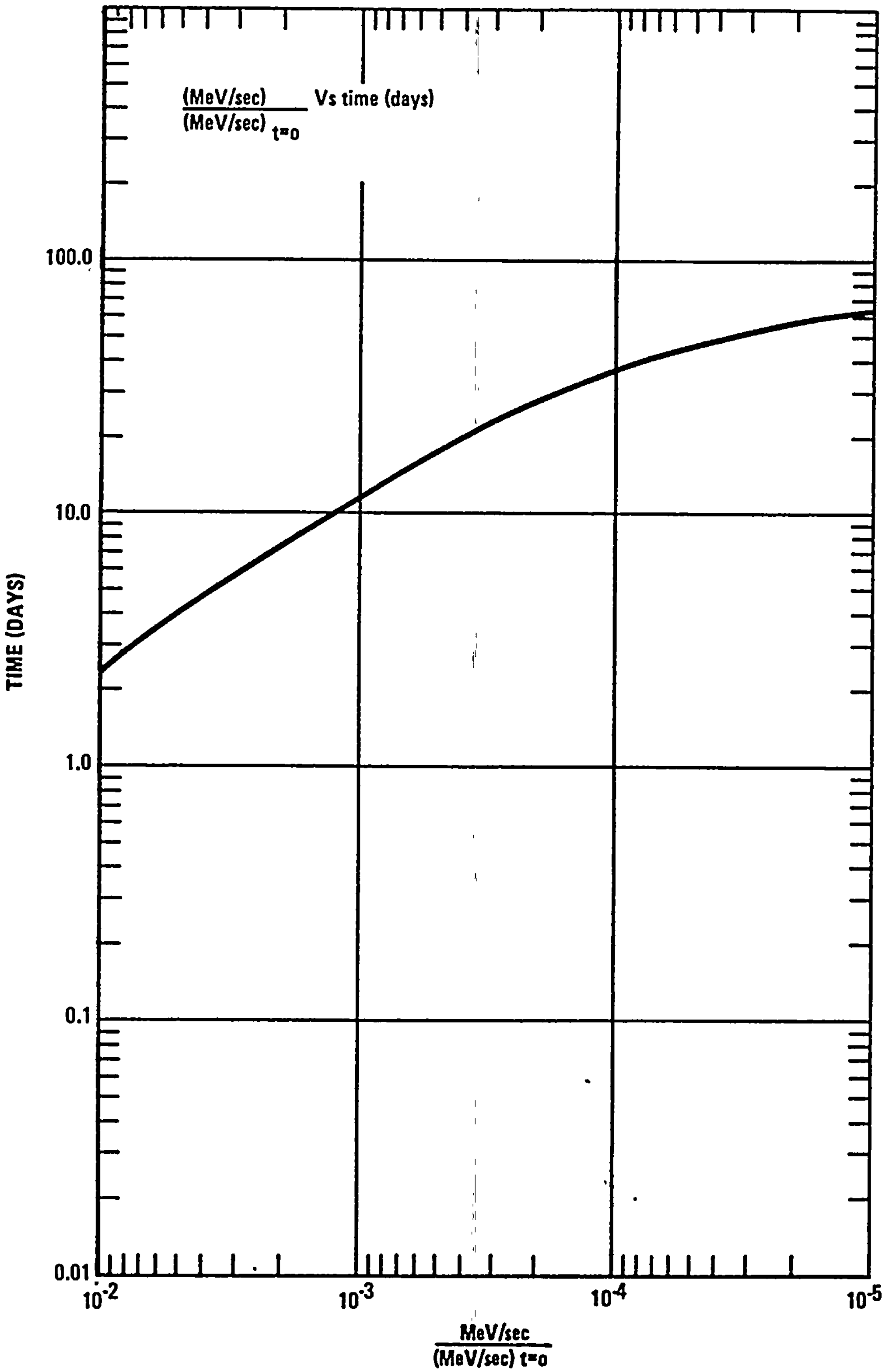
For additional information, refer to FSAR section 11.5 (Radiation Monitoring) and FSAR chapter 12 (Radiation Protection).



Palo Verde Nuclear Generating Station
LLIR

DECAY CURVE - SOURCE A
(Sheet 1 of 2)

Figure II.B.2-1



Palo Verde Nuclear Generating Station
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DECAY CURVE - SOURCE A
(Sheet 2 of 2)

Figure II.B.2-1

[illegible]

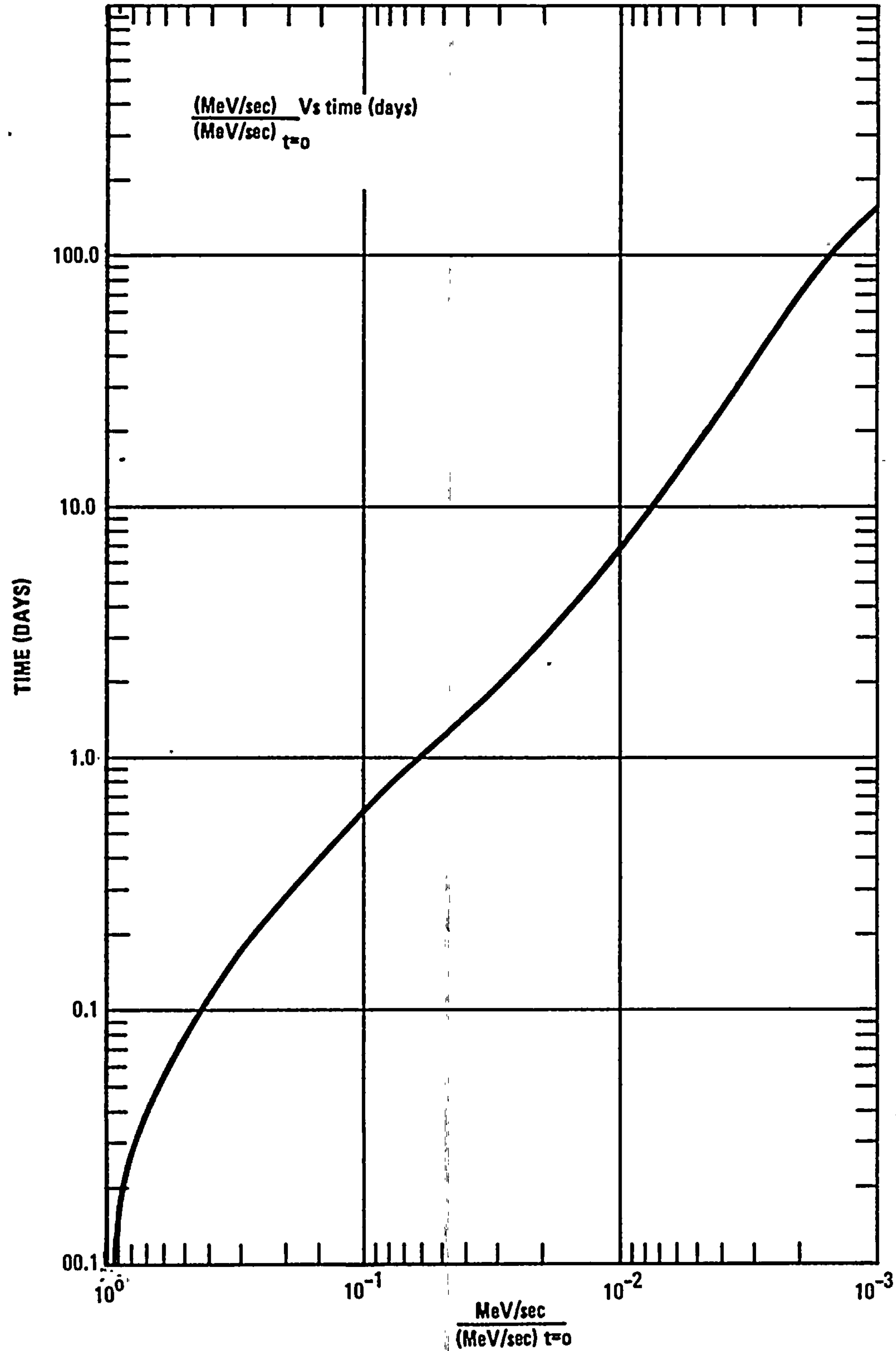
[illegible]

[illegible]

[illegible]

[illegible]

[illegible] [illegible]



Palo Verde Nuclear Generating Station
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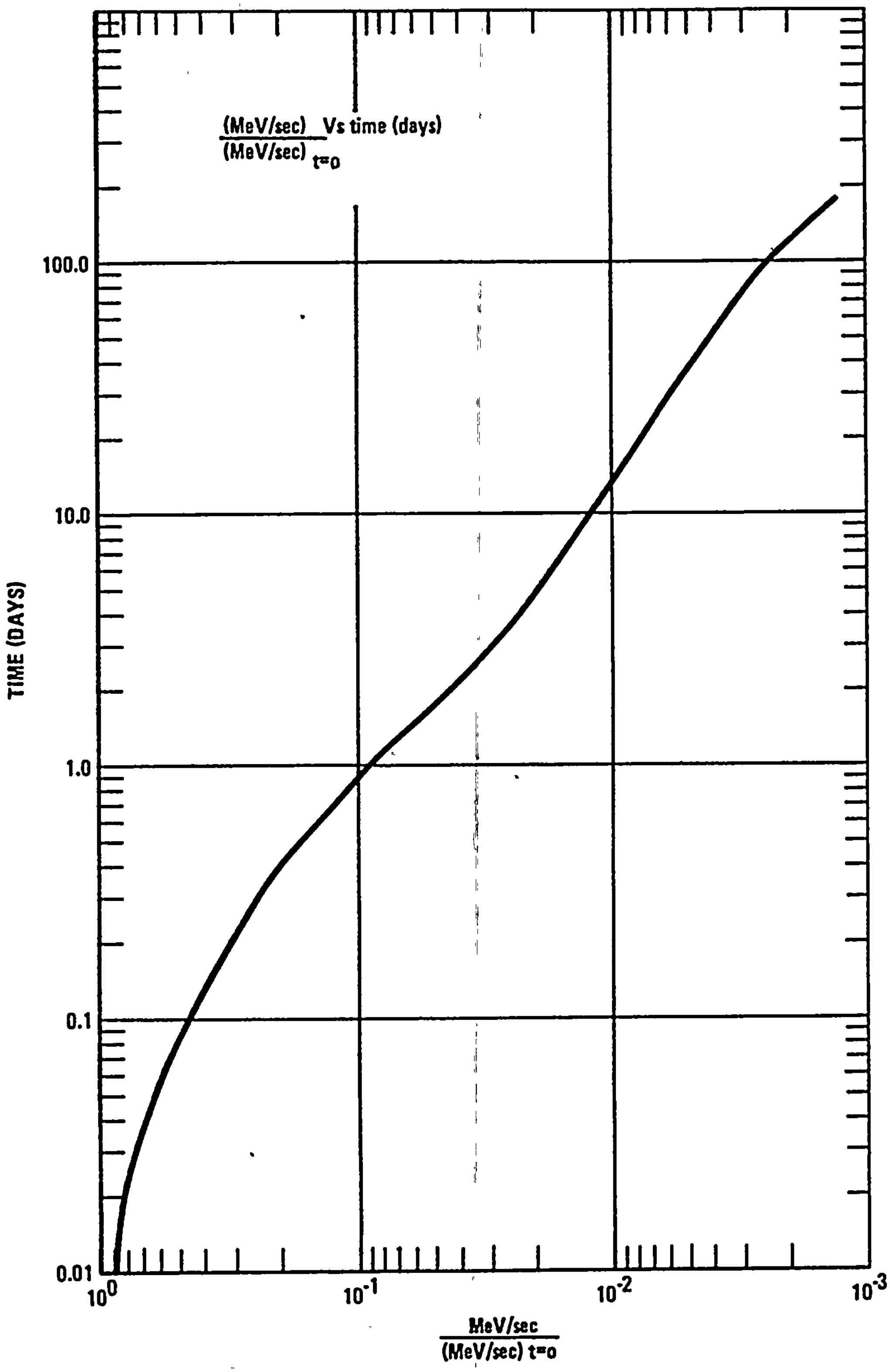
DECAY CURVE - SOURCE B

Figure II.B.2-2

1



1

Palo Verde Nuclear Generating Station
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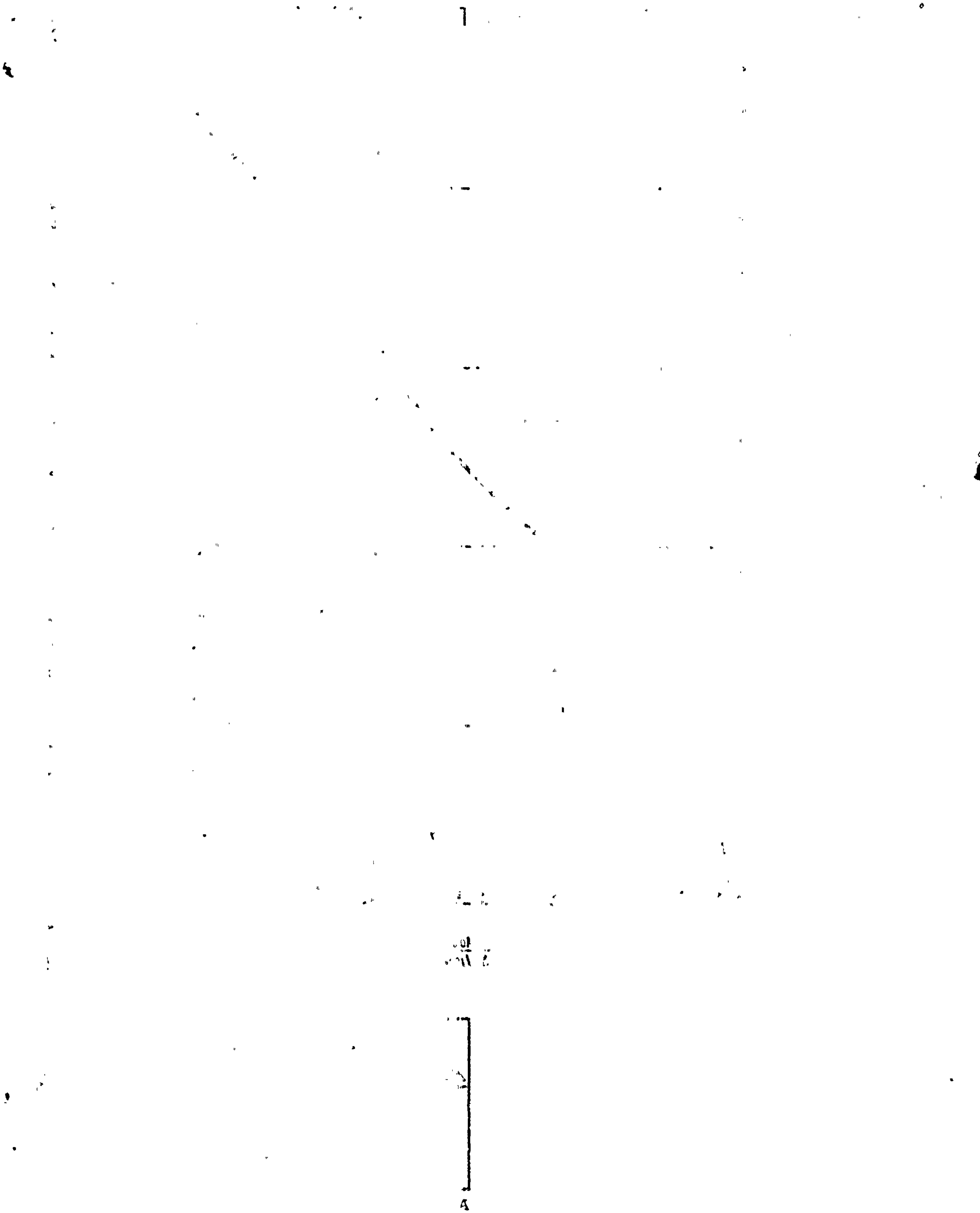
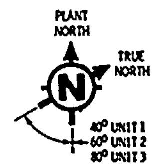
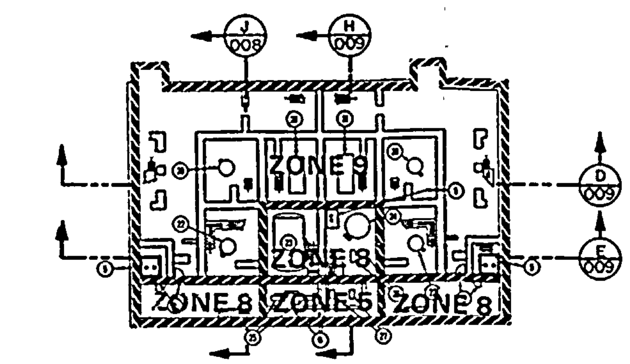
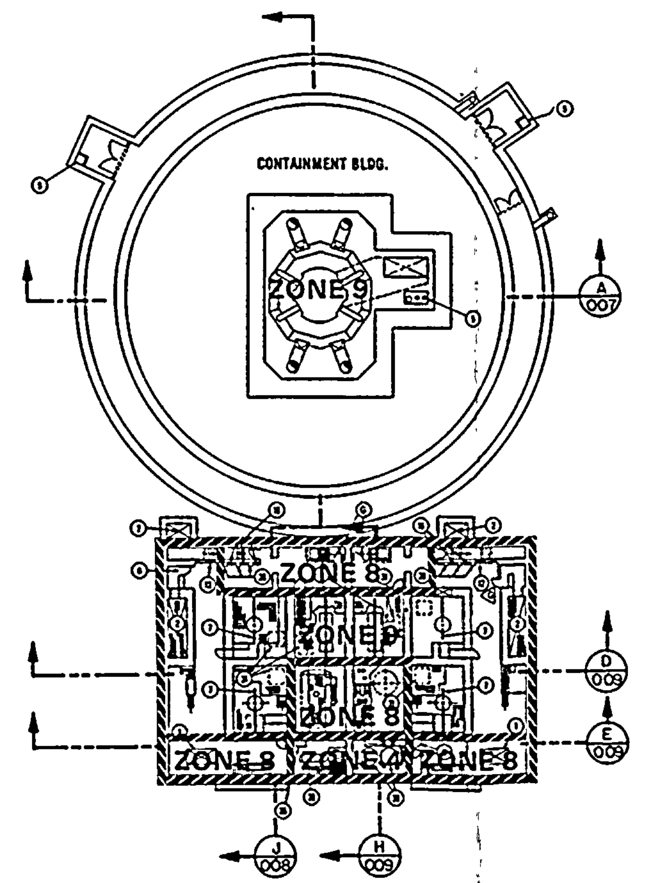
DECAY CURVE - SOURCE C
Figure II.B.2-3



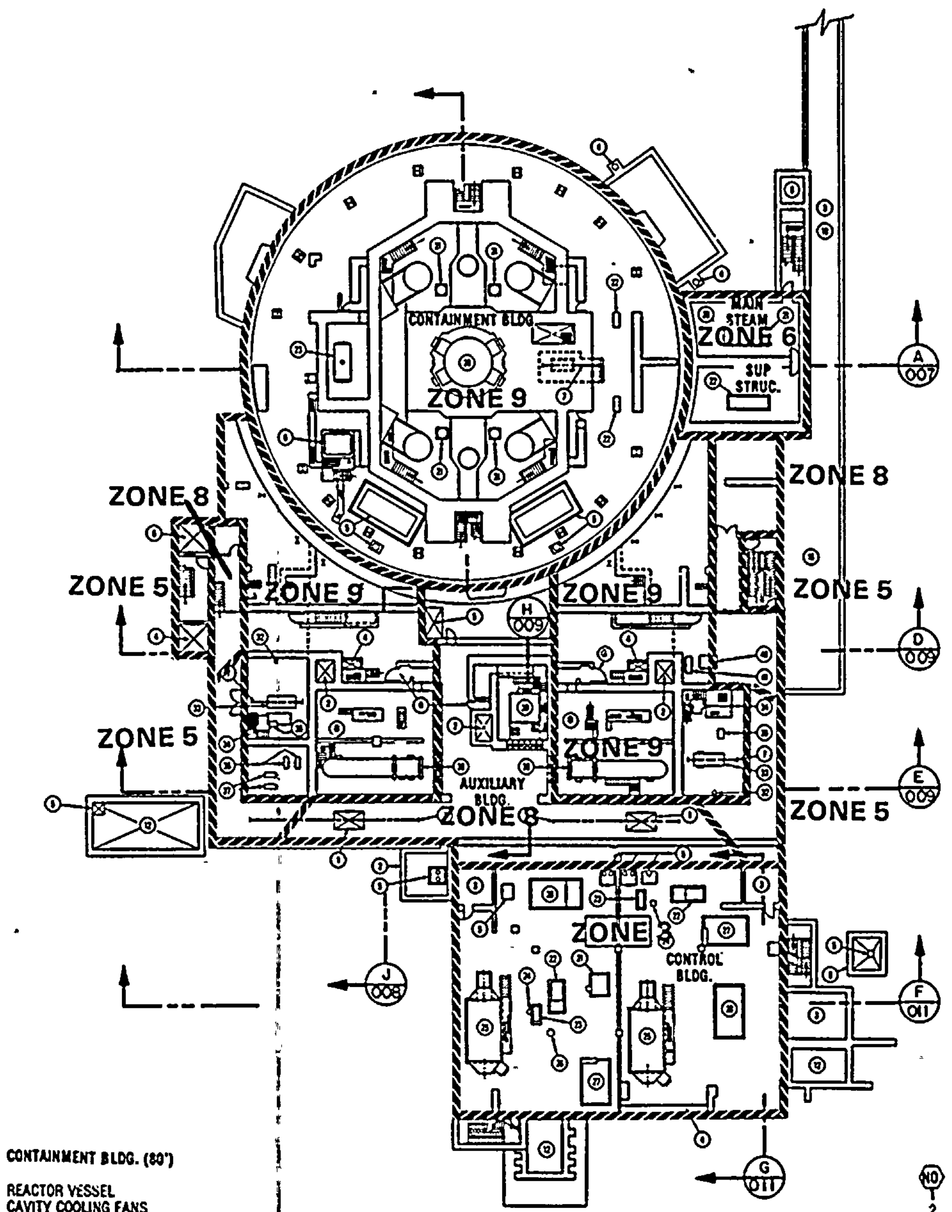




AUXILIARY BLDG.
ELEV. 40 FT. — 0 IN. LEVEL D



AUXILIARY BLDG.
ELEV. 51 FT. — 6 IN. LEVEL C



NOTE: DEPICTED ZONES DO NOT INCORPORATE SOURCE REDUCTIONS DUE TO DECAY OR SINGLE ESF TRAIN OPERATION. REFER TO FSAR SECTION 12.2 FOR DECAY CORRECTIONS. NORMAL OPERATION SOURCES ARE NOT INCLUDED IN THESE ZONES.

CONTAINMENT BLDG. (80')

- 20 REACTOR VESSEL
- 21 CAVITY COOLING FANS
- 22 WET LAY-UP PUMPS
- 23 REACTOR DRAIN TANK

MSSS (81')

- 20 AUX. FEEDWATER TURBINE PACKAGE
- 21 TUBINE DRIVEN PUMP
- 22 MOTOR DRIVEN PUMP

CONTROL BLDG. (74')

- 20 CONTROL BLDG. NORMAL AHU'S
- 21 CONTROL BLDG. OUTSIDE AIR WASHER UNIT
- 22 CONTROL BLDGE. ESF SWGR RM. ESSENTIAL AHU'S
- 23 ESSENTIAL CHILLED WATER CIRCULATION PUMPS
- 24 CHEMICAL ADDITION TANKS
- 25 CONTROL ROOM ESSENTIAL AHU'S
- 26 ESSENTIAL CHILLED WATER EXPANSION TANK
- 27 ESSENTIAL CHILLERS

AUX. BLDG. BETWEEN 40' AND 100'

- 20 CONT. SPRAY PUMPS
- 21 HPSI. PUMPS
- 22 LPSI. PUMPS
- 23 EQUIP. DRAIN TANK
- 24 COOLING WATER HOLD-UP TANK
- 25 HEATER DRAIN PUMPS
- 26 COOLING WATER HOLD-UP TANK PUMPS
- 27 CHEMICAL DRAIN TANK PUMPS
- 28 TRANSFORMERS
- 29 AUX. STEAM CONDENSATE RECEIVER TANK
- 30 CHEMICAL DRAIN TANKS
- 31 ESSENTIAL AIR CONDITIONING UNITS
- 32 ECW CHEMICAL ADDITION TANKS
- 33 ECW PUMPS
- 34 ECW PUMP ROOM ESSENTIAL A.C.U.
- 35 ECW RADIATION MONITORS
- 36 REACTOR MAKE-UP WATER PUMPS
- 37 BORIC ACID MAKE-UP PUMPS
- 38 SHUTDOWN COOLING HEAT EXCHANGERS
- 39 GAS STRIPPER
- 40 GAS STRIPPER CONTROL CABINET
- 41 GAS STRIPPER INSTRUMENT RACK

NO	DESCRIPTION
1	HATCH
2	PIPE CHASE
3	ELEC. CHASE
4	HVAC. CHASE
5	SUMP
6	ELEVATOR
7	MONORAIL
8	VALVE GALLERY
9	MCC
10	LOAD CENTER
11	SWITCHGEAR
12	EQUIP. OR ACCESS OPENING
13	TRANSMITTER RACK
14	VOLTAGE REG.
15	RELAY CAB.
16	LIGHTING PANEL
17	DISTRIBUTION PANEL
18	JUNCTION BOX
19	PIPE TRENCH

LEGEND:

- ZONE 1 $X < 0.5$ mR/hr
- ZONE 2 $0.5 \leq X < 2.5$ mR/hr
- ZONE 3 $2.5 \leq X < 15$ mR/hr
- ZONE 4 $15 \leq X < 100$ mR/hr
- ZONE 5 $100 \leq X < 1000$ mR/hr
- ZONE 6 $1 \leq X < 5$ R/hr
- ZONE 7 $5 \leq X < 10$ R/hr
- ZONE 8 $10 \leq X < 100$ R/hr
- ZONE 9 $100 \leq X$ R/hr

- ← DENOTES UP U.N.O.
- INDICATES ONE-WAY DOOR OR CONTROLLED FROM INSIDE OF BLDG
- G GATE
- ▷ INDICATES CONTROLLED ACCESS AND/OR ROLL UP DOOR

1

Palo Verde Nuclear Generating Station
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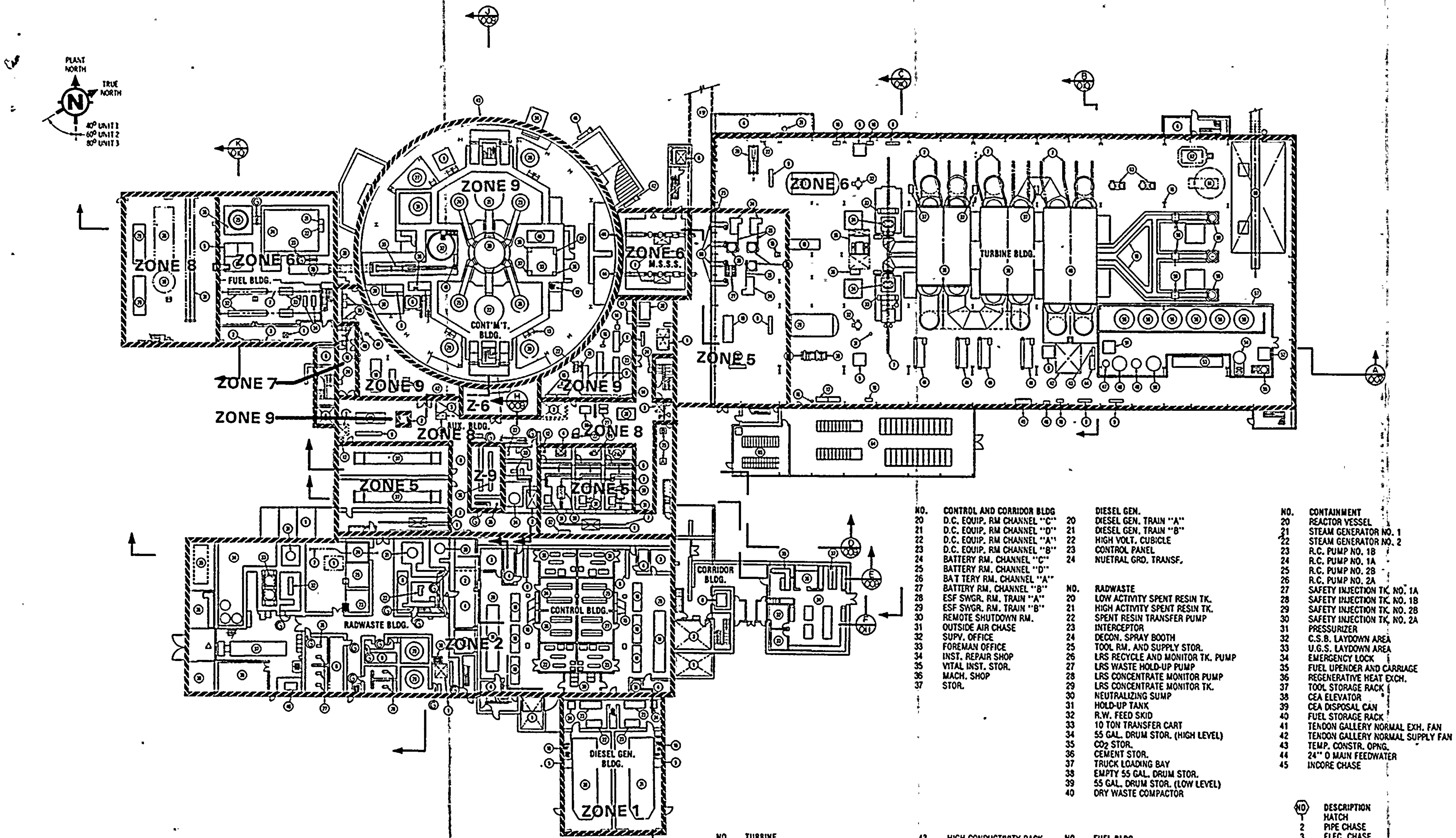
RADIATION ZONES
LOCA WITH SUMP RECIRCULATION
BETWEEN EL. 40'-0" & 100'-0"

Figure II.B.2-4



NOTE. DEPICTED ZONES DO NOT INCORPORATE SOURCE REDUCTIONS DUE TO DECAY OR SINGLE ESF TRAIN OPERATION. REFER TO FSAR SECTION 12.2 FOR DECAY CORRECTIONS. NORMAL OPERATION SOURCES ARE NOT INCLUDED IN THESE ZONES.

NO.	CONTROL AND CORRIDOR BLDG
20	D.C. EQUIP. RM CHANNEL "C"
21	D.C. EQUIP. RM CHANNEL "D"
22	D.C. EQUIP. RM CHANNEL "A"
23	D.C. EQUIP. RM CHANNEL "B"
24	BATTERY RM. CHANNEL "C"
25	BATTERY RM. CHANNEL "D"
26	BAT TERY RM. CHANNEL "A"
27	BATTERY RM. CHANNEL "B"
28	ESF SWGR. RM. TRAIN "A"
29	ESF SWGR. RM. TRAIN "B"
30	REMOTE SHUTDOWN RM.
31	OUTSIDE AIR CHASE
32	SUPV. OFFICE
33	FOREMAN OFFICE
34	INST. REPAIR SHOP
35	VITAL INST. STOR.
36	MACH. SHOP
37	STOR.

	DIESEL GEN.
20	DIESEL GEN. TRAIN "A"
21	DIESEL GEN. TRAIN "B"
22	HIGH VOLT. CUBICLE
23	CONTROL PANEL
24	NUETRAL GRD. TRANSF.

NO.	RADWASTE
20	LOW ACTIVITY SPENT RESIN TK.
21	HIGH ACTIVITY SPENT RESIN TK.
22	SPENT RESIN TRANSFER PUMP
23	INTERCEPTOR
24	DECON. SPRAY BOOTH
25	TOOL RM. AND SUPPLY STOR.
26	LRS RECYCLE AND MONITOR TK. PUMP
27	LRS WASTE HOLD-UP PUMP
28	LRS CONCENTRATE MONITOR PUMP
29	LRS CONCENTRATE MONITOR TK.
30	NEUTRALIZING SUMP
31	HOLD-UP TANK
32	R.W. FEED SKID
33	10 TON TRANSFER CART
34	55 GAL. DRUM STOR. (HIGH LEVEL)
35	CO_2 STOR.
36	CEMENT STOR.
37	TRUCK LOADING BAY
38	EMPTY 55 GAL. DRUM STOR.
39	55 GAL. DRUM STOR. (LOW LEVEL)
40	DRY WASTE COMPACTOR

NO.	CONTAINMENT
20	REACTOR VESSEL
21	STEAM GENERATOR NO. 1
22	STEAM GENERATOR NO. 2
23	R.C. PUMP NO. 1B
24	R.C. PUMP NO. 1A
25	R.C. PUMP NO. 2B
26	R.C. PUMP NO. 2A
27	SAFETY INJECTION TK. NO. 1A
28	SAFETY INJECTION TK. NO. 1B
29	SAFETY INJECTION TK. NO. 2B
30	SAFETY INJECTION TK. NO. 2A
31	PRESSURIZER
32	C.S.B. LAYDOWN AREA
33	U.G.S. LAYDOWN AREA
34	EMERGENCY LOCK
35	FUEL UPENDER AND CARRIAGE
36	REGENERATIVE HEAT EXCH.
37	TOOL STORAGE RACK
38	CEA ELEVATOR
39	CEA DISPOSAL CAN
40	FUEL STORAGE RACK
41	TENDON GALLERY NORMAL EXH. FAN
42	TENDON GALLERY NORMAL SUPPLY FAN
43	TEMP. CONSTR. OPNG.
44	24" O MAIN FEEDWATER
45	INCORE CHASE

NO.	AUXILIARY
20	H_2 RECOMBINER
21	ESSENTIAL ACU
22	RAD. MONITOR
23	ELEC. PENET. RM TRAIN "B"
24	CONCENTRATOR PANEL
25	H_2 RECOMBINER CONTROL PANEL
26	H_2 COMPRESSOR
27	H_2 PURGE EXH.
28	POST LOCA ANALYZER
29	CVCS CHEM ADDIT.
30	CHARGING PUMPS
31	AIR HANDLING UNITS
32	TRANSFER SWITCH
33	CRUD PUMP
34	CRUD TANK
35	SEAL INJ. HEAT EXCH.
36	LETDOWN HEAT EXCH.
37	ECW HEAT EXCH.
38	SCR POWER CONTROLLER
39	HEATER PANELS
40	LIGHTING TRANSF
41	PENET. ALARM
42	ELEC. PENET. RM. TRAIN "C"

NO.	TURBINE
20	AUX. F.W. PUMP
21	BLOWDOWN STACK
22	HVAC ANNUC. BD.
23	SEQUENCE CONT. PNL.
24	SERV. AND INST. AIR COMP.
25	AFTERCOOLER
26	AIR REC.
27	AIR DRYER
28	HTR. DRN. TK.
29	AMMONIA ADDIT. PUMP
30	HYDRAZINE ADDIT. PUMP
31	STM. PKG. EXH. DRN. TK.
32	HTR. DRN. PUMP
33	F.W. PUMP TURB.
34	F.W. PUMP TURB.
35	EHC POWER UNIT
36	F.W. PUMP TURB L.O. CENT.
37	HOTWELL COND. ANALYZERS
38	CONDENSER "A"
39	CONDENSER "B"
40	CONDENSER "C"
41	VACUUM PUMPS
42	FUTURE SPACE (COND. TUBE CLNG. EQUIP.)
43	HIGH CONDUCTIVITY RACK
44	HIGH SODIUM ANALYZER
45	POWER INST. PNL.
46	VOLT. REG.
47	WASTE COLL. TK.
48	CATION TK
49	ANION TK
50	MIX AND HOLD TK
51	LOW TDS SUMP
52	HIGH TDS SUMP
53	CONTROL PNL.
54	CAUSTIC PNL.
55	ACID SKID
56	COND. POLISH. DEMIN. TK
57	FUTURE WALLS
58	COND. PUMP STRAINER
59	COND. PUMP STRAINER
60	R.R. RAILS
61	PORTA-BOR
62	PORTA-BOR CONTROL CENTER
63	TURB. COOLING WTR. PUMP
64	SWITCHGEAR RM.
65	BATTERY RM
66	MAIN STM LINES

NO.	FUEL BLDG
20	TRANSFER TUBE
21	FAILED FUEL CONTAINER
22	STOR. RACKS (IN-CODE INST. SURV. SAMPLES)
23	TOOL STOR.
24	SPENT FUEL STOR. RACKS
25	CASK
26	CASK LOADING PIT
27	STORAGE RM
28	CASK PERSONNEL BARRIER LAYDOWN
29	EXH. ESSENTIAL AIR FILTRATION UNIT
30	UPPER AND LOWER CRUSH CONE LAYDOWN
31	R.R. RAILS
32	SPENT FUEL POOL HEAT EXCH.
33	SPENT FUEL COOLING PUMP
34	POOL CLEANUP PUMP
35	NEW FUEL ELEV.

NO	DESCRIPTION
1	HATCH
2	PIPE CHASE
3	ELEC. CHASE
4	HVAC. CHASE
5	SUMP
6	ELEVATOR
7	MONORAIL
8	VALVE GALLERY
9	MCC
10	LOAD CENTER
11	SWITCHGEAR
12	EQUIP. OR ACCESS OPENING
13	TRANSMITTER RACK
14	VOLTAGE REG.
15	RELAY CAB.
16	LIGHTING PANEL
17	DISTRIBUTION PANEL
18	JUNCTION BOX
19	PIPE TRENCH

LEGEND:

ZONE 1 $X < 0.5$ mR/hr
ZONE 2 $0.5 \leq X < 2.5$ mR/hr
ZONE 3 $2.5 \leq X < 15$ mR/hr
ZONE 4 $15 \leq X < 100$ mR/hr
ZONE 5 $100 \leq X < 1000$ mR/hr
ZONE 6 $1 \leq X < 5$ R/hr
ZONE 7 $5 \leq X < 10$ R/hr
ZONE 8 $10 \leq X < 100$ R/hr
ZONE 9 $100 \leq X$ R/hr

DENOTES UP U.N.O.
INDICATES ONE-WAY DOOR OR CONTROLLED FROM INSIDE OF BLDG
GATE
INDICATES CONT. ACCESS &OR ROLL DRS

Palo Verde Nuclear Generating Station
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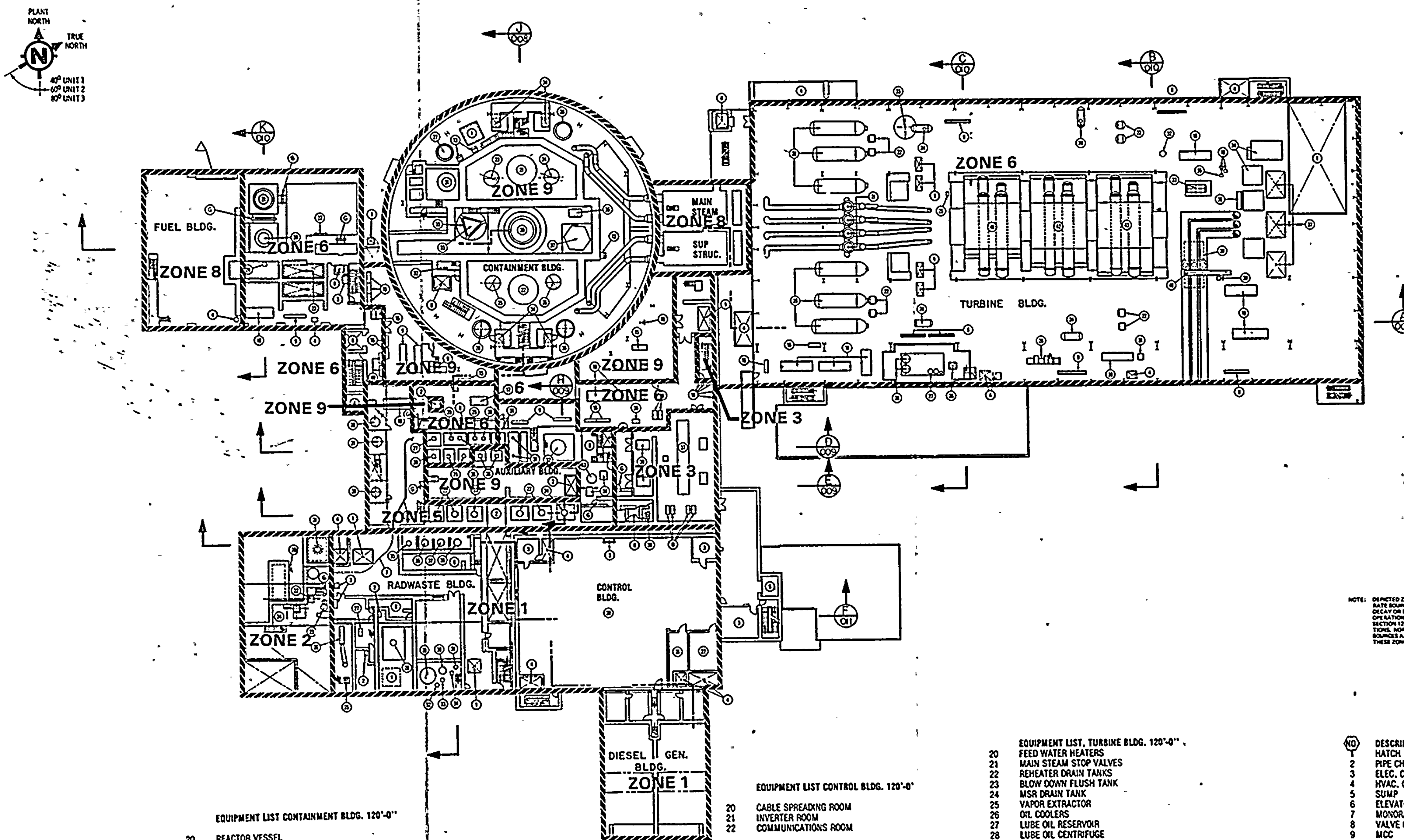
RADIATION ZONES
LOCA WITH SUMP RECIRCULATION
AT EL. 100'-0"

Figure II.B.2-5

August 1981 Amendment 1



NOTE: DEPICTED ZONES DO NOT INCORPORATE SOURCE REDUCTIONS DUE TO DECAY OR SINGLE ESF TRAIN OPERATION. REFER TO FSAR SECTION 12.2 FOR DECAY CORRECTIONS. NORMAL OPERATION SOURCES ARE NOT INCLUDED IN THESE ZONES.

1

EQUIPMENT LIST CONTAINMENT BLDG. 120'-0"

20 REACTOR VESSEL
21 STEAM GENERATOR NO. 1
22 STEAM GENERATOR NO. 2
23 REACTOR COOLANT PUMP 1A
24 REACTOR COOLANT PUMP 1B
25 REACTOR COOLANT PUMP 2A
26 REACTOR COOLANT PUMP 2B
27 SAFETY INJECTION TANK 1A
28 SAFETY INJECTION TANK 1B
29 SAFETY INJECTION TANK 2A
30 SAFETY INJECTION TANK 2B
31 PRESSURIZER
32 REGENERATIVE HTx
33 CORE SUPPORT BARREL AREA
34 NORMAL ACC
35 C.S.B LIFT RIG
36 SEAL TABLE
37 UPPER GUIDE STRUCT. LIFT RIG
38
39
40

EQUIPMENT LIST FOR RADWASTE BLDG ELEV. 120

20 TV MONTOR
21
22
23
24
25 EVAPORATOR CONCENTR PUMPS
26 EVAPORATOR MAIN RECYCLE PUMP
27 EVAPORATOR STEAM CONDENSATE PUMP
28 BORIC ACID CONCENTRATOR
29 CAUSTIC TANK
30 ACID TANK
31 ANTI-FOAM TANK
32 CAUSTIC BATCH TANK
33 ACID BATCH TYPE
34 ANTI-FOAM PUMP
35 BORIC ACID CON. INOx
36 L.R.S. MIXED BED INOx
37 L.R.S. MIXED BED IONx
38 L.R.S. ADSORPTION BED

EQUIPMENT LIST CONTROL BLDG. 120'-0'

20 CABLE SPREADING ROOM
21 INVERTER ROOM
22 COMMUNICATIONS ROOM

EQUIPMENT LIST AND AUXILIARY BLDG. ELEV. 120'-0"

20 ECW SURGE TANK
21 BORIC ACID BATCH TANK
22 FUEL POOL PURIFICATION ION EXCH.
23 PRE HOLD UP ION EXCH.
24 DEBORATING ION EXCH.
25 SAFETY INJECTION FILTERS
26 PURIFICATION FILTERS
27 BORIC ACID FILER
28 REACTOR DRAIN FILTER
29 LRS FILTERS
30 FUEL POOL PURIFICATION FILTERS
31 CRUD TANK FILTERS
32 VOLUME CONTROL TANK
33 SPRAY CHEM STORAGE
34 SPRAY CHEM ADD. PUMPS
35 EXHAUST RAD MONITOR
36 AUX CABINETS
37 CEDM CONTROL CAB. S
38 PROCESS RAD MONITOR
39 MG SET
40 HEAT TRACE CAB.

EQUIPMENT LIST, TURBINE BLDG. 120'-0"

20 FEED WATER HEATERS
21 MAIN STEAM STOP VALVES
22 REHEATER DRAIN TANKS
23 BLOW DOWN FLUSH TANK
24 MSR DRAIN TANK
25 VAPOR EXTRACTOR
26 OIL COOLERS
27 LUBE OIL RESERVOIR
28 LUBE OIL CENTRIFUGE
29 STEAM SEAL PKG.
30 MAIN CONDENSER EVAC. SYS. FILTER PKG.
31 POST FILTER BLOWER
32 SURGE TANK
33 H_2 SEAL OIL UNIT
34 GENERATOR STATOR COOLING UNIT
35 DC. MTR. STR.
36 H_2 AND STATOR COOLING CONTROL
37 CONDENSATE PUMPS AND REMOVE
38 H_2 DETECTION CABINET
39 ISO PHASE BUS COOLING
40 AUX DETRAINING TANK
41 CONDENSER "A"
42 CONDENSER "B"
43 CONDENSER "C"

EQUIPMENT LIST FUEL BLDG 120'-0"

20 CASK HEAD LAYDOWN
21 CASK
22 TOOL STORAGE
23 NEW FUEL STORAGE RACKS
24 HANDLING SLEEVE

NO	DESCRIPTION
1	HATCH
2	PIPE CHASE
3	ELEC. CHASE
4	HVAC. CHASE
5	SUMP
6	ELEVATOR
7	MONORAIL
8	VALVE GALLERY
9	MCC
10	LOAD CENTER
11	SWITCHGEAR
12	EQUIP. OR ACCESS OPENING
13	TRANSMITTER RACK
14	VOLTAGE REG.
15	RELAY CAB.
16	LIGHTING PANEL
17	DISTRIBUTION PANEL
18	JUNCTION BOX
19	PIPE TRENCH

LEGEND:

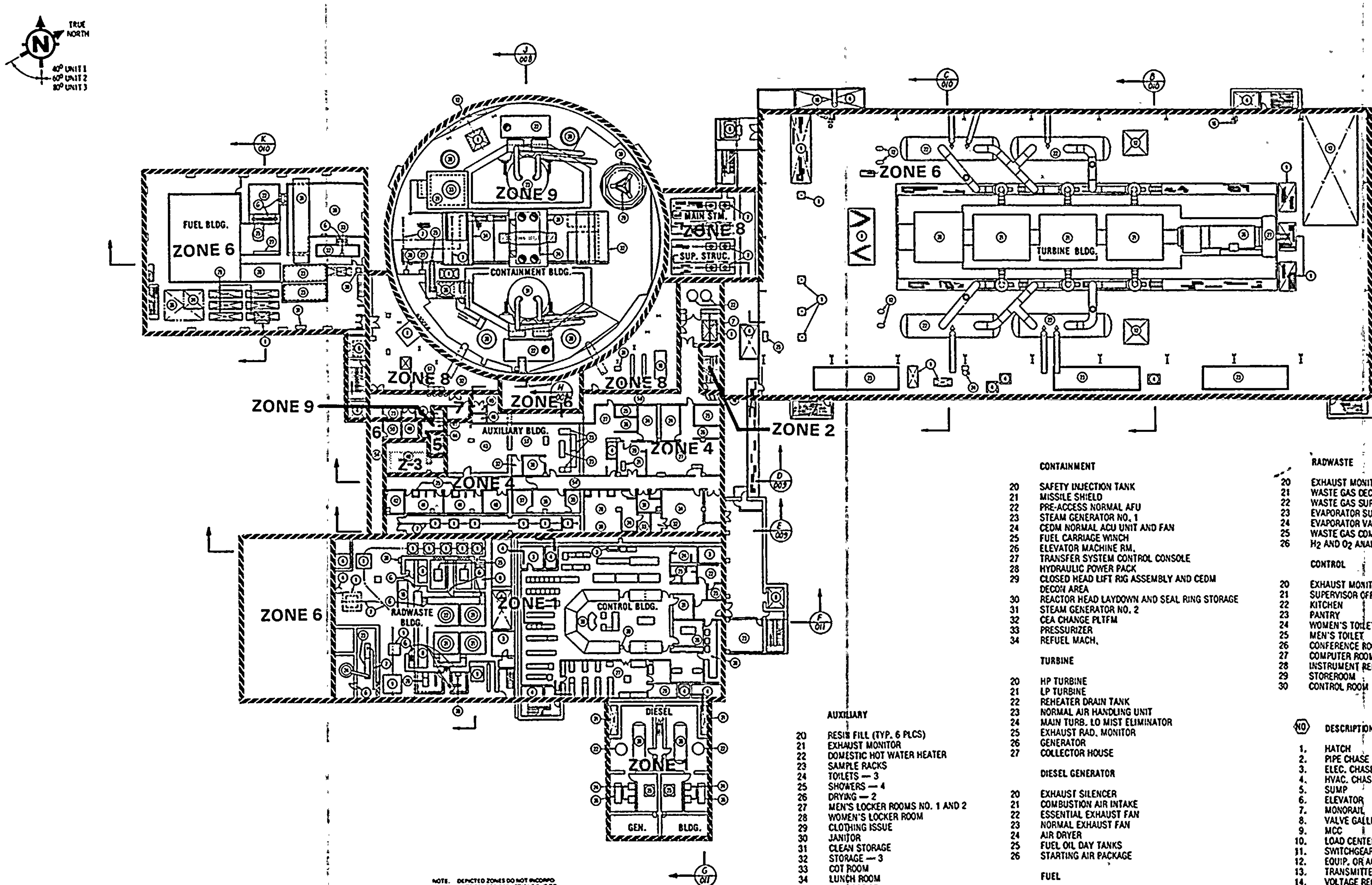
ZONE 1 $X < 0.5$ mR/hr
ZONE 2 $0.5 \leqslant X < 2.5$ mR/hr
ZONE 3 $2.5 \leqslant X < 15$ mR/hr
ZONE 4 $15 \leqslant X < 100$ mR/hr
ZONE 5 $100 \leqslant X < 1000$ mR/hr
ZONE 6 $1 \leqslant X < 5$ R/hr
ZONE 7 $5 \leqslant X < 10$ R/hr
ZONE 8 $10 \leqslant X < 100$ R/hr
ZONE 9 $100 \leqslant X$ R/hr

→ DENOTES UP U.N.O.
INDICATES ONE-WAY DOOR OR CONTROLLED FROM INSIDE OF. BLDG
Ⓖ G - GATE
△ INDICATES CONTROLLED ACCESS AND/OR ROLL-UP DOOR

Palo Verde Nuclear Generating Station
LLIR

RADIATION ZONES
LOCA WITH SUMP RECIRCULATION
BETWEEN EL. 120'-0" & 140'-0"

Figure II.B.2-6



NOTE. DEPICTED ZONES DO NOT INCORPORATE SOURCE REDUCTIONS DUE TO DECAY OR SINGLE ESF TRAIN OPERATION. REFER TO FSAR SECTION 12.2 FOR DECAY CORRECTIONS. NORMAL OPERATION SOURCES ARE NOT INCLUDED IN THESE ZONES.

CHANGE

AUXILIARY

20 RESIN FILL (TYP. 6 PLCS)
21 EXHAUST MONITOR
22 DOMESTIC HOT WATER HEATER
23 SAMPLE RACKS
24 TOILETS — 3
25 SHOWERS — 4
26 DRYING — 2
27 MEN'S LOCKER ROOMS NO. 1 AND 2
28 WOMEN'S LOCKER ROOM
29 CLOTHING ISSUE
30 JANITOR
31 CLEAN STORAGE
32 STORAGE — 3
33 COT ROOM
34 LUNCH ROOM
35 H. P. OFFICE
36 HEALTH PHYSICS
37 INST. RESP. MAINT. ISSUE
38 OFFICE
39 SAMPLING AREA
40 FIRST AID
41 PERSONNEL DECON AREA — 2
42 H. P. SAMPLE COUNT. ROOM
43 HOT LAB
44 SAMPLE ROOM HOOD AND SINK
45 FLAMMABLE STORAGE
46 CYLINDER STORAGE
47 PIPING AREA
48 COUNTING ROOM
49 TOOL AND EQUIP. ROOM
50 SPECIAL DECON.
51 DECON. ROOM
52 EMERGENCY SHOWER AND EYE WASH
53 COLD LAB
54 CORRIDOR

CONTAINMENT

20 SAFETY INJECTION TANK
21 MISSILE SHIELD
22 PRE-ACCESS NORMAL AFU
23 STEAM GENERATOR NO. 1
24 CEDM NORMAL ACU UNIT AND FAN
25 FUEL CARRIAGE WINCH
26 ELEVATOR MACHINE RM.
27 TRANSFER SYSTEM CONTROL CONSOLE
28 HYDRAULIC POWER PACK
29 CLOSED HEAD LIFT RIG ASSEMBLY AND CEDM DECON AREA
30 REACTOR HEAD LAYDOWN AND SEAL RING STORAGE
31 STEAM GENERATOR NO. 2
32 CEA CHANGE PLTFM
33 PRESSURIZER
34 REFUEL MACH.

TURBINE

20 HP TURBINE
21 LP TURBINE
22 REHEATER DRAIN TANK
23 NORMAL AIR HANDLING UNIT
24 MAIN TURB. LO MIST ELIMINATOR
25 EXHAUST RAD. MONITOR
26 GENERATOR
27 COLLECTOR HOUSE

DIESEL GENERATOR

20 EXHAUST SILENCER
21 COMBUSTION AIR INTAKE
22 ESSENTIAL EXHAUST FAN
23 NORMAL EXHAUST FAN
24 AIR DRYER
25 FUEL OIL DAY TANKS
26 STARTING AIR PACKAGE

FUEL

20 SPENT FUEL POOL
21 SPENT FUEL HANDLING MACHINE
22 CASK LOADING PIT
23 NEW FUEL STORAGE COVERS
24 CASK LIFTING RIG LAYDOWN AREA
25 DECON PIT
26 DECON PIT COVER
27 GATE STORAGE
28 NEW FUEL INSPECTION PIT
29 NEW FUEL CONTAINERS LAYDOWN AREA
30 UPENDER HYD. DRIVE UNIT
31 POOL COOLING PURIFICATION PANEL
32 CANAL
33 NEW FUEL ELEVATOR

RADWASTE

20 EXHAUST MONITOR
21 WASTE GAS DECAY TANK
22 WASTE GAS SURGE TANK
23 EVAPORATOR SURFACE CONDENSER
24 EVAPORATOR VAPOR BODY
25 WASTE GAS COMPRESSOR
26 H_2 AND O_2 ANALYZER

CONTROL

20 EXHAUST MONITOR
21 SUPERVISOR OFFICE
22 KITCHEN
23 PANTRY
24 WOMEN'S TOILET
25 MEN'S TOILET
26 CONFERENCE ROOM
27 COMPUTER ROOM
28 INSTRUMENT REPAIR ROOM
29 STOREROOM
30 CONTROL ROOM

NO DESCRIPTION

1. HATCH
2. PIPE CHASE
3. ELEC. CHASE
4. HVAC. CHASE
5. SUMP
6. ELEVATOR
7. MONORAIL
8. VALVE GALLERY
9. MCC
10. LOAD CENTER
11. SWITCHGEAR
12. EQUIP. OR ACCESS OPENING
13. TRANSMITTER RACK
14. VOLTAGE REG.
15. RELAY CAB.
16. LIGHTING PANEL
17. DISTRIBUTION PANEL
18. JUNCTION BOX
19. PIPE TRENCH

1

LEGEND:

ZONE 1 $X < 0.5$ mR/hr
ZONE 2 $0.5 \leqslant X < 2.5$ mR/hr
ZONE 3 $2.5 \leqslant X < 15$ mR/hr
ZONE 4 $15 \leqslant X < 100$ mR/hr
ZONE 5 $100 \leqslant X < 1000$ mR/hr
ZONE 6 $1 \leqslant X < 5$ R/hr
ZONE 7 $5 \leqslant X < 10$ R/hr
ZONE 8 $10 \leqslant X < 100$ R/hr
ZONE 9 $100 \leqslant X$ R/hr

→ DENOTES UP U.N.O.
INDICATES ONE-WAY DOOR OR CONTROLLED FROM INSIDE OF BLDG
G-GATE
△ INDICATES CONTROLLED ACCESS AND/OR ROLL UP DOOR

Palo Verde Nuclear Generating Station
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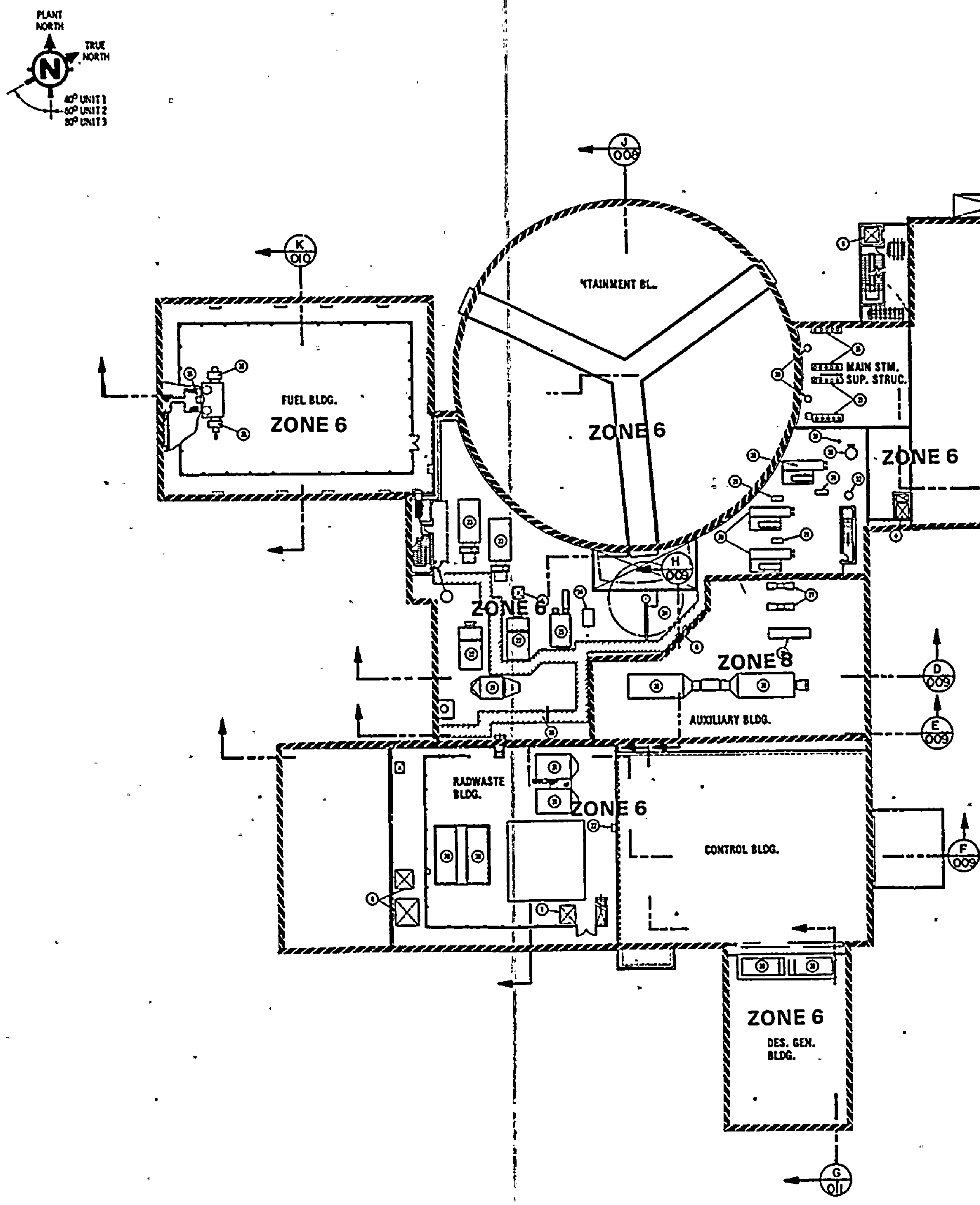
RADIATION ZONES
LOCA WITH SUMP RECIRCULATION
BETWEEN EL. 140'-0" & 200'-0"

Figure II.B.2-7



RADWASTE BLDG

20 RADWASTE BLDG. NORMAL SUPPLY UNIT
21 RADWASTE BLDG. NORMAL EXHAUST AFU
22 RADWASTE BLDG. EXHAUST MONITOR

MAIN STM SUPP. STRUC.

20 ATMOSPHERIC DUMP VENTS
21 MAIN STEAM RELIEF VENTS

DIESEL BLDG

20 DIESEL GENERATOR EXHAUST STACK

TURBINE BLDG.

20 TURBINE BLDG. NORMAL EXHAUST FANS

FUEL BLDG

20 NORMAL EXHAUST FAN UNIT
21 FUEL BLDG EXHAUST MONITOR
22. HVAC EXHAUST RISER

AUXILIARY BLDG

20 AUX. BLDG EXHAUST NORMAL AFU
21 ACCESS CONTROL AREA AHU
22 AUX. BLDG. SUPPLY AHU
23 FUEL BLDG. SUPPLY AHU
24 CONTROL BLDG PRE-ACCESS AHU
25 CONTROL BLDG REFUELING PURGE AHU
26 CONTROL PURGE EXHAUST NORMAL AFU
27 CONTROL BLDG. EXHAUST FANS
28 NORMAL CHILLER
29 NORMAL CHILLED WATER RECIRC. PUMP
30 NUCLEAR COOLING WATER SURGE TANK
31 NORMAL CHILLED WATER CHEM. ADDITION TANK
32 NORMAL CHILLED WATER EXPANSION TANK
33 FUME HOOD EXHAUST FANS
34 5 TON JIB CRANE
35 ACCESS AREA

NO	DESCRIPTION
1	HATCH
2	PIPE CHASE
3	ELEC. CHASE
4	HVAC. CHASE
5	SUMP
6	ELEVATOR
7	MONORAIL
8	VALVE GALLERY
9	MCC
10	LOAD CENTER
11	SWITCHGEAR
12	EQUIP. OR ACCESS OPENING
13	TRANSMITTER RACK
14	VOLTAGE REG.
15	RELAY CAB.
16	LIGHTING PANEL
17	DISTRIBUTION PANEL
18	JUNCTION BOX
19	PIPE TRENCH

NOTE: DEPICTED ZONES DO NOT INCORPORATE SOURCE REDUCTIONS DUE TO DECAY OR SINGLE ESF TRAIN OPERATION. REFER TO FSAR SECTION 12.2 FOR DECAY CORRECTIONS. NORMAL OPERATION SOURCES ARE NOT INCLUDED IN THESE ZONES.

DENOTES UP U.N.O.
INDICATES ONE-WAY DOOR OR CONTROLLED FROM INSIDE OF BLDG
GATE
INDICATES CONTROLLED ACCESS AND/OR ROLL UP DOOR

LEGEND:
ZONE 1 $X < 0.5$ mR/hr
ZONE 2 $0.5 \leqslant X < 2.5$ mR/hr
ZONE 3 $2.5 \leqslant X < 15$ mR/hr
ZONE 4 $15 \leqslant X < 100$ mR/hr
ZONE 5 $100 \leqslant X < 1000$ mR/hr
ZONE 6 $1 \leqslant X < 5$ R/hr
ZONE 7 $5 \leqslant X < 10$ R/hr
ZONE 8 $10 \leqslant X < 100$ R/hr
ZONE 9 $100 \leqslant X$ R/hr

Palo Verde Nuclear Generating Station
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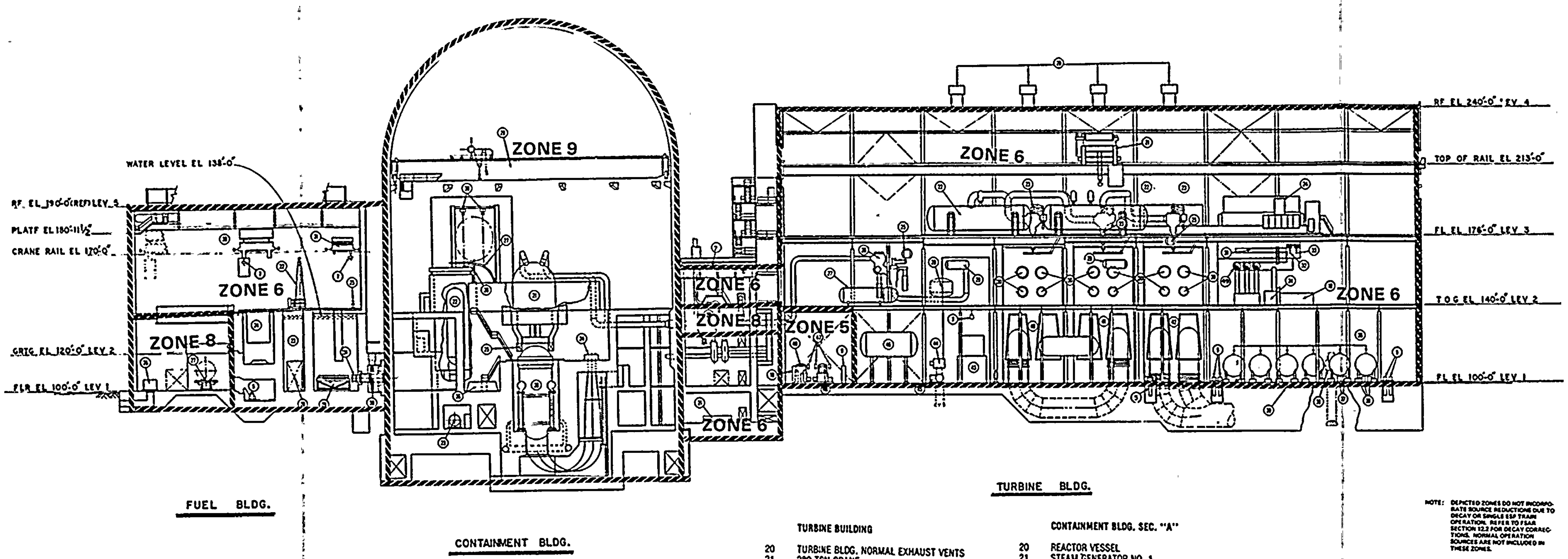
RADIATION ZONES
LOCA WITH SUMP RECIRCULATION
AT ROOF ELEVATION

Figure II.B.2-8

August 1981 Amendment 1



SECTION A
002
003
004
005
006

TURBINE BUILDING

20 TURBINE BLDG. NORMAL EXHAUST VENTS
21 280 TON CRANE
22 MSR MOISTURE SEPARATOR REHEATER
23 CONTROL INSULATION VALVES
24 GENERATOR
25 REHEATOR DRAIN TANK
26 MAIN STEAM STOP AND CONTROL VALVE
27 FEEDWATER HEATER
28 BLOWDOWN FLUSH TANK
29 MSR DRAIN TANK
30 L.P. FEEDWATER HEATERS
31 AUX. DETRAINING TANK
32 LINE TERMINAL
33 NEUTRAL TERMINAL
34 P.T. CUBICLE
35 CNDS POLISHING TANK
36 CONDENSATE PUMP
37 HYDRAZINE PUMPS
38 AMMONIA PUMPS
39 CNDS PUMP STRAINER
40 CONDENSER "A"
41 CONDENSER "B"
42 CONDENSER "C"
43 FEEDWATER PUMP TURBINE
44 HEATER DRAIN PUMP
45 FEEDWATER TUB. LUBE OIL CENTRIFUGE
46 HEATER DRAIN TANK
47 AFTER COOLER AND MOIST. SEPEX.
48 AIR RECEIVER
49 ISOPHASE BUS. DUCT
50

CONTAINMENT BLDG. SEC. "A"

20 REACTOR VESSEL
21 STEAM GENERATOR NO. 1
22 PRESSURIZER
23 REACTOR DRAIN TANK
24 SEAL TABLE
25 R.V. HEAD COOLING SHROUD
26 CORE SUPPORT BARREL STAND
27 EQUIPMENT HATCH
28 JIB CRANE
29 POLAR CRANE
30 ELECTRIC HOIST

MAIN STEAM SUPPORT STRUCTURE

20 MAIN STEAM ISOLATION VALVE
21 AUX. FEED TURBINE PUMP

FUEL BUILDING

20 150 TON CASK HANDLING CRANE
21 10 TON FUEL HANDLING CRANE
22 SPENT FUEL HANDLING MACHINE
23 SPENT FUEL POOL
24 DECON. PIT
25 TRANSFER TUBE VALVE HANDWHEEL
26 EXHAUST A.F.U.
27 RAILROAD CARS
28 SPENT FUEL RACKS
29 FUEL CARRIAGE AND UPENDER
30 FUEL TRANSFER TUBE
31 TRANSFER TUBE VALVE
32

NOTE: DEPICTED ZONES DO NOT INCORPORATE SOURCE REDUCTIONS DUE TO DECAY OR SINGLE ESF TRAIN OPERATION. REFER TO FSAR SECTION 12.2 FOR DECAY CORRECTIONS. NORMAL OPERATION SOURCES ARE NOT INCLUDED IN THESE ZONES.

NO GENERAL DESCRIPTION

1 HATCH
2 PIPE CHASE
3 ELEC. CHASE
4 HVAC. CHASE
5 SUMP
6 ELEVATOR
7 MONORAIL
8 VALVE GALLERY
9 MCC
10 LOAD CENTER
11 SWITCHGEAR
12 EQUIP. OR ACCESS OPENING
13 TRANSMITTER RACK
14 VOLTAGE REG.
15 RELAY CAB.
16 LIGHTING PANEL
17 DISTRIBUTION PANEL
18 JUNCTION BOX
19 PIPE TRENCH

DENOTES UP U.N.O.
INDICATES ONE-WAY DOOR OR CONTROLLED FROM INSIDE OF BLDG
GATE
△ INDICATES CONTROLLED ACCESS AND/OR ROLL UP DOOR

1

LEGEND:

ZONE 1 $X < 0.5$ mR/hr
ZONE 2 $0.5 \leq X < 2.5$ mR/hr
ZONE 3 $2.5 \leq X < 15$ mR/hr
ZONE 4 $15 \leq X < 100$ mR/hr
ZONE 5 $100 \leq X < 1000$ mR/hr
ZONE 6 $1 \leq X < 5$ R/hr
ZONE 7 $5 \leq X < 10$ R/hr
ZONE 8 $10 \leq X < 100$ R/hr
ZONE 9 $100 \leq X$ R/hr

Palo Verde Nuclear Generating Station
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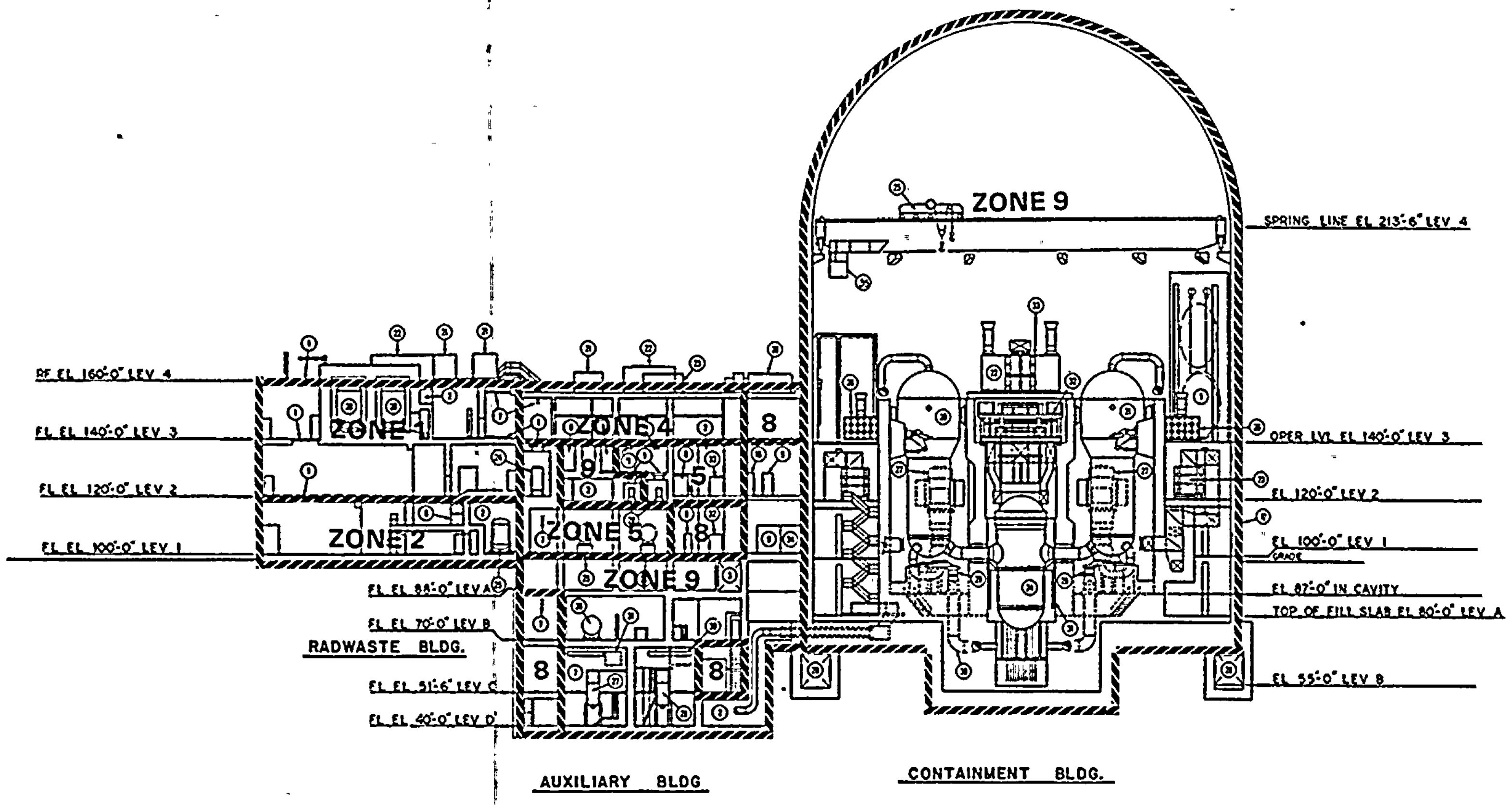
RADIATION ZONES
LOCA WITH SUMP RECIRCULATION
SECTION A

Figure II.B.2-9



SECTION J
002
003
004
005
006

NOTE. DEPICTED ZONES DO NOT INCORPORATE SOURCE REDUCTIONS DUE TO DECAY OR SINGLE ESF TRAIN OPERATION. REFER TO FSAR SECTION 12.2 FOR DECAY CORRECTIONS. NORMAL OPERATION SOURCES ARE NOT INCLUDED IN THESE ZONES.

NO	GENERAL DESCRIPTION
1	HATCH
2	PIPE CHASE
3	ELEC. CHASE
4	HVAC. CHASE
5	SUMP
6	ELEVATOR
7	MONORAIL
8	VALVE GALLERY
9	MCC
10	LOAD CENTER
11	SWITCHGEAR
12	EQUIP. OR ACCESS OPENING
13	TRANSMITTER RACK
14	VOLTAGE REG.
15	RELAY CAB.
16	LIGHTING PANEL
17	DISTRIBUTION PANEL
18	JUNCTION BOX
19	PIPE TRENCH

DENOTES UP U.N.O.
INDICATES ONE-WAY DOOR OR CONTROLLED FROM INSIDE OF BLDG
GATE
INDICATES CONT ACCESS &/OR ROLL UP DOORS

LEGEND:
ZONE 1 $X < 0.5$ mR/hr
ZONE 2 $0.5 \leq X < 2.5$ mR/hr
ZONE 3 $2.5 \leq X < 15$ mR/hr
ZONE 4 $15 \leq X < 100$ mR/hr
ZONE 5 $100 \leq X < 1000$ mR/hr
ZONE 6 $1 \leq X < 5$ R/hr
ZONE 7 $5 \leq X < 10$ R/hr
ZONE 8 $10 \leq X < 100$ R/hr
ZONE 9 $100 \leq X$ R/hr

	RADWASTE BLDG.		AUXILIARY BLDG		CONTAINMENT
20	WASTE GAS DECAY TANK	20	FUEL BLDG. SUPPLY AHU	20	STEAM GENERATOR NO. 2
21	RADWASTE BLDG. NORMAL EXCH. AHU	21	AUX. BLDG. ACCESS CONTROL AHU	21	STEAM GENERATOR NO. 1
22	RADWASTE BLDG. NORMAL SUPP. AHU	22	AUX. BLDG. SUPPLY AHU	22	CONTAINMENT CEDM NORMAL ACU
23		23	CONT. BLDG REFUELING PURGE AHU	23	NORMAL A.C.U.
24		24	PRE-HOLDUP ION EXCH.	24	REACTOR VESSEL
25	LOW ACTIVITY SPENT RESIN TANK	25	ECW HEAT EXCHANGER	25	POLAR CRANE
		26	SHUTDOWN COOLING HEAT EXCHANGER	26	CONTAIN. PRE-ACCESS NORMAL AFU
		27	LPSI PUMP	27	R.C. PUMP
		28	FILTERS	28	TENDON GALLERY
		29	CONT SPRAY PUMP	29	CAVITY COOLING FAN
		30	AUX. BLDG CS PUMP ROOM ESSENTIAL ACU	30	CAVITY COOLING DUCT
		31	AUX. BLDG. LPSI PUMP ROOM ESSENTIAL ACU	31	EX. CORE NEUTRON DETECTORS
		32	POST LOCA ANALYZER	32	C.E.D.M.
		33	120 V.A.C. DIST. PANEL	33	CEDM NORMAL AHU AND FAN
		34	LOAD CENTER	34	EMERGENCY SUMP SCREEN
				35	CAB

Palo Verde Nuclear Generating Station
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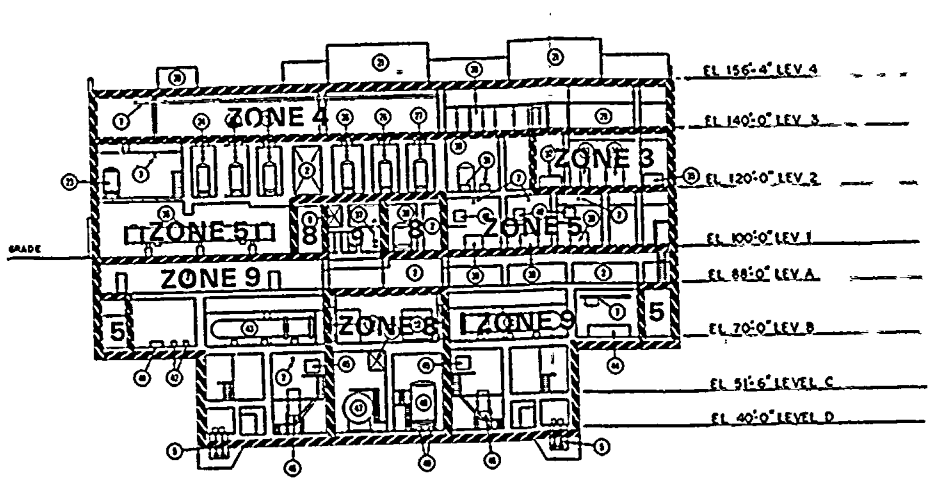
RADIATION ZONES
LOCA WITH SUMP RECIRCULATION
SECTION J-J

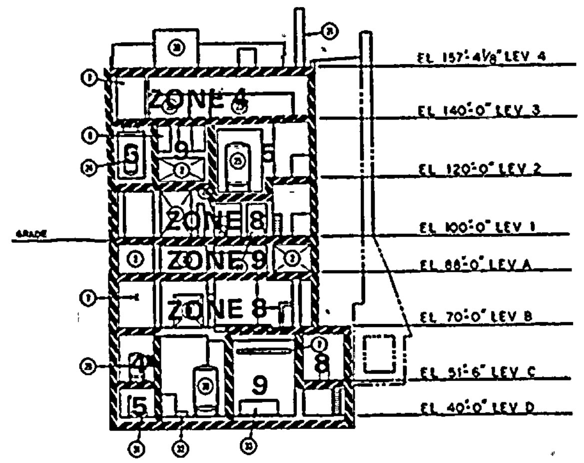
Figure II.B.2-10

1



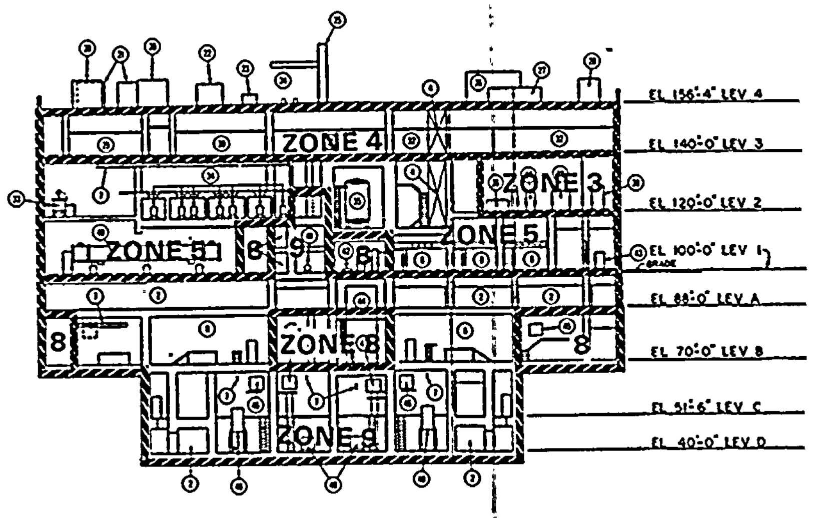
AUXILIARY BLDG.

SECTION E
002
003
004
005
006



AUXILIARY BLDG.

SECTION H
002
003
004
005
006



AUXILIARY BLDG.

SECTION D
002
003
004
005
006

NOTE: DEPICTED ZONES DO NOT INCORPORATE SOURCE REDUCTIONS DUE TO DECAY OR SINGLE ESF TRAIN OPERATION. REFER TO FSAR SECTION 12.2 FOR DECAY CORRECTIONS. NORMAL OPERATION SOURCES ARE NOT INCLUDED IN THESE ZONES.

AUXILIARY BUILDING SEC. "H"

20 AUX BLDG. EXHAUST A.F.U.
21 5 TON JIB CRANE
22 HEALTH PHYSICS
23 COLD LAB
24 DEBORATING ION EXCHANGER
25 VOLUME CONTROL TANK
26 CRUD PUMP
27 GAS STRIPPER
28 BLOCK OUT EFFLUENT RAD. MONITOR
29 CHEM. DRAIN TANK
30 COOLING WATER HOLD UP TANK
31 CHEM. DRAIN TANK PUMP
32 COOLING WATER HOLDUP TK PUMP
33 HPSI PUMP

AUXILIARY BUILDING SEC. "E"

20 AUX. BUILD. ACCESS CONT. AHU
21 AUX. BLDG. EXHAUST A.F.U.
23 ECW SURGE TANK
24 FUEL POOL ION EXCHANGE
25 PRE HOLDUP ION EXCHANGE
26 PURIFICATION ION EXCHANGE
27 DEBORATION ION EXCHANGE
28 WOMENS LOCKER ROOM
29 LUNCH ROOM
30 SPRAY CHEM. STORAGE TANK
31 SPRAY CHEM. STOR. PUMPS
32 MG SET
33 CONTROL PANEL
34 CEDM CONTR. CAB
35 CEDM CONTR. CAB R.M. A.C.U.
36 ECW HEAT EXCHANGE
37 SEAL INJECTION Hx
38 CRUD TANK
39 CHARGING PUMPS
40 A.C.U.
41 BORIC ACID MAKE UP PUMPS
42 REACTOR MAKE UP WATER PUMPS
43 SHUTDOWN COOLING Hx
44 ECW PUMP
45 LPSI PUMP R.M. A.C.U.
46 LPSI PUMPS
47 EQUIP. DRAIN TANK
48 COOLING WATER HOLD UP TANK
49 COOLING WATER HOLD UP TANK PUMPS
50

AUXILIARY BUILDING SEC. "D"

20 AUX BLDG. SUPPLY AHU
21 FUEL BLDG. SUPPLY AHU
22 CONT. BLDG. REFUELING PURGE AHU
23 CONT. BLDG. PRE-ACCESS AHU
24 FUME HOOD EXHAUST FANS
25 5 TON JIB CRANE
26 NORMAL CHILLER
27 CONT. PURGE EXHAUST AFU
28 STAIR WAY
29 COUNTING ROOM
30 HOT LAB
31 COLD LAB
32 MEN LOCKERS
33 BORIC ACID BATCHING TANK
34 FILTERS
35 VOLUME CONTROL TK
36 MG SET
37 CONTROL CAB
38 CEDM CONTR. CAB
39 CEDM CONTR. R.M. A.C.U.
40 ECW Hx
41 LET DOWN Hx
42 CRUD PUMP
43 H_2 RECOMBINER CONT. PANEL
44 GAS STRIPPER
45 ECW PUMP ROOM A.C.U.
46 CONT. SPRAY PUMP R.M. A.C.U.
47 HPSI PUMP RM. A.C.U.
48 CONT SPRAY PUMP
49 HPSI PUMP
50

(NO) GENERAL DESCRIPTION

1 HATCH
2 PIPE CHASE
3 ELEC. CHASE
4 HVAC. CHASE
5 SUMP
6 ELEVATOR
7 MONORAIL
8 VALVE GALLERY
9 MCC
10 LOAD CENTER
11 SWITCHGEAR
12 EQUIP. OR ACCESS OPENING
13 TRANSMITTER RACK
14 VOLTAGE REG.
15 RELAY CAB.
16 LIGHTING PANEL
17 DISTRIBUTION PANEL
18 JUNCTION BOX
19 PIPE TRENCH

DENOTES UP U.N.O.
INDICATES ONE-WAY DOOR OR CONTROLLED FROM INSIDE OF BLDG
G-GATE
△ INDICATES CONTROLLED ACCESS &/OR ROLL-UP DOORS

LEGEND:
ZONE 1 X < 0.5 mR/hr
ZONE 2 0.5 ≤ X < 2.5 mR/hr
ZONE 3 2.5 ≤ X < 15 mR/hr
ZONE 4 15 ≤ X < 100 mR/hr
ZONE 5 100 ≤ X < 1000 mR/hr
ZONE 6 1 ≤ X < 5 R/hr
ZONE 7 5 ≤ X < 10 R/hr
ZONE 8 10 ≤ X < 100 R/hr
ZONE 9 100 ≤ X R/hr

1

Palo Verde Nuclear Generating Station
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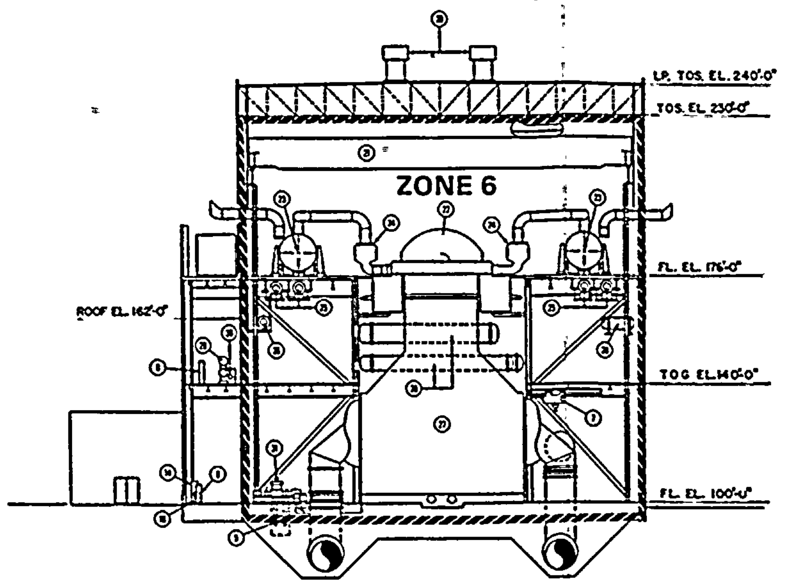
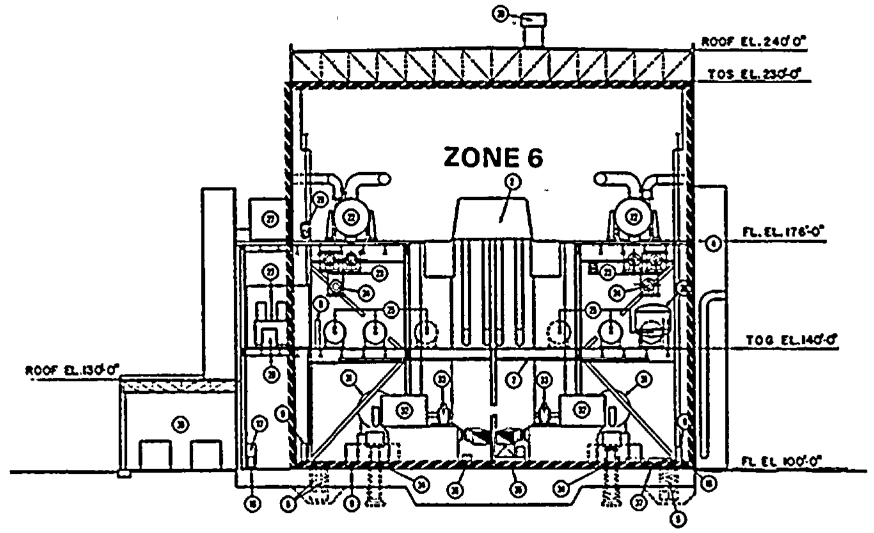
RADIATION ZONES
LOCA WITH SUMP RECIRCULATION
SECTIONS D, E, H

Figure II.B.2-11



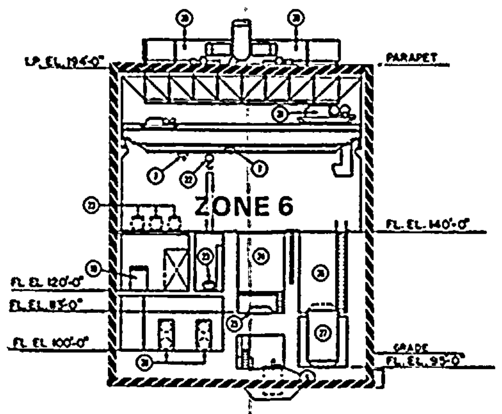
TURBINE BLDG.

SECTION B 003 004 005 006



TURBINE BLDG.

SECTION C 003 004 005 006



FUEL BLDG.

SECTION K 003 004 005 006

EQUIPMENT LIST FUEL BUILD. SEC. "K"

20 NORMAL EXHAUST FANS
21 150 TON CRANE
22 10 TON NEW FUEL CRANE
23 NEW FUEL CONTAINERS
24 DECON PIT
25 CASK LEAD LAYDOWN AREA
26 LOAD PIT
27 CASK
28 SPENT FUEL POOL Hx

EQUIPMENT LIST TURBINE BUILD. SEC. "B"

20 NORMAL EXHAUST FANS
21 280 TON CRANE
22 LOW PRESS. TURBINE
23 MOISTURE SEPARATOR REHEATOR
24 CONTROL INTERCEPT VALVE
25 REHEATER DRAIN TANKS
26 MSR DRAIN TANKS
27 CONDENSER "C"
28 L.P. FEED WATER HEATERS
29 STEAM SEAL PKG EXHAUSTER
30 MAIN CONDENSER EVAC. SYS. FILTER PKG
31 VACUUM PUMP

EQUIPMENT LIST TURBINE SECTION "C"

20 NORMAL EXHAUST FANS
21 H.P. TURBINE
22 MOISTURE SEPARATOR REHEATOR
23 REHEATER DRAIN TANKS
24 M.S.R. DRAIN TANK
25 FEEDWATER HEATERS
26 BLOW DOWN FLASH TANK
27 NORMAL AIR HANDLING UNIT
28 MAIN TURBINE L.O. CENTRIFUGE
29 MAIN TURBINE L.O. MIST ELIMINATOR
30 SWITCHGEAR ROOM
31 HEATER DRAIN TANK
32 FEEDWATER PUMP TURBINE
33 FEEDWATER PUMP
34 FEEDWATER DRAIN PUMP
35 EHC POWER UNIT
36 F.W. PUMP TURBINE L.O. CENTRIFUGE
37 AUX. F.W. PUMP
38 NORMAL A.H.U.

FUEL OIL STORAGE TANK

20 F.O. PUMP
21 STORAGE TANK

NOTE: DEPICTED ZONES DO NOT INCORPORATE SOURCE REDUCTIONS DUE TO DECAY OR SINGLE ESF TRAIN OPERATION. REFER TO FSAR SECTION 12.2 FOR DECAY CORRECTIONS. NORMAL OPERATION SOURCES ARE NOT INCLUDED IN THESE ZONES.

(NO) GENERAL DESCRIPTION

1 HATCH
2 PIPE CHASE
3 ELEC. CHASE
4 HVAC. CHASE
5 SUMP
6 ELEVATOR
7 MONORAIL
8 VALVE GALLERY
9 MCC
10 LOAD CENTER
11 SWITCHGEAR
12 EQUIP. OR ACCESS OPENING
13 TRANSMITTER RACK
14 VOLTAGE REG.
15 RELAY CAB.
16 LIGHTING PANEL
17 DISTRIBUTION PANEL
18 JUNCTION BOX
19 PIPE TRENCH

DENOTES UP U.N.O.
■ INDICATES ONE-WAY DOOR OR CONTROLLED FROM INSIDE OF BLDG
Ⓖ GATE
△ INDICATES CONTROLLED ACCESS &/OR ROLL-UP DOORS

1

LEGEND:

ZONE 1 $X < 0.5$ mR/hr
ZONE 2 $0.5 \leq X < 2.5$ mR/hr
ZONE 3 $2.5 \leq X < 15$ mR/hr
ZONE 4 $15 \leq X < 100$ mR/hr
ZONE 5 $100 \leq X < 1000$ mR/hr
ZONE 6 $1 \leq X < 5$ R/hr
ZONE 7 $5 \leq X < 10$ R/hr
ZONE 8 $10 \leq X < 100$ R/hr
ZONE 9 $100 \leq X$ R/hr

Palo Verde Nuclear Generating Station
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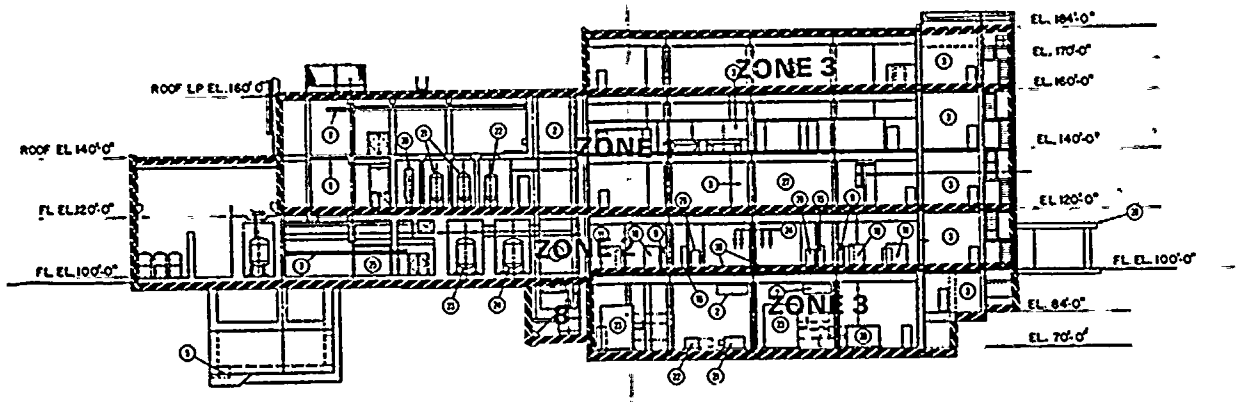
RADIATION ZONES
LOCA WITH SUMP RECIRCULATION
SECTIONS B, C, K

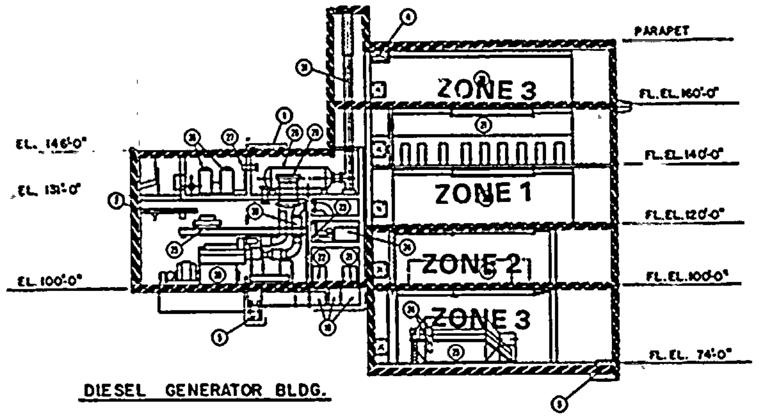
Figure II.B.2-12



RADWASTE BLDG.

CONTROL BLDG.

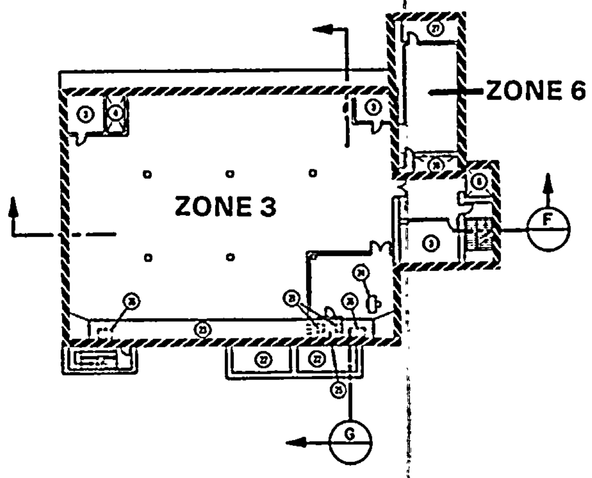
SECTION F



DIESEL GENERATOR BLDG.

CONTROL BLDG.

SECTION G



CONTROL BLDG.

PLAN EL. 160'-0"

AUXILIARY BUILDING EL. 88 FT. — 6 IN.

19
20 GAS STRIPPER
21 NONNUCLEAR SAMPLE COOLERS
22 FLOOR GRATING AND FIRE DAMPER
23 PIPEWAY "TRAIN "A" "
24 PIPEWAY TRAIN "B"

CONTROL BUILDING EL. 160 FT. — 0 IN.

20 SPRINKLING VALVE STATION
21 CHLORINE DETECTOR
22 DIESEL GEN. EXHAUST STACKS
23 H.V.A.C. OUTSIDE AIR INTAKE PLENUM
24 NORMAL SMOKE EXHAUST FAN
25 OUTSIDE AIR CHASE
26 GASEOUS RADIATION MONITOR
27 BATTERY ROOM

DIESEL GENERATOR SEC. "E"

19
20 DIESEL GENERATOR DGB-H01
21 LOW VOLTAGE CON.
22 HIGH VOLTAGE CON.
23 H.V.A.C. CONTROL POWER
24 DIESEL GENERATOR CONTROL RM.
25 5 TON CRANE
26 STARTING AIR PACKAGE
27 VENT FAN
28 EXHAUST SILENCER
29 DIESEL EXHAUST FAN
30 INTAKE AIR SILENCER
31 DIESEL EXHAUST STACK

CONTROL BUILDING SEC. "G"

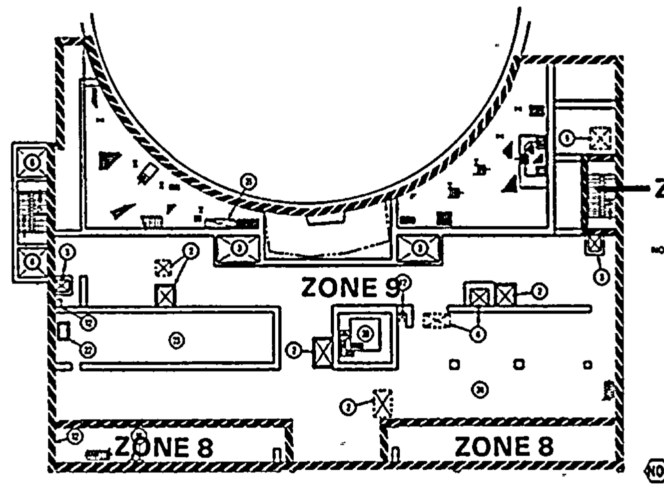
20 UPPER CABLE SPREADING ROOM
21 MAIN CONTROL ROOM
22 LOWER CABLE SPREADING ROOM
23 4.16 KV SWGR
24 COOLING COIL
25 ESSENTIAL A.H.V.

RADWASTE BUILDING SEC. "F"

20 B.A.C. ION EXCHANGER
21 L.R.S. ION EXCHANGER
22 L.R.S. ABSORPTION BED
23 HIGH ACTIVITY SPENT RESIN TANK
24 LOW ACTIVITY SPENT RESIN TANK
25 DECON BOOTH

CONTROL BUILDING SEC. "F"

20
21 CONTROL BLDG. OUTSIDE AIR. AIR WASHER UNIT
22 CONTROL BLDG. ESF SWGR A.H.V.
23 CONTROL RM. ESSENTIAL A.H.V.
24 EXHAUST FANS
25 MAIN CONTROL ROOM
26 UPPER CABLE SPREADING RM.
27 LOWER CABLE SPREADING RM.
28 CANOPY ROOF
29 EMERGENCY EYE WASH AND ABOVE
30 BATTERIES



AUXILIARY BLDG.

PLAN EL. 88'-0"

GENERAL
(NO) DESCRIPTION

1 HATCH
2 PIPE CHASE
3 ELEC. CHASE
4 HVAC. CHASE
5 SUMP
6 ELEVATOR
7 MONORAIL
8 VALVE GALLERY
9 MCC
10 LOAD CENTER
11 SWITCHGEAR
12 EQUIP. OR ACCESS OPENING
13 TRANSMITTER RACK
14 VOLTAGE REG.
15 RELAY CAB.
16 LIGHTING PANEL
17 DISTRIBUTION PANEL
18 JUNCTION BOX
19 PIPE TRENCH

DENOTES UP U.N.O.
INDICATES ONE-WAY DOOR OR CONTROLLED FROM INSIDE OF BLDG
(G) GATE
△ INDICATES CONT. ACCESS &/OR ROLLUP DRS.

LEGEND:
ZONE 1 $X < 0.5$ mR/hr
ZONE 2 $0.5 \leqslant X < 2.5$ mR/hr
ZONE 3 $2.5 \leqslant X < 15$ mR/hr
ZONE 4 $15 \leqslant X < 100$ mR/hr
ZONE 5 $100 \leqslant X < 1000$ mR/hr
ZONE 6 $1 \leqslant X < 5$ R/hr
ZONE 7 $5 \leqslant X < 10$ R/hr
ZONE 8 $10 \leqslant X < 100$ R/hr
ZONE 9 $100 \leqslant X$ R/hr

1

Palo Verde Nuclear Generating Station
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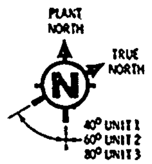
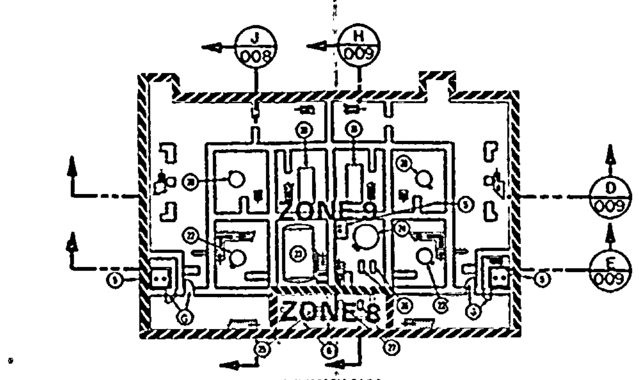
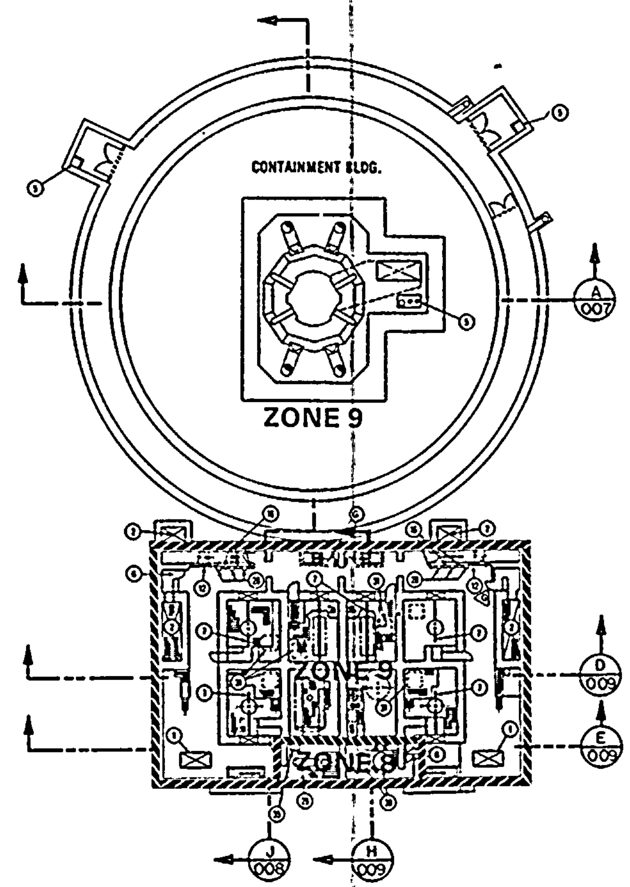
RADIATION ZONES
LOCA WITH SUMP RECIRCULATION
PLAN AT AUX. 88', CONTROL 160',
SECTS. F, G
Figure II.B.2-13

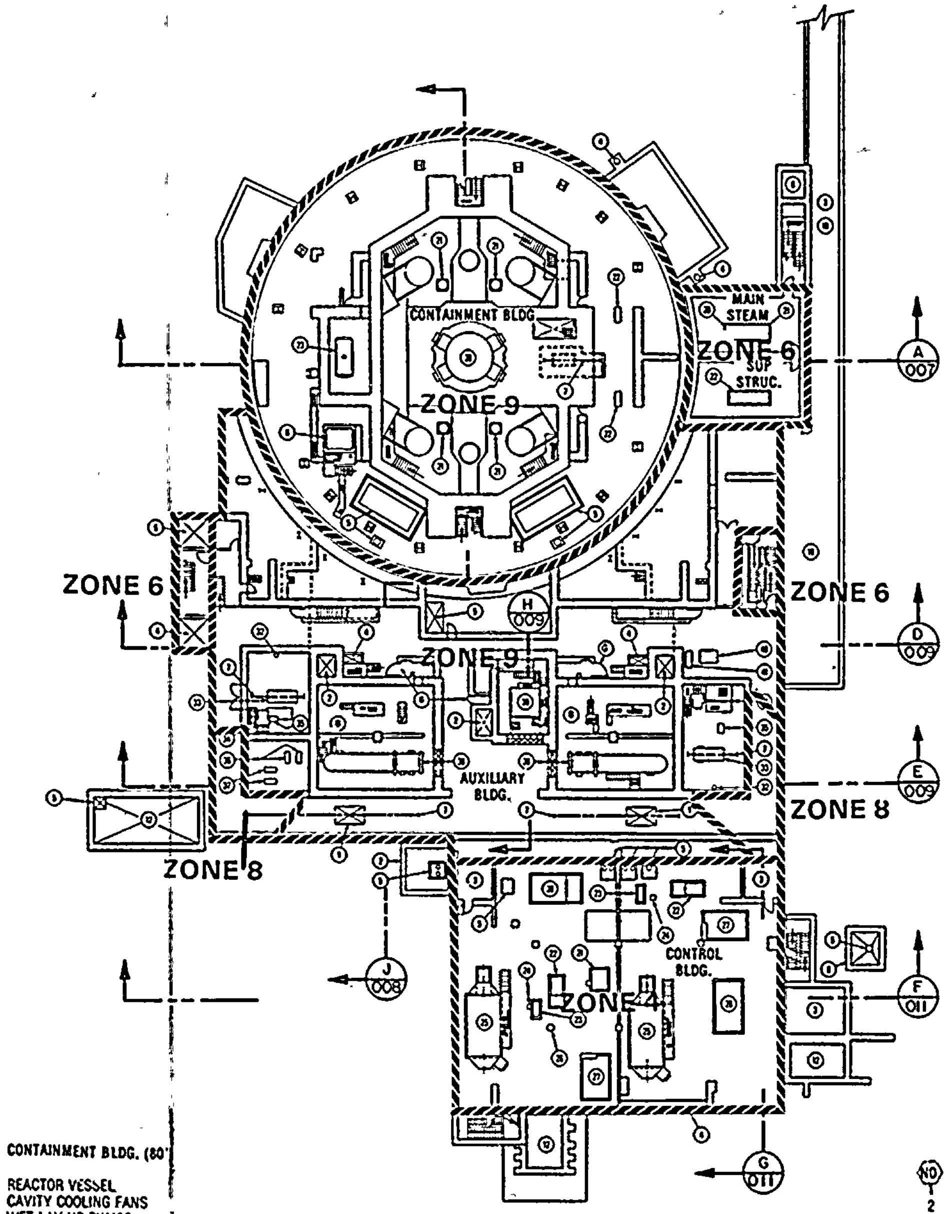




AUXILIARY BLDG.
ELEV. 40 FT. — 0 IN. LEVEL D



AUXILIARY BLDG.
ELEV. 51 FT. — 6 IN. LEVEL C



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CONTAINMENT BLDG. (80')

20	REACTOR VESSEL
21	CAVITY COOLING FANS
22	WET LAY-UP PUMPS
23	REACTOR DRAIN TANK

MSSS (81')

20	AUX. FEEDWATER TURBINE PACKAGE
21	TUBINE DRIVEN PUMP
22	MOTOR DRIVEN PUMP

CONTROL BLDG. (74')

20	CONTROL BLDG. NORMAL AHU'S
21	CONTROL BLDG. OUTSIDE AIR WASHER UNIT
22	CONTROL BLDGE. ESF SWGR RM. ESSENTIAL AHU'S
23	ESSENTIAL CHILLED WATER CIRCULATION PUMPS
24	CHEMICAL ADDITION TANKS
25	CONTROL ROOM ESSENTIAL AHU'S
26	ESSENTIAL CHILLED WATER EXPANSION TANK
27	ESSENTIAL CHILLERS

AUX. BLDG. BETWEEN 40' AND 100'

20	CONT. SPRAY PUMPS
21	HPSI. PUMPS
22	LPSI. PUMPS
23	EQUIP. DRAIN TANK
24	COOLING WATER HOLD-UP TANK
25	HEATER DRAIN PUMPS
26	COOLING WATER HOLD-UP TANK PUMPS
27	CHEMICAL DRAIN TANK PUMPS
28	TRANSFORMERS
29	AUX. STEAM CONDENSATE RECEIVER TANK
30	CHEMICAL DRAIN TANKS
31	ESSENTIAL AIR CONDITIONING UNITS
32	ECW CHEMICAL ADDITION TANKS
33	ECW PUMPS
34	ECW PUMP ROOM ESSENTIAL A.C.U.
35	ECW RADIATION MONITORS
36	REACTOR MAKE-UP WATER PUMPS
37	BORIC ACID MAKE-UP PUMPS
38	SHUTDOWN COOLING HEAT EXCHANGERS
39	GAS STRIPPER
40	GAS STRIPPER CONTROL CABINET
41	GAS STRIPPER INSTRUMENT RACK

NO	DESCRIPTION
1	HATCH
2	PIPE CHASE
3	ELEC. CHASE
4	HVAC CHASE
5	SUMP
6	ELEVATOR
7	MONORAIL
8	VALVE GALLERY
9	MCC
10	LOAD CENTER
11	SWITCHGEAR
12	EQUIP. OR ACCESS OPENING
13	TRANSMITTER RACK
14	VOLTAGE REG.
15	RELAY CAB.
16	LIGHTING PANEL
17	DISTRIBUTION PANEL
18	JUNCTION BOX
19	PIPE TRENCH

LEGEND:

ZONE 1	$X < 0.5$ mR/hr
ZONE 2	$0.5 \leqslant X < 2.5$ mR/hr
ZONE 3	$2.5 \leqslant X < 15$ mR/hr
ZONE 4	$15 \leqslant X < 100$ mR/hr
ZONE 5	$100 \leqslant X < 1000$ mR/hr
ZONE 6	$1 \leqslant X < 5$ R/hr
ZONE 7	$5 \leqslant X < 10$ R/hr
ZONE 8	$10 \leqslant X < 100$ R/hr
ZONE 9	$100 \leqslant X$ R/hr

→ DENOTES UP U.N.O.
INDICATES ONE-WAY DOOR OR CONTROLLED FROM INSIDE OF BLDG
GATE
▷ INDICATES CONTROLLED ACCESS AND/OR ROLL UP DOOR

Palo Verde Nuclear Generating Station
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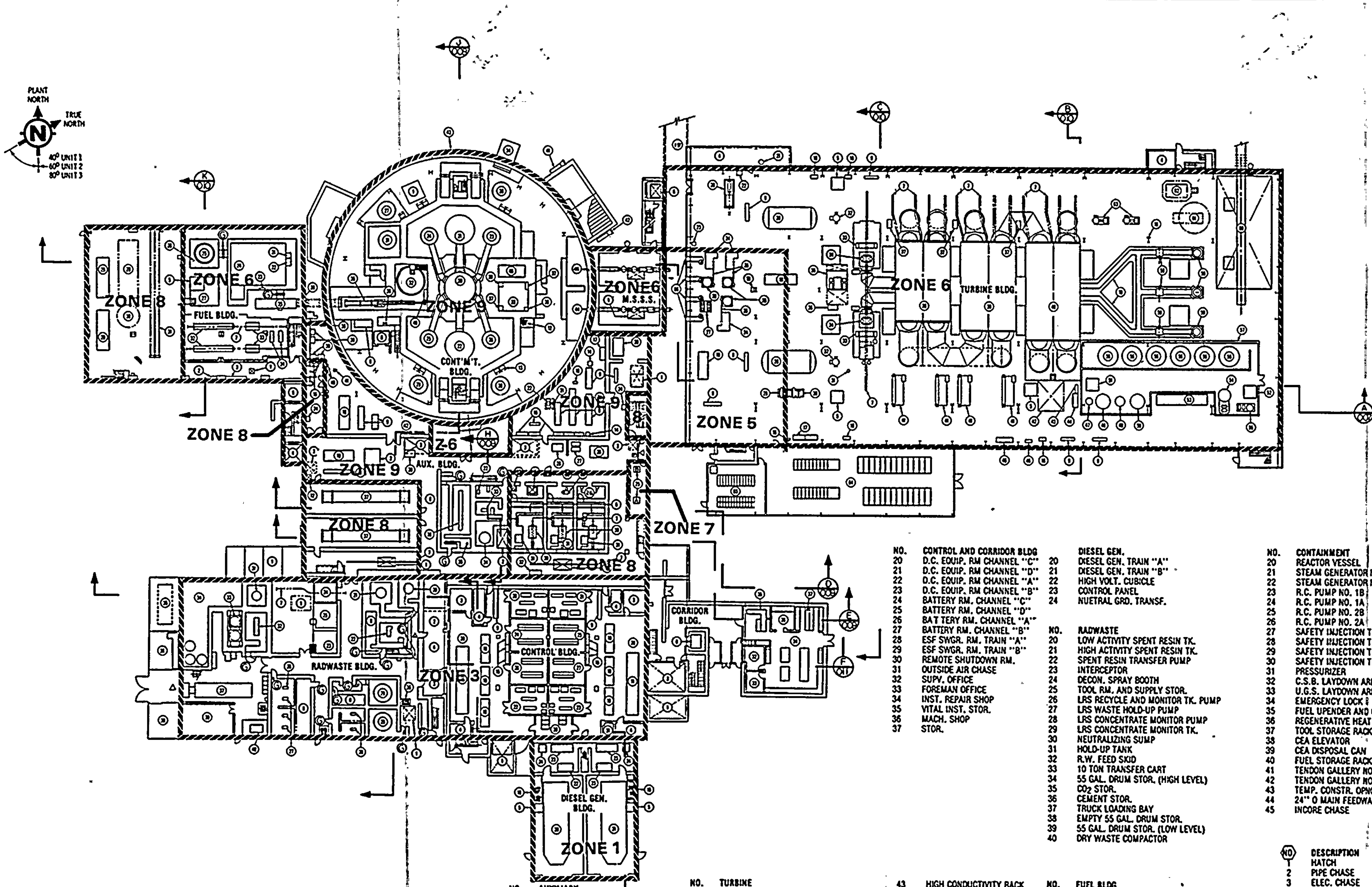
RADIATION ZONES - LOCA
DEGRADED CORE - INTACT PRIMARY
BETWEEN EL. 40'-0" & 100'-0"

Figure II.B.2-14



NOTE: DEPICTED ZONES DO NOT INCORPORATE SOURCE REDUCTIONS DUE TO DECAY OR SINGLE ESF TRAIN OPERATION. REFER TO FSAR SECTION 12.2 FOR DECAY CORRECTIONS. NORMAL OPERATION SOURCES ARE NOT INCLUDED IN THESE ZONES.

NO.	CONTROL AND CORRIDOR BLDG
20	D.C. EQUIP. RM CHANNEL "C"
21	D.C. EQUIP. RM CHANNEL "D"
22	D.C. EQUIP. RM CHANNEL "A"
23	D.C. EQUIP. RM CHANNEL "B"
24	BATTERY RM. CHANNEL "C"
25	BATTERY RM. CHANNEL "D"
26	BAT TERY RM. CHANNEL "A"
27	BATTERY RM. CHANNEL "B"
28	ESF SWGR. RM. TRAIN "A"
29	ESF SWGR. RM. TRAIN "B"
30	REMOTE SHUTDOWN RM.
31	OUTSIDE AIR CHASE
32	SUPV. OFFICE
33	FOREMAN OFFICE
34	INST. REPAIR SHOP
35	VITAL INST. STOR.
36	MACH. SHOP
37	STOR.

	DIESEL GEN.
20	DIESEL GEN. TRAIN "A"
21	DIESEL GEN. TRAIN "B"
22	HIGH VOLT. CUBICLE
23	CONTROL PANEL
24	NUETRAL GRD. TRANSF.

NO.	RADWASTE
20	LOW ACTIVITY SPENT RESIN TK.
21	HIGH ACTIVITY SPENT RESIN TK.
22	SPENT RESIN TRANSFER PUMP
23	INTERCEPTOR
24	DECON. SPRAY BOOTH
25	TOOL RM. AND SUPPLY STOR.
26	LRS RECYCLE AND MONITOR TK. PUMP
27	LRS WASTE HOLD-UP PUMP
28	LRS CONCENTRATE MONITOR PUMP
29	LRS CONCENTRATE MONITOR TK.
30	NEUTRALIZING SUMP
31	HOLD-UP TANK
32	R.W. FEED SKID
33	10 TON TRANSFER CART
34	55 GAL. DRUM STOR. (HIGH LEVEL)
35	CO_2 STOR.
36	CEMENT STOR.
37	TRUCK LOADING BAY
38	EMPTY 55 GAL. DRUM STOR.
39	55 GAL. DRUM STOR. (LOW LEVEL)
40	DRY WASTE COMPACTOR

NO.	CONTAINMENT
20	REACTOR VESSEL
21	STEAM GENERATOR NO. 1
22	STEAM GENERATOR NO. 2
23	R.C. PUMP NO. 1B
24	R.C. PUMP NO. 1A
25	R.C. PUMP NO. 2B
26	R.C. PUMP NO. 2A
27	SAFETY INJECTION TK. NO. 1A
28	SAFETY INJECTION TK. NO. 1B
29	SAFETY INJECTION TK. NO. 2B
30	SAFETY INJECTION TK. NO. 2A
31	PRESSURIZER
32	C.S.B. LAYDOWN AREA
33	U.G.S. LAYDOWN AREA
34	EMERGENCY LOCK
35	FUEL UPENDER AND CARRIAGE
36	REGENERATIVE HEAT EXCH.
37	TOOL STORAGE RACK
38	CEA ELEVATOR
39	CEA DISPOSAL CAN
40	FUEL STORAGE RACK
41	TENDON GALLERY NORMAL EXH. FAN
42	TENDON GALLERY NORMAL SUPPLY FAN
43	TEMP. CONSTR. OPNG.
44	24" O MAIN FEEDWATER
45	INCORE CHASE

NO.	AUXILIARY
20	H_2 RECOMBINER
21	ESSENTIAL ACU
22	RAD. MONITOR
23	ELEC. PENET. RM TRAIN "B"
24	CONCENTRATOR PANEL
25	H_2 RECOMBINER CONTROL PANEL
26	H_2 COMPRESSOR
27	H_2 PURGE EXH.
28	POST LOCA ANALYZER
29	CVCS CHEM ADDIT.
30	CHARGING PUMPS
31	AIR HANDLING UNITS
32	TRANSFER SWITCH
33	CRUD PUMP
34	CRUD TANK
35	SEAL INJ. HEAT EXCH.
36	LETDOWN HEAT EXCH.
37	ECW HEAT EXCH.
38	SCR POWER CONTROLLER
39	HEATER PANELS
40	LIGHTING TRANSF
41	PENET. ALARM
42	ELEC. PENET. RM. TRAIN "C"

NO.	TURBINE
20	AUX. F.W. PUMP
21	BLOWDOWN STACK
22	HVAC ANNUC. BD.
23	SEQUENCE CONT. PNL.
24	SERV. AND INST. AIR COMP.
25	AFTERCOOLER
26	AIR REC.
27	AIR DRYER
28	HTR. DRN. TK.
29	AMMONIA ADDIT. PUMP
30	HYDRAZINE ADDIT. PUMP
31	STM. PKG. EXH. DRN. TK.
32	HTR. DRN. PUMP
33	F.W. PUMP TURB.
34	F.W. PUMP TURB.
35	EHC POWER UNIT
36	F.W. PUMP TURB L.O. CENT.
37	HOTWELL COND. ANALYZERS
38	CONDENSER "A"
39	CONDENSER "B"
40	CONDENSER "C"
41	VACUUM PUMPS
42	FUTURE SPACE (COND. TUBE CLNG. EQUIP.)
43	HIGH CONDUCTIVITY RACK
44	HIGH SODIUM ANALYZER
45	POWER INST. PNL.
46	VOLT. REG.
47	WASTE COLL. TK.
48	CATION TK
49	ANION TK
50	MIX AND HOLD TK
51	LOW TDS SUMP
52	HIGH TDS SUMP
53	CONTROL PNL.
54	CAUSTIC PNL.
55	ACID SKID
56	COND. POLISH. DEMIN. TK
57	FUTURE WALLS
58	COND. PUMP STRAINER
59	COND. PUMP STRAINER
60	R.R. RAILS
61	PORTA-BOR
62	PORTA-BOR CONTROL CENTER
63	TURB. COOLING WTR. PUMP
64	SWITCHGEAR RM.
65	BATTERY RM
66	MAIN STM LINES

NO.	FUEL BLDG
20	TRANSFER TUBE
21	FAILED FUEL CONTAINER
22	STOR. RACKS (IN-CODE INST. SURV. SAMPLES)
23	TOOL STOR.
24	SPENT FUEL STOR. RACKS
25	CASK
26	CASK LOADING PIT
27	STORAGE RM
28	CASK PERSONNEL BARRIER LAYDOWN
29	EXH. ESSENTIAL AIR FILTRATION UNIT
30	UPPER AND LOWER CRUSH CONE LAYDOWN
31	R.R. RAILS
32	SPENT FUEL POOL HEAT EXCH.
33	SPENT FUEL COOLING PUMP
34	POOL CLEANUP PUMP
35	NEW FUEL ELEV.

NO	DESCRIPTION
1	HATCH
2	PIPE CHASE
3	ELEC. CHASE
4	HVAC. CHASE
5	SUMP
6	ELEVATOR
7	MONORAIL
8	VALVE GALLERY
9	MCC
10	LOAD CENTER
11	SWITCHGEAR
12	EQUIP. OR ACCESS OPENING
13	TRANSMITTER RACK
14	VOLTAGE REG.
15	RELAY CAB.
16	LIGHTING PANEL
17	DISTRIBUTION PANEL
18	JUNCTION BOX,
19	PIPE TRENCH

LEGEND:

ZONE 1 $X < 0.5$ mR/hr
ZONE 2 $0.5 \leqslant X < 2.5$ mR/hr
ZONE 3 $2.5 \leqslant X < 15$ mR/hr
ZONE 4 $15 \leqslant X < 100$ mR/hr
ZONE 5 $100 \leqslant X < 1000$ mR/hr
ZONE 6 $1 \leqslant X < 5$ R/hr
ZONE 7 $5 \leqslant X < 10$ R/hr
ZONE 8 $10 \leqslant X < 100$ R/hr
ZONE 9 $100 \leqslant X$ R/hr

DENOTES UP U.N.O.
INDICATES ONE-WAY DOOR OR CONTROLLED FROM INSIDE OF BLDG
(G) GATE
△ INDICATES CONT ACCESS &/OR ROLL DRS

1

Palo Verde Nuclear Generating Station
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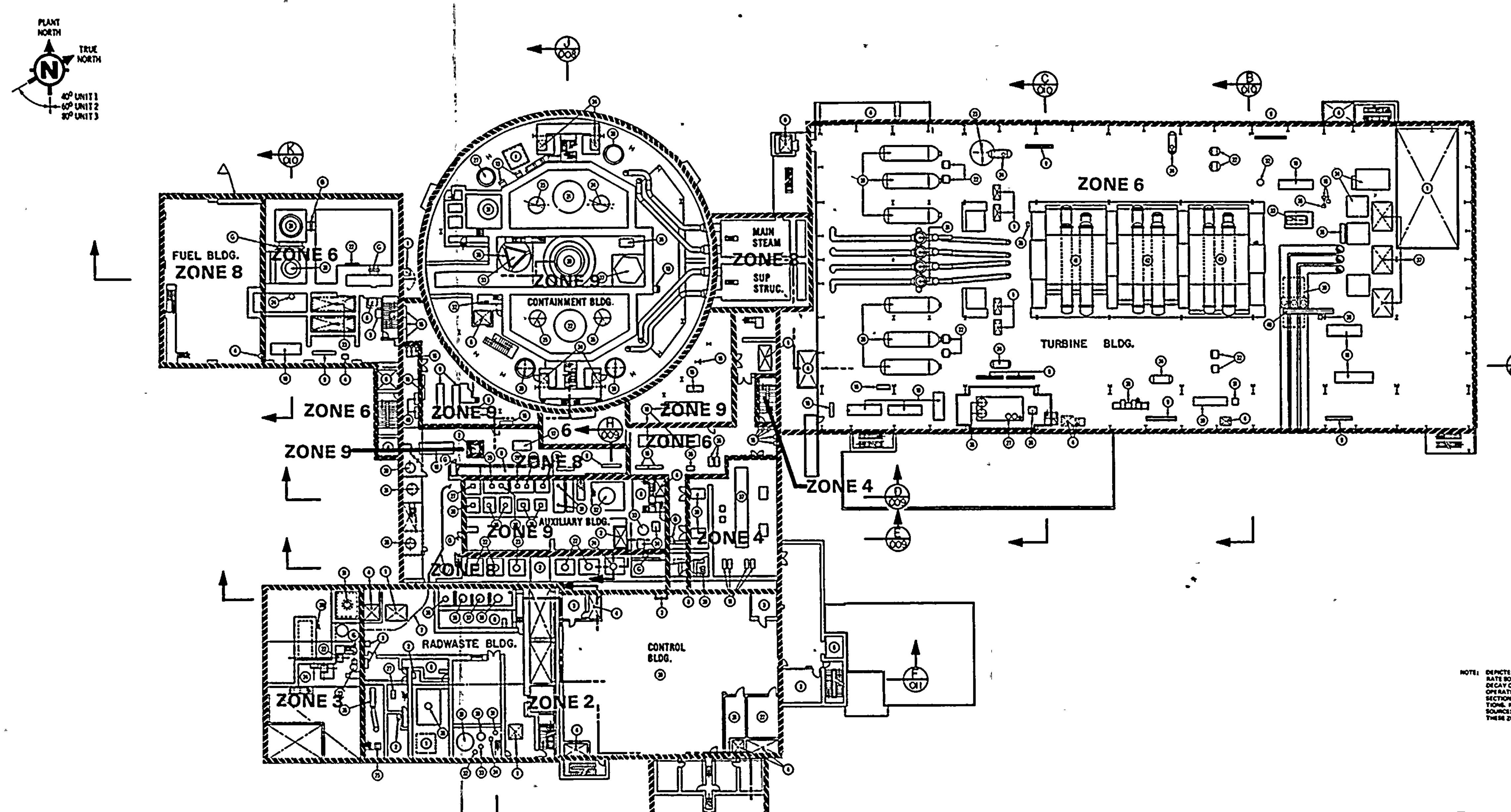
RADIATION ZONES - LOCA
DEGRADED CORE - INTACT PRIMARY
AT EL. 100'-0"

Figure II.B.2-15

August 1981 Amendment 1



NOTE: DEPICTED ZONES DO NOT INCORPORATE SOURCE REDUCTIONS DUE TO DECAY OR SINGLE ESF TRAIN OPERATION. REFER TO FSAR SECTION 12.2 FOR DECAY CORRECTIONS. NORMAL OPERATION SOURCES ARE NOT INCLUDED IN THESE ZONES.

EQUIPMENT LIST CONTAINMENT BLDG. 120'-0"

20 REACTOR VESSEL
21 STEAM GENERATOR NO. 1
22 STEAM GENERATOR NO. 2
23 REACTOR COOLANT PUMP 1A
24 REACTOR COOLANT PUMP 1B
25 REACTOR COOLANT PUMP 2A
26 REACTOR COOLANT PUMP 2B
27 SAFETY INJECTION TANK 1A
28 SAFETY INJECTION TANK 1B
29 SAFETY INJECTION TANK 2A
30 SAFETY INJECTION TANK 2B
31 PRESSURIZER
32 REGENERATIVE HTx
33 CORE SUPPORT BARREL AREA
34 NORMAL ACC
35 C.S.B LIFT RIG
36 SEAL TABLE
37 UPPER GUIDE STRUCT. LIFT RIG
38
39
40

EQUIPMENT LIST FOR RADWASTE BLDG ELEV. 120

20 TV MONTOR
21
22
23
24
25 EVAPORATOR CONCENTR PUMPS
26 EVAPORATOR MAIN RECYCLE PUMP
27 EVAPORATOR STEAM CONDENSATE PUMP
28 BORIC ACID CONCENTRATOR
29 CAUSTIC TANK
30 ACID TANK
31 ANTI-FOAM TANK
32 CAUSTIC BATCH TANK
33 ACID BATCH TYPE
34 ANTI-FOAM PUMP
35 BORIC ACID CON. INOx
36 L.R.S. MIXED BED INOx
37 L.R.S. MIXED BED IONx
38 L.R.S. ADSORPTION BED

EQUIPMENT LIST CONTROL BLDG. 120'-0'

20 CABLE SPREADING ROOM
21 INVERTER ROOM
22 COMMUNICATIONS ROOM

EQUIPMENT LIST AND AUXILIARY BLDG. ELEV. 120'-0"

20 ECW SURGE TANK
21 BORIC ACID BATCH TANK
22 FUEL POOL PURIFICATION ION EXCH.
23 PRE HOLD UP ION EXCH.
24 DEBORATING ION EXCH.
25 SAFETY INJECTION FILTERS
26 PURIFICATION FILTERS
27 BORIC ACID FILER
28 REACTOR DRAIN FILTER
29 LRS FILTERS
30 FUEL POOL PURIFICATION FILTERS
31 CRUD TANK FILTERS
32 VOLUME CONTROL TANK
33 SPRAY CHEM STORAGE
34 SPRAY CHEM ADD. PUMPS
35 EXHAUST RAD MONITOR
36 AUX CABINETS
37 CEDM CONTROL CAB. S
38 PROCESS RAD MONITOR
39 MG SET
40 HEAT TRACE CAB.

EQUIPMENT LIST, TURBINE BLDG. 120'-0"

20 FEED WATER HEATERS
21 MAIN STEAM STOP VALVES
22 REHEATER DRAIN TANKS
23 BLOW DOWN FLUSH TANK
24 MSR DRAIN TANK
25 VAPOR EXTRACTOR
26 OIL COOLERS
27 LUBE OIL RESERVOIR
28 LUBE OIL CENTRIFUGE
29 STEAM SEAL PKG.
30 MAIN CONDENSER EVAC. SYS. FILTER PKG.
31 POST FILTER BLOWER
32 SURGE TANK
33 H_2 SEAL OIL UNIT
34 GENERATOR STATOR COOLING UNIT
35 DC. MTR. STR.
36 H_2 AND STATOR COOLING CONTROL
37 CONDENSATE PUMPS AND REMOVE
38 H_2 DETECTION CABINET
39 ISO PHASE BUS COOLING
40 AUX DETRAINING TANK
41 CONDENSER "A"
42 CONDENSER "B"
43 CONDENSER "C"

EQUIPMENT LIST FUEL BLDG 120'-0"

20 CASK HEAD LAYDOWN
21 CASK
22 TOOL STORAGE
23 NEW FUEL STORAGE RACKS
24 HANDLING SLEEVE

NO	DESCRIPTION
1	HATCH
2	PIPE CHASE
3	ELEC. CHASE
4	HVAC. CHASE
5	SUMP
6	ELEVATOR
7	MONORAIL
8	VALVE GALLERY
9	MCC
10	LOAD CENTER
11	SWITCHGEAR
12	EQUIP. OR ACCESS OPENING
13	TRANSMITTER RACK
14	VOLTAGE REG.
15	RELAY CAB.
16	LIGHTING PANEL
17	DISTRIBUTION PANEL
18	JUNCTION BOX
19	PIPE TRENCH

LEGEND:

ZONE 1 $X < 0.5$ mR/hr
ZONE 2 $0.5 \leqslant X < 2.5$ mR/hr
ZONE 3 $2.5 \leqslant X < 15$ mR/hr
ZONE 4 $15 \leqslant X < 100$ mR/hr
ZONE 5 $100 \leqslant X < 1000$ mR/hr
ZONE 6 $1 \leqslant X < 5$ R/hr
ZONE 7 $5 \leqslant X < 10$ R/hr
ZONE 8 $10 \leqslant X < 100$ R/hr
ZONE 9 $100 \leqslant X$ R/hr

→ DENOTES UP U.N.O.
INDICATES ONE-WAY DOOR OR CONTROLLED FROM INSIDE OF BLDG
Ⓖ G - GATE
△ INDECATES CONTROLLED ACCESS AND/OR ROLL-UP DOOR

1

Palo Verde Nuclear Generating Station
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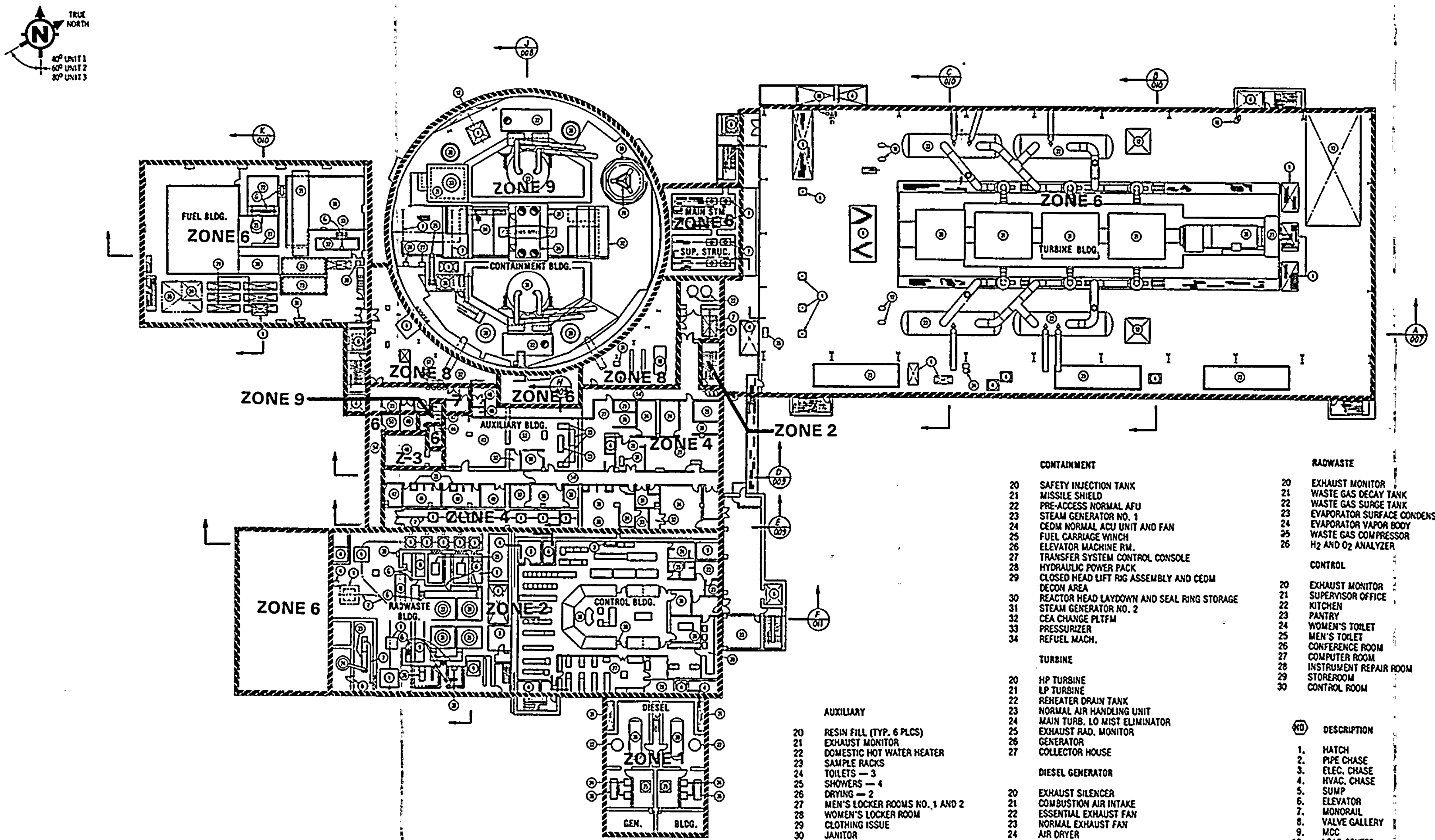
RADIATION ZONES - LOCA
DEGRADED CORE - INTACT PRIMARY
BETWEEN EL. 120'-0" & 140'-0"

Figure II.B.2-16



CONTAINMENT

20 SAFETY INJECTION TANK
21 MISSILE SHIELD
22 PRE-ACCESS NORMAL AFU
23 STEAM GENERATOR NO. 1
24 CEDM NORMAL ACU UNIT AND FAN
25 FUEL CARRIAGE WINCH
26 ELEVATOR MACHINE RM.
27 TRANSFER SYSTEM CONTROL CONSOLE
28 HYDRAULIC POWER PACK
29 CLOSED HEAD LIFT RIG ASSEMBLY AND CEDM DECON AREA
30 REACTOR HEAD LAYDOWN AND SEAL RING STORAGE
31 STEAM GENERATOR NO. 2
32 CEA CHANGE PLTFM
33 PRESSURIZER
34 REFUEL MACH.

TURBINE

20 HP TURBINE
21 LP TURBINE
22 REHEATER DRAIN TANK
23 NORMAL AIR HANDLING UNIT
24 MAIN TURB. LO MIST ELIMINATOR
25 EXHAUST RAD. MONITOR
26 GENERATOR
27 COLLECTOR HOUSE

DIESEL GENERATOR

20 EXHAUST SILENCER
21 COMBUSTION AIR INTAKE
22 ESSENTIAL EXHAUST FAN
23 NORMAL EXHAUST FAN
24 AIR DRYER
25 FUEL OIL DAY TANKS
26 STARTING AIR PACKAGE

FUEL

20 SPENT FUEL POOL
21 SPENT FUEL HANDLING MACHINE
22 CASK LOADING PIT
23 NEW FUEL STORAGE COVERS
24 CASK LIFTING RIG LAYDOWN AREA
25 DECON PIT
26 DECON PIT COVER
27 GATE STORAGE
28 NEW FUEL INSPECTION PIT
29 NEW FUEL CONTAINERS LAYDOWN AREA
30 UPENDER HYD. DRIVE UNIT
31 POOL COOLING PURIFICATION PANEL
32 CANAL
33 NEW FUEL ELEVATOR

AUXILIARY

20 RESIN FILL (TYP. 6 PLCS)
21 EXHAUST MONITOR
22 DOMESTIC HOT WATER HEATER
23 SAMPLE RACKS
24 TOILETS — 3
25 SHOWERS — 4
26 DRYING — 2
27 MEN'S LOCKER ROOMS NO. 1 AND 2
28 WOMEN'S LOCKER ROOM
29 CLOTHING ISSUE
30 JANITOR
31 CLEAN STORAGE
32 STORAGE — 3
33 COT ROOM
34 LUNCH ROOM
35 H. P. OFFICE
36 HEALTH PHYSICS
37 INST. RESP. MAINT. ISSUE
38 OFFICE
39 SAMPLING AREA
40 FIRST AID
41 PERSONNEL DECON AREA — 2
42 H. P. SAMPLE COUNT. ROOM
43 HOT LAB
44 SAMPLE ROOM HOOD AND SINK
45 FLAMMABLE STORAGE
46 CYLINDER STORAGE
47 PIPING AREA
48 COUNTING ROOM
49 TOOL AND EQUIP. ROOM
50 SPECIAL DECON.
51 DECON. ROOM
52 EMERGENCY SHOWER AND EYE WASH
53 COLD LAB
54 CORRIDOR

RADWASTE

20 EXHAUST MONITOR
21 WASTE GAS DECAY TANK
22 WASTE GAS SURGE TANK
23 EVAPORATOR SURFACE CONDENSER
24 EVAPORATOR VAPOR BODY
25 WASTE GAS COMPRESSOR
26 H_2 AND O_2 ANALYZER

CONTROL

20 EXHAUST MONITOR
21 SUPERVISOR OFFICE
22 KITCHEN
23 PANTRY
24 WOMEN'S TOILET
25 MEN'S TOILET
26 CONFERENCE ROOM
27 COMPUTER ROOM
28 INSTRUMENT REPAIR ROOM
29 STOREROOM
30 CONTROL ROOM

NO	DESCRIPTION
1.	HATCH
2.	PIPE CHASE
3.	ELEC. CHASE
4.	HVAC. CHASE
5.	SUMP
6.	ELEVATOR
7.	MONORAIL
8.	VALVE GALLERY
9.	MCC
10.	LOAD CENTER
11.	SWITCHGEAR
12.	EQUIP. OR ACCESS OPENING
13.	TRANSMITTER RACK
14.	VOLTAGE REG.
15.	RELAY CAB.
16.	LIGHTING PANEL
17.	DISTRIBUTION PANEL
18.	JUNCTION BOX
19.	PIPE TRENCH

NOTE: DEPICTED ZONES DO NOT INCORPORATE SOURCE REDUCTIONS DUE TO DECAY OR SINGLE ESF TRAIN OPERATION. REFER TO FSAR SECTION 12.2 FOR DECAY CORRECTIONS. NORMAL OPERATION SOURCES ARE NOT INCLUDED IN THESE ZONES.

LEGEND:

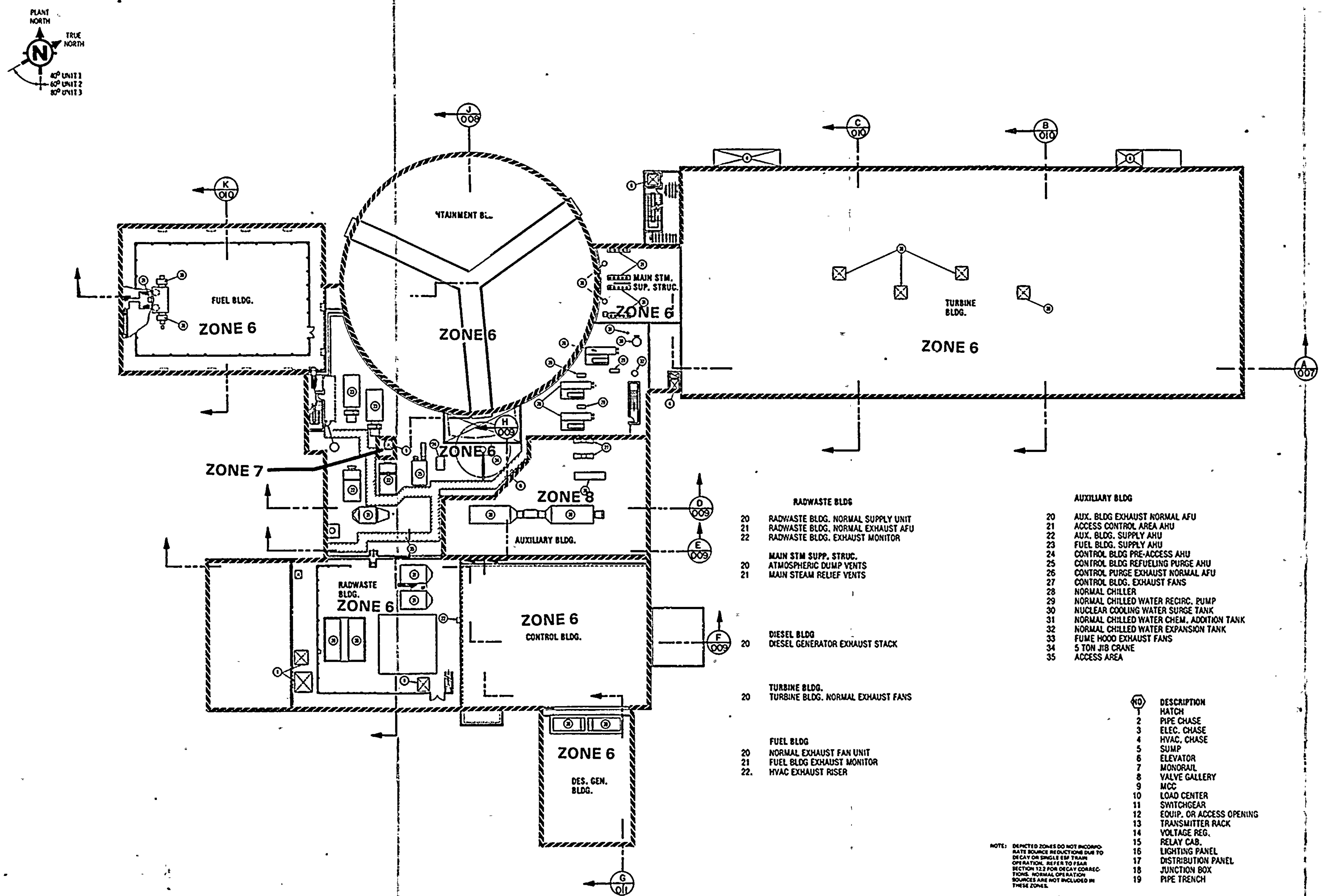
ZONE 1 $X < 0.5$ mR/hr
ZONE 2 $0.5 \leq X < 2.5$ mR/hr
ZONE 3 $2.5 \leq X < 15$ mR/hr
ZONE 4 $15 \leq X < 100$ mR/hr
ZONE 5 $100 \leq X < 1000$ mR/hr
ZONE 6 $1 \leq X < 5$ R/hr
ZONE 7 $5 \leq X < 10$ R/hr
ZONE 8 $10 \leq X < 100$ R/hr
ZONE 9 $100 \leq X$ R/hr

→ DENOTES UP U.N.O.
▨ INDICATES ONE-WAY DOOR OR CONTROLLED FROM INSIDE OF BLDG
G-GATE
△ INDICATES CONTROLLED ACCESS AND/OR ROLL UP DOOR

Palo Verde Nuclear Generating Station
LLIR

RADIATION ZONES - LOCA
DEGRADED CORE - INTACT PRIMARY
BETWEEN EL. 140'-0" & 200'-0"

Figure II.B.2-17



RADWASTE BLDG

20 RADWASTE BLDG. NORMAL SUPPLY UNIT
21 RADWASTE BLDG. NORMAL EXHAUST AFU
22 RADWASTE BLDG. EXHAUST MONITOR

MAIN STM SUPP. STRUC.

20 ATMOSPHERIC DUMP VENTS
21 MAIN STEAM RELIEF VENTS

DIESEL BLDG

20 DIESEL GENERATOR EXHAUST STACK

TURBINE BLDG.

20 TURBINE BLDG. NORMAL EXHAUST FANS

FUEL BLDG

20 NORMAL EXHAUST FAN UNIT
21 FUEL BLDG EXHAUST MONITOR
22. HVAC EXHAUST RISER

AUXILIARY BLDG

20 AUX. BLDG EXHAUST NORMAL AFU
21 ACCESS CONTROL AREA AHU
22 AUX. BLDG. SUPPLY AHU
23 FUEL BLDG. SUPPLY AHU
24 CONTROL BLDG PRE-ACCESS AHU
25 CONTROL BLDG REFUELING PURGE AHU
26 CONTROL PURGE EXHAUST NORMAL AFU
27 CONTROL BLDG. EXHAUST FANS
28 NORMAL CHILLER
29 NORMAL CHILLED WATER RECIRC. PUMP
30 NUCLEAR COOLING WATER SURGE TANK
31 NORMAL CHILLED WATER CHEM. ADDITION TANK
32 NORMAL CHILLED WATER EXPANSION TANK
33 FUME HOOD EXHAUST FANS
34 5 TON JIB CRANE
35 ACCESS AREA

NO	DESCRIPTION
1	HATCH
2	PIPE CHASE
3	ELEC. CHASE
4	HVAC. CHASE
5	SUMP
6	ELEVATOR
7	MONORAIL
8	VALVE GALLERY
9	MCC
10	LOAD CENTER
11	SWITCHGEAR
12	EQUIP. OR ACCESS OPENING
13	TRANSMITTER RACK
14	VOLTAGE REG.
15	RELAY CAB.
16	LIGHTING PANEL
17	DISTRIBUTION PANEL
18	JUNCTION BOX
19	PIPE TRENCH

NOTE: DEPICTED ZONES DO NOT INCORPORATE SOURCE REDUCTIONS DUE TO DECAY OR SINGLE ESF TRAIN OPERATION. REFER TO FSAR SECTION 12.2 FOR DECAY CORRECTIONS. NORMAL OPERATION SOURCES ARE NOT INCLUDED IN THESE ZONES.

— DENOTES UP U.N.O.
INDICATES ONE-WAY DOOR OR CONTROLLED FROM INSIDE OF BLDG
Ⓖ GATE
△ INDICATES CONTROLLED ACCESS AND/OR ROLL UP DOOR

LEGEND:

ZONE 1 $X < 0.5$ mR/hr
ZONE 2 $0.5 \leq X < 2.5$ mR/hr
ZONE 3 $2.5 \leq X < 15$ mR/hr
ZONE 4 $15 \leq X < 100$ mR/hr
ZONE 5 $100 \leq X < 1000$ mR/hr
ZONE 6 $1 \leq X < 5$ R/hr
ZONE 7 $5 \leq X < 10$ R/hr
ZONE 8 $10 \leq X < 100$ R/hr
ZONE 9 $100 \leq X$ R/hr

1

Palo Verde Nuclear Generating Station
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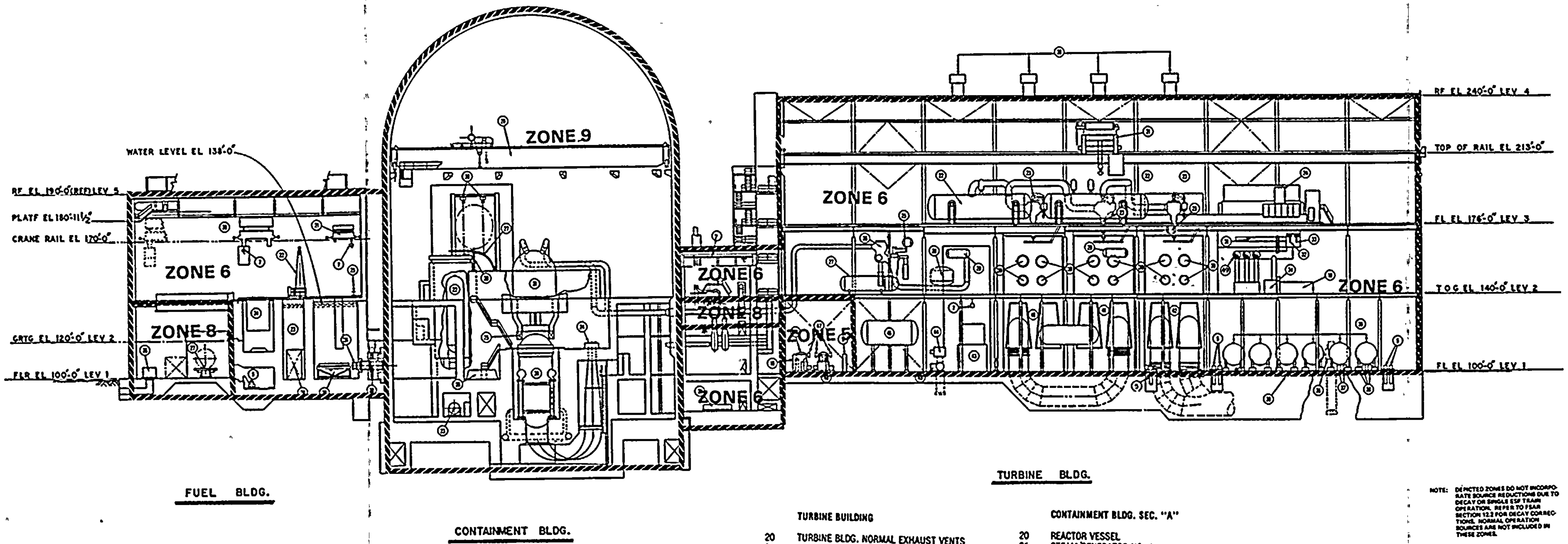
RADIATION ZONES - LOCA
DEGRADED CORE - INTACT PRIMARY
AT ROOF ELEVATION

Figure II.B.2-18



TURBINE BUILDING

20 TURBINE BLDG. NORMAL EXHAUST VENTS
21 280 TON CRANE
22 MSR MOISTURE SEPARATOR REHEATER
23 CONTROL INSULATION VALVES
24 GENERATOR
25 REHEATOR DRAIN TANK
26 MAIN STEAM STOP AND CONTROL VALVE
27 FEEDWATER HEATER
28 BLOWDOWN FLUSH TANK
29 MSR DRAIN TANK
30 L.P. FEEDWATER HEATERS
31 AUX. DETRAINING TANK
32 LINE TERMINAL
33 NEUTRAL TERMINAL
34 P.T. CUBICLE
35 CNDS POLISHING TANK
36 CONDENSATE PUMP
37 HYDRAZINE PUMPS
38 AMMONIA PUMPS
39 CNDS PUMP STRAINER
40 CONDENSER "A"
41 CONDENSER "B"
42 CONDENSER "C"
43 FEEDWATER PUMP TURBINE
44 HEATER DRAIN PUMP
45 FEEDWATER TUB. LUBE OIL CENTRIFUGE
46 HEATER DRAIN TANK
47 AFTER COOLER AND MOIST. SEPEX.
48 AIR RECEIVER
49 ISOPHASE BUS DUCT
50

CONTAINMENT BLDG. SEC. "A"

20 REACTOR VESSEL
21 STEAM GENERATOR NO. 1
22 PRESSURIZER
23 REACTOR DRAIN TANK
24 SEAL TABLE
25 R.V. HEAD COOLING SHROUD
26 CORE SUPPORT BARREL STAND
27 EQUIPMENT HATCH
28 JIB CRANE
29 POLAR CRANE
30 ELECTRIC HOIST

MAIN STEAM SUPPORT STRUCTURE

20 MAIN STEAM ISOLATION VALVE
21 AUX. FEED TURBINE PUMP

FUEL BUILDING

20 150 TON CASK HANDLING CRANE
21 10 TON FUEL HANDLING CRANE
22 SPENT FUEL HANDLING MACHINE
23 SPENT FUEL POOL
24 DECON. PIT
25 TRANSFER TUBE VALVE HANDWHEEL
26 EXHAUST A.F.U.
27 RAILROAD CARS
28 SPENT FUEL RACKS
29 FUEL CARRIAGE AND UPENDER
30 FUEL TRANSFER TUBE
31 TRANSFER TUBE VALVE
32

SECTION A
002
003
004
005
006

NOTE: DEPICTED ZONES DO NOT INCORPORATE SOURCE REDUCTIONS DUE TO DECAY OR SINGLE ESF TRAIN OPERATION. REFER TO FSAR SECTION 12.2 FOR DECAY CORRECTIONS. NORMAL OPERATION SOURCES ARE NOT INCLUDED IN THESE ZONES.

NO GENERAL DESCRIPTION

1 HATCH
2 PIPE CHASE
3 ELEC. CHASE
4 HVAC. CHASE
5 SUMP
6 ELEVATOR
7 MONORAIL
8 VALVE GALLERY
9 MCC
10 LOAD CENTER
11 SWITCHGEAR
12 EQUIP. OR ACCESS OPENING
13 TRANSMITTER RACK
14 VOLTAGE REG.
15 RELAY CAB.
16 LIGHTING PANEL
17 DISTRIBUTION PANEL
18 JUNCTION BOX
19 PIPE TRENCH

DENOTES UP U.N.O.
INDICATES ONE-WAY DOOR OR CONTROLLED FROM INSIDE OF BLDG
GATE
△ INDICATES CONTROLLED ACCESS AND/OR ROLL UP DOOR

LEGEND:
ZONE 1 $X < 0.5$ mR/hr
ZONE 2 $0.5 \leq X < 2.5$ mR/hr
ZONE 3 $2.5 \leq X < 15$ mR/hr
ZONE 4 $15 \leq X < 100$ mR/hr
ZONE 5 $100 \leq X < 1000$ mR/hr
ZONE 6 $1 \leq X < 5$ R/hr
ZONE 7 $5 \leq X < 10$ R/hr
ZONE 8 $10 \leq X < 100$ R/hr
ZONE 9 $100 \leq X$ R/hr

1

Palo Verde Nuclear Generating Station
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RADIATION ZONES - LOCA
DEGRADED CORE - INTACT PRIMARY
SECTION A

Figure II.B.2-19

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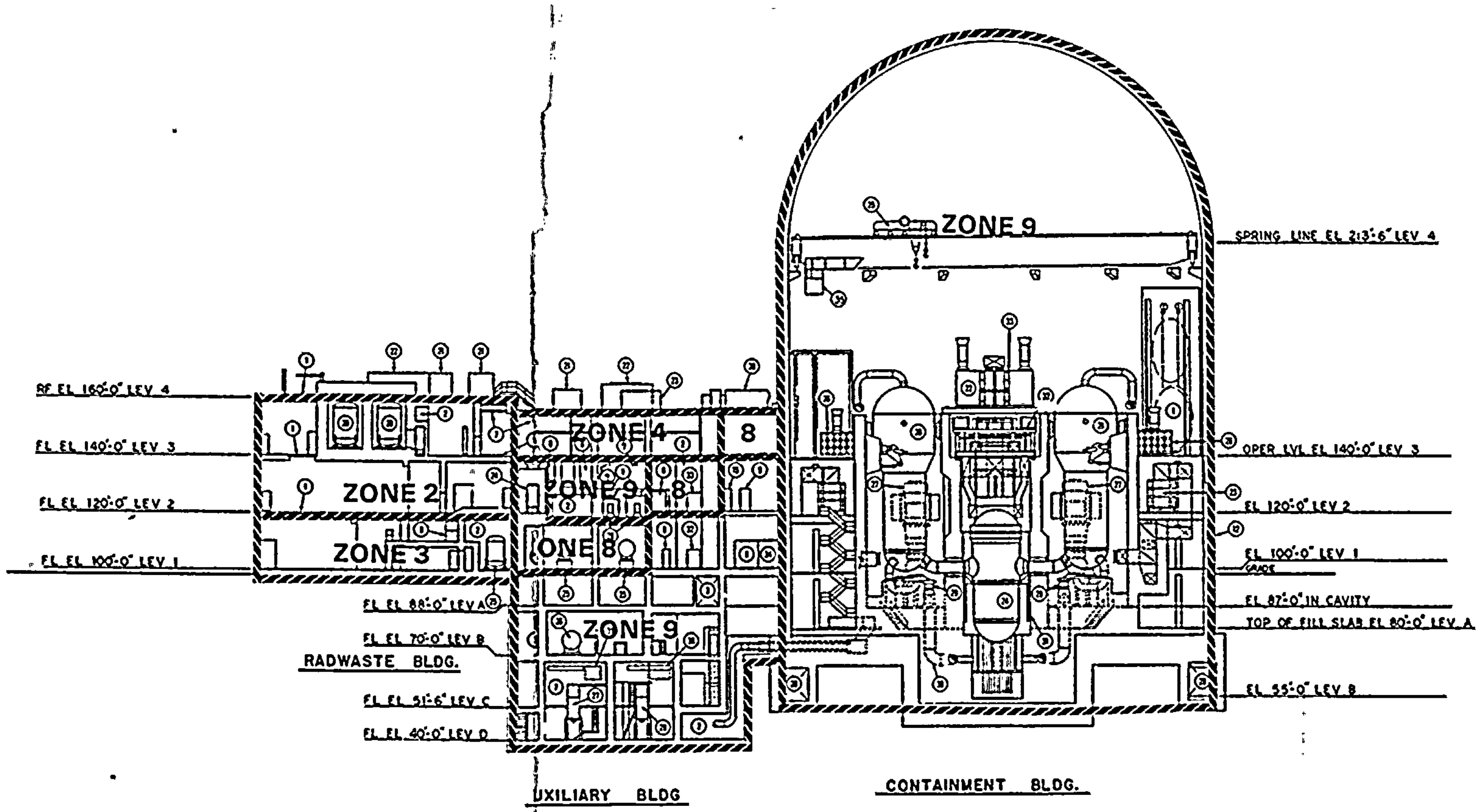

SECTION J
002
003
004
005
006

NOTE: DEPICTED ZONES DO NOT INCORPORATE SOURCE REDUCTIONS DUE TO DECAY OR SINGLE ESF TRAIN OPERATION. REFER TO FSAR SECTION 12.2 FOR DECAY CORRECTIONS. NORMAL OPERATION SOURCES ARE NOT INCLUDED IN THESE ZONES.

NO	GENERAL DESCRIPTION
1	HATCH
2	PIPE CHASE
3	ELEC. CHASE
4	HVAC. CHASE
5	SUMP
6	ELEVATOR
7	MONORAIL
8	VALVE GALLERY
9	MCC
10	LOAD CENTER
11	SWITCHGEAR
12	EQUIP. OR ACCESS OPENING
13	TRANSMITTER RACK
14	VOLTAGE REG.
15	RELAY CAB.
16	LIGHTING PANEL
17	DISTRIBUTION PANEL
18	JUNCTION BOX
19	PIPE TRENCH

— DENOTES UP U.N.O.
▪ INDICATES ONE-WAY DOOR OR CONTROLLED FROM INSIDE OF BLDG
Ⓖ GATE
△ INDICATES CONT. ACCESS &/OR ROLL UP DOORS

RADWASTE BLDG.

20	WASTE GAS DECAY TANK
21	RADWASTE BLDG. NORMAL EXCH. AHU
22	RADWASTE BLDG. NORMAL SUPP. AHU
23	
24	
25	LOW ACTIVITY SPENT RESIN TANK

AUXILIARY BLDG

20	FUEL BLDG. SUPPLY AHU
21	AUX. BLDG. ACCESS CONTROL AHU
22	AUX. BLDG. SUPPLY AHU
23	CONT. BLDG REFUELING PURGE AHU
24	PRE-HOLDUP ION EXCH.
25	ECW HEAT EXCHANGER
26	SHUTDOWN COOLING HEAT EXCHANGER
27	LPSI PUMP
28	FILTERS
29	CONT SPRAY PUMP
30	AUX. BLDG CS PUMP ROOM ESSENTIAL ACU
31	AUX. BLDG. LPSI PUMP ROOM ESSENTIAL ACU
32	POST LOCA ANALYZER
33	120 V.A.C. DIST. PANEL
34	LOAD CENTER

CONTAINMENT

20	STEAM GENERATOR NO. 2
21	STEAM GENERATOR NO. 1
22	CONTAINMENT CEDM NORMAL ACU
23	NORMAL A.C.U.
24	REACTOR VESSEL
25	POLAR CRANE
26	CONTAIN. PRE-ACCESS NORMAL AFU
27	R.C. PUMP
28	TENDON GALLERY
29	CAVITY COOLING FAN
30	CAVITY COOLING DUCT
31	EX. CORE NEUTRON DETECTORS
32	C.E.D.M.
33	CEDM NORMAL AHU AND FAN
34	EMERGENCY SUMP SCREEN
35	CAB

LEGEND:

ZONE 1 $X < 0.5$ mR/hr
ZONE 2 $0.5 \leq X < 2.5$ mR/hr
ZONE 3 $2.5 \leq X < 15$ mR/hr
ZONE 4 $15 \leq X < 100$ mR/hr
ZONE 5 $100 \leq X < 1000$ mR/hr
ZONE 6 $1 \leq X < 5$ R/hr
ZONE 7 $5 \leq X < 10$ R/hr
ZONE 8 $10 \leq X < 100$ R/hr
ZONE 9 $100 \leq X$ R/hr

1

Palo Verde Nuclear Generating Station
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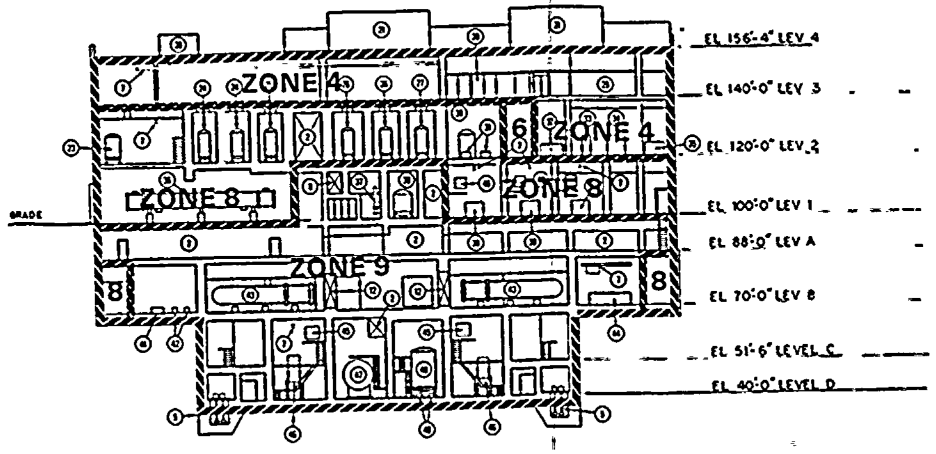
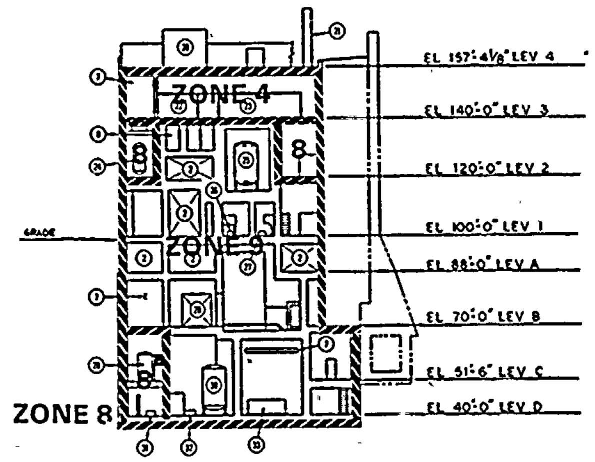
RADIATION ZONES - LOCA
DEGRADED CORE - INTACT PRIMARY
SECTION J-J

Figure II.B.2-20



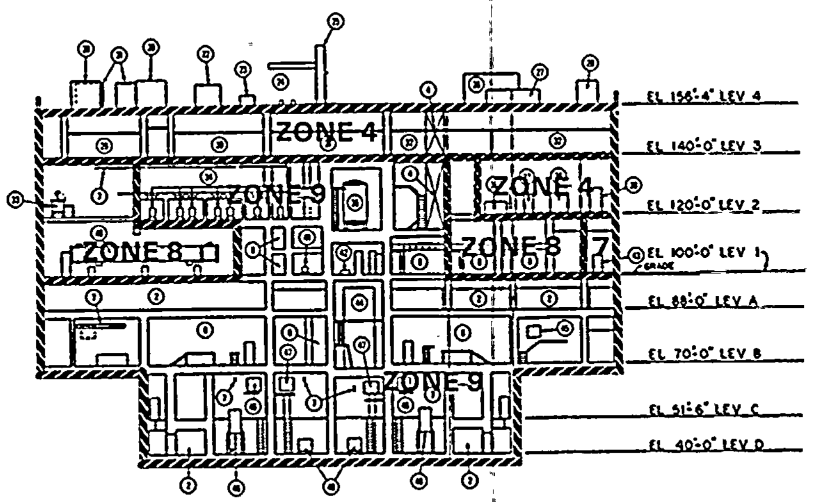
AUXILIARY BLDG.

SECTION E 002 003 004 005 006



AUXILIARY BLDG.

SECTION H 002 003 004 005 006



AUXILIARY BLDG.

SECTION D 002 003 004 005 006

NOTE: DEPICTED ZONES DO NOT INCORPORATE SOURCE REDUCTIONS DUE TO DECAY OR SINGLE ESF TRAIN OPERATION. REFER TO FSAR SECTION 12.2 FOR DECAY CORRECTIONS. NORMAL OPERATION SOURCES ARE NOT INCLUDED IN THESE ZONES.

AUXILIARY BUILDING SEC. "H"

20 AUX BLDG. EXHAUST A.F.U.
21 5 TON JIB CRANE
22 HEALTH PHYSICS
23 COLD LAB
24 DEBORATING ION EXCHANGER
25 VOLUME CONTROL TANK
26 CRUD PUMP
27 GAS STRIPPER
28 BLOCK OUT EFFLUENT RAD. MONITOR
29 CHEM. DRAIN TANK
30 COOLING WATER HOLD UP TANK
31 CHEM. DRAIN TANK PUMP
32 COOLING WATER HOLDUP TK PUMP
33 HPSI PUMP

AUXILIARY BUILDING SEC. "E"

20 AUX. BUILD. ACCESS CONT. AHU
21 AUX. BLDG. EXHAUST A.F.U.
23 ECW SURGE TANK
24 FUEL POOL ION EXCHANGE
25 PRE HOLDUP ION EXCHANGE
26 PURIFICATION ION EXCHANGE
27 DEBORATION ION EXCHANGE
28 WOMENS LOCKER ROOM
29 LUNCH ROOM
30 SPRAY CHEM. STORAGE TANK
31 SPRAY CHEM. STOR. PUMPS
32 MG SET
33 CONTROL PANEL
34 CEDM CONTR. CAB
35 CEDM CONTR. CAB R.M. A.C.U.
36 ECW HEAT EXCHANGE
37 SEAL INJECTION Hx
38 CRUD TANK
39 CHARGING PUMPS
40 A.C.U.
41 BORIC ACID MAKE UP PUMPS
42 REACTOR MAKE UP WATER PUMPS
43 SHUTDOWN COOLING Hx
44 ECW PUMP
45 LPSI PUMP R.M. A.C.U.
46 LPSI PUMPS
47 EQUIP. DRAIN TANK
48 COOLING WATER HOLD UP TANK
49 COOLING WATER HOLD UP TANK PUMPS
50

AUXILIARY BUILDING SEC. "D"

20 AUX BLDG. SUPPLY AHU
21 FUEL BLDG. SUPPLY AHU
22 CONT. BLDG. REFUELING PURGE AHU
23 CONT. BLDG. PRE-ACCESS AHU
24 FUME HOOD EXHAUST FANS
25 5 TON JIB CRANE
26 NORMAL CHILLER
27 CONT. PURGE EXHAUST AFU
28 STAIR WAY
29 COUNTING ROOM
30 HOT LAB
31 COLD LAB
32 MEN LOCKERS
33 BORIC ACID BATCHING TANK
34 FILTERS
35 VOLUME CONTROL TK
36 MG SET
37 CONTROL CAB
38 CEDM CONTR. CAB
39 CEDM CONTR. R.M. A.C.U.
40 ECW Hx
41 LET DOWN Hx
42 CRUD PUMP
43 H_2 RECOMBINER CONT. PANEL
44 GAS STRIPPER
45 ECW PUMP ROOM A.C.U.
46 CONT. SPRAY PUMP R.M. A.C.U.
47 HPSI PUMP RM. A.C.U.
48 CONT SPRAY PUMP
49 HPSI PUMP
50

(NO) GENERAL DESCRIPTION

1 HATCH
2 PIPE CHASE
3 ELEC. CHASE
4 HVAC. CHASE
5 SUMP
6 ELEVATOR
7 MONORAIL
8 VALVE GALLERY
9 MCC
10 LOAD CENTER
11 SWITCHGEAR
12 EQUIP. OR ACCESS OPENING
13 TRANSMITTER RACK
14 VOLTAGE REG.
15 RELAY CAB.
16 LIGHTING PANEL
17 DISTRIBUTION PANEL
18 JUNCTION BOX
19 PIPE TRENCH

—— DENOTES UP U.N.O.
▪ INDICATES ONE-WAY DOOR OR CONTROLLED FROM INSIDE OF BLDG
G-GATE
△ INDICATES CONTROLLED ACCESS &/OR ROLL-UP DOORS

LEGEND:

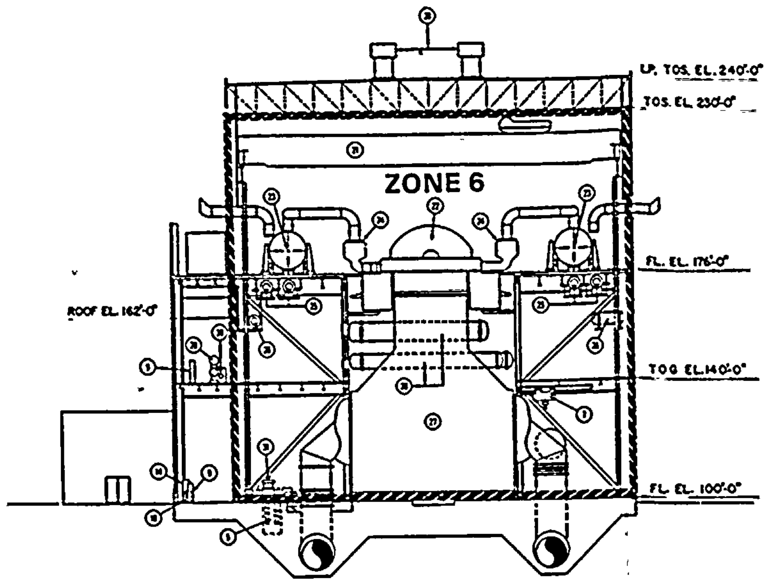
ZONE 1 $X < 0.5$ mR/hr
ZONE 2 $0.5 \leqslant X < 2.5$ mR/hr
ZONE 3 $2.5 \leqslant X < 15$ mR/hr
ZONE 4 $15 \leqslant X < 100$ mR/hr
ZONE 5 $100 \leqslant X < 1000$ mR/hr
ZONE 6 $1 \leqslant X < 5$ R/hr
ZONE 7 $5 \leqslant X < 10$ R/hr
ZONE 8 $10 \leqslant X < 100$ R/hr
ZONE 9 $100 \leqslant X$ R/hr

1

Palo Verde Nuclear Generating Station
LLIR

RADIATION ZONES - LOCA
DEGRADED CORE - INTACT PRIMARY
SECTIONS D, E, H

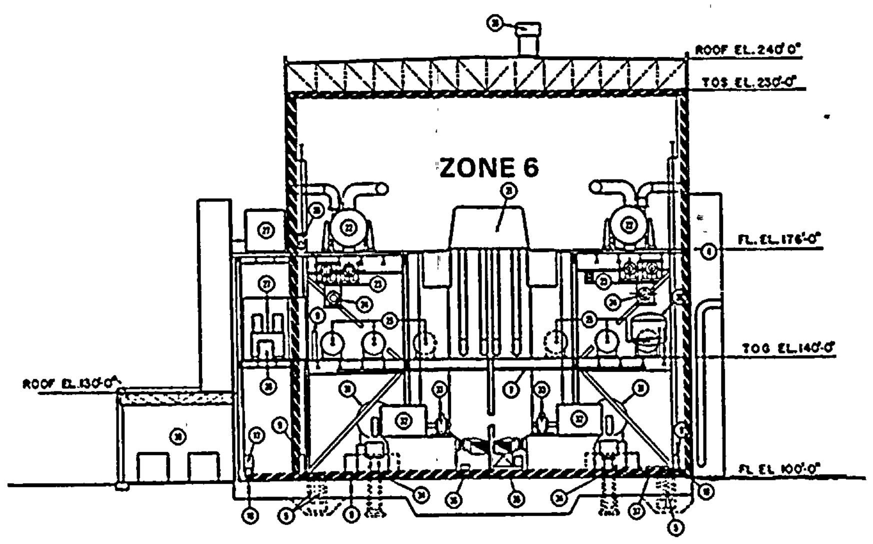
Figure II.B.2-21



TURBINE BLDG.

SECTION B

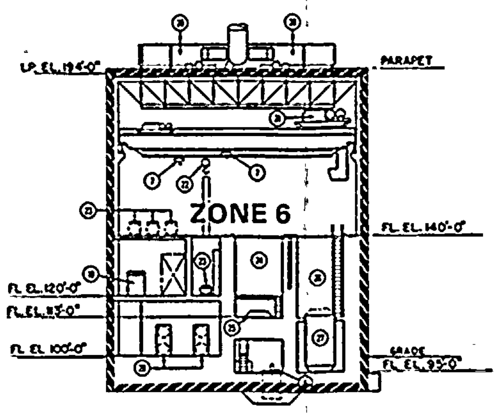
003
004
005
006



TURBINE BLDG.

SECTION C

003
004
005
006



FUEL BLDG.

SECTION K

003
004
005
006

EQUIPMENT LIST FUEL BUILD. SEC. "K"

20 NORMAL EXHAUST FANS
21 150 TON CRANE
22 10 TON NEW FUEL CRANE
23 NEW FUEL CONTAINERS
24 DECON PIT
25 CASK LEAD LAYDOWN AREA
26 LOAD PIT
27 CASK
28 SPENT FUEL POOL HX

EQUIPMENT LIST TURBINE BUILD. SEC. "B"

20 NORMAL EXHAUST FANS
21 280 TON CRANE
22 LOW PRESS. TURBINE
23 MOISTURE SEPARATOR REHEATOR
24 CONTROL INTERCEPT VALVE
25 REHEATER DRAIN TANKS
26 MSR DRAIN TANKS
27 CONDENSER "C"
28 L.P. FEED WATER HEATERS
29 STEAM SEAL PKG EXHAUSTER
30 MAIN CONDENSER EVAC. SYS. FILTER PKG
31 VACUUM PUMP

EQUIPMENT LIST TURBINE SECTION "C"

20 NORMAL EXHAUST FANS
21 H.P. TURBINE
22 MOISTURE SEPARATOR REHEATOR
23 REHEATER DRAIN TANKS
24 M.S.R. DRAIN TANK
25 FEEDWATER HEATERS
26 BLOW DOWN FLASH TANK
27 NORMAL AIR HANDLING UNIT
28 MAIN TURBINE L.O. CENTRIFUGE
29 MAIN TURBINE L.O. MIST ELIMINATOR
30 SWITCHGEAR ROOM
31 HEATER DRAIN TANK
32 FEEDWATER PUMP TURBINE
33 FEEDWATER PUMP
34 FEEDWATER DRAIN PUMP
35 EHC POWER UNIT
36 F.W. PUMP TURBINE L.O. CENTRIFUGE
37 AUX. F.W. PUMP
38 NORMAL A.H.U.

FUEL OIL STORAGE TANK

20 F.O. PUMP
21 STORAGE TANK

NOTE: DEPICTED ZONES DO NOT INCORPORATE SOURCE REDUCTIONS DUE TO DECAY OR SINGLE ESF TRAIN OPERATION. REFER TO FSAR SECTION 12.2 FOR DECAY CORRECTIONS. NORMAL OPERATION SOURCES ARE NOT INCLUDED IN THESE ZONES.

GENERAL

NO	DESCRIPTION
1	HATCH
2	PIPE CHASE
3	ELEC. CHASE
4	HVAC. CHASE
5	SUMP
6	ELEVATOR
7	MONORAIL
8	VALVE GALLERY
9	MCC
10	LOAD CENTER
11	SWITCHGEAR
12	EQUIP. OR ACCESS OPENING
13	TRANSMITTER RACK
14	VOLTAGE REG.
15	RELAY CAB.
16	LIGHTING PANEL
17	DISTRIBUTION PANEL
18	JUNCTION BOX
19	PIPE TRENCH

— DENOTES UP U.N.O.

■ INDICATES ONE-WAY DOOR OR CONTROLLED FROM INSIDE OF BLDG

Ⓖ GATE

△ INDICATES CONTROLLED ACCESS &/OR ROLL-UP DOORS

1

LEGEND:

ZONE 1 $X < 0.5$ mR/hr
ZONE 2 $0.5 \leq X < 2.5$ mR/hr
ZONE 3 $2.5 \leq X < 15$ mR/hr
ZONE 4 $15 \leq X < 100$ mR/hr
ZONE 5 $100 \leq X < 1000$ mR/hr
ZONE 6 $1 \leq X < 5$ R/hr
ZONE 7 $5 \leq X < 10$ R/hr
ZONE 8 $10 \leq X < 100$ R/hr
ZONE 9 $100 \leq X$ R/hr

Palo Verde Nuclear Generating Station
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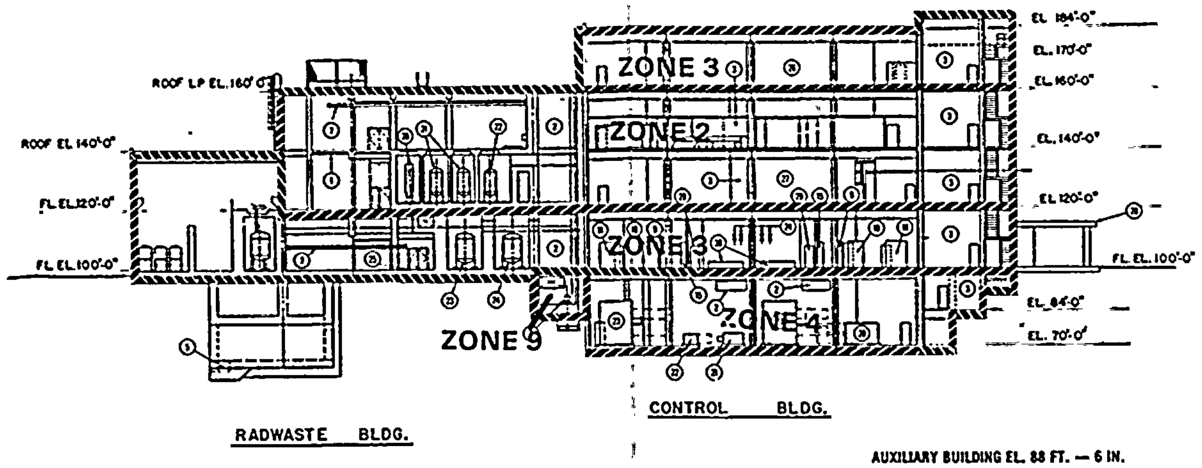
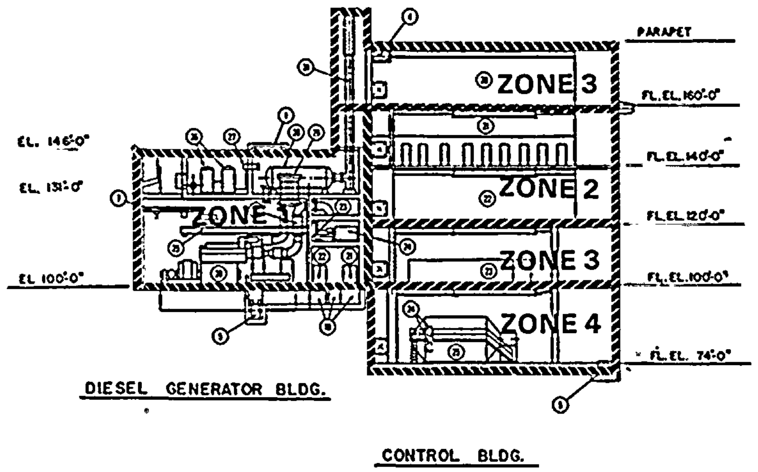
RADIATION ZONES - LOCA
DEGRADED CORE - INTACT PRIMARY
SECTIONS B, C, K

Figure II.B.2-22





AUXILIARY BUILDING EL. 88 FT. — 6 IN.

19
20 GAS STRIPPER
21 NONNUCLEAR SAMPLE COOLERS
22 FLOOR GRATING AND FIRE DAMPER
23 PIPEWAY "TRAIN "A" "
24 PIPEWAY TRAIN "B"

CONTROL BUILDING EL. 160 FT. — 0 IN.

20 SPRINKLING VALVE STATION
21 CHLORINE DETECTOR
22 DIESEL GEN. EXHAUST STACKS
23 H.V.A.C. OUTSIDE AIR INTAKE PLENUM
24 NORMAL SMOKE EXHAUST FAN
25 OUTSIDE AIR CHASE
26 GASEOUS RADIATION MONITOR
27 BATTERY ROOM

DIESEL GENERATOR SEC. "E"

19
20 DIESEL GENERATOR DGB-H01
21 LOW VOLTAGE CON.
22 HIGH VOLTAGE CON.
23 H.V.A.C. CONTROL POWER
24 DIESEL GENERATOR CONTROL RM.
25 5 TON CRANE
26 STARTING AIR PACKAGE
27 VENT FAN
28 EXHAUST SILENCER
29 DIESEL EXHAUST FAN
30 INTAKE AIR SILENCER
31 DIESEL EXHAUST STACK

CONTROL BUILDING SEC. "G"

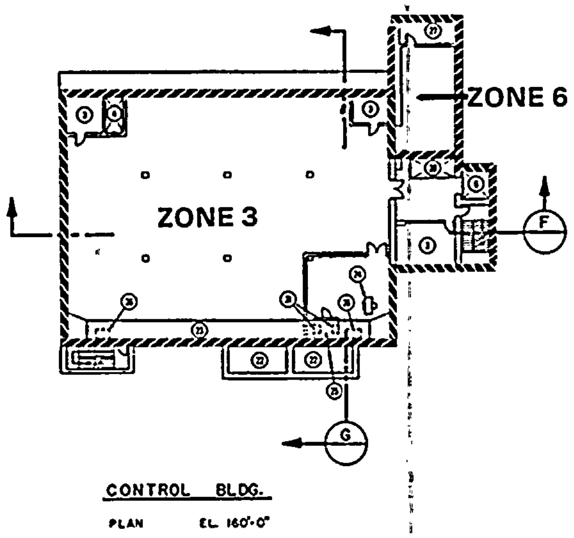
20 UPPER CABLE SPREADING ROOM
21 MAIN CONTROL ROOM
22 LOWER CABLE SPREADING ROOM
23 4.16 KV SWGR
24 COOLING COIL
25 ESSENTIAL A.H.V.

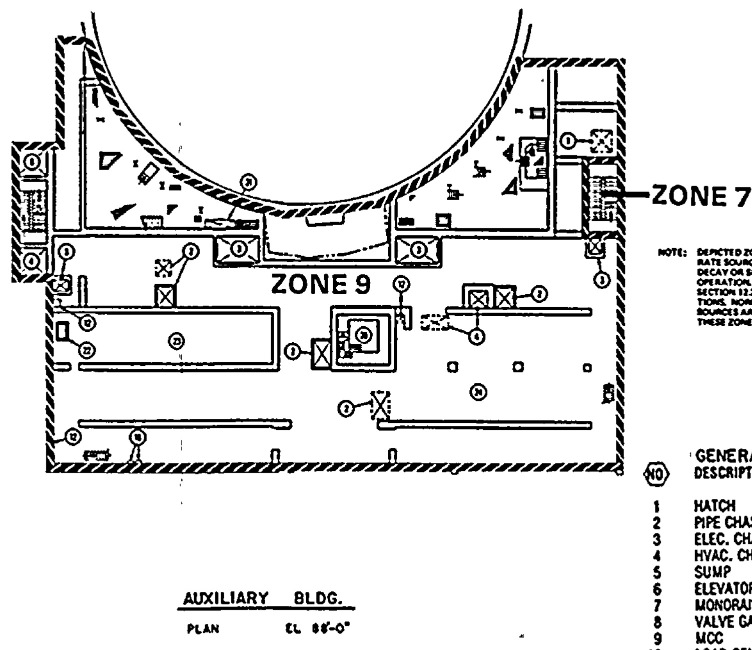
RADWASTE BUILDING SEC. "F"

20 B.A.C. ION EXCHANGER
21 L.R.S. ION EXCHANGER
22 L.R.S. ABSORPTION BED
23 HIGH ACTIVITY SPENT RESIN TANK
24 LOW ACTIVITY SPENT RESIN TANK
25 DECON BOOTH

CONTROL BUILDING SEC. "F"

20
21 CONTROL BLDG. OUTSIDE AIR, AIR WASHER UNIT
22 CONTROL BLDG. ESF SWGR A.H.V.
23 CONTROL RM. ESSENTIAL A.H.V.
24 EXHAUST FANS
25 MAIN CONTROL ROOM
26 UPPER CABLE SPREADING RM.
27 LOWER CABLE SPREADING RM.
28 CANOPY ROOF.
29 EMERGENCY EYE WASH AND ABOVE
30 BATTERIES





NOTE: DEPICTED ZONES DO NOT INCORPORATE SOURCE REDUCTIONS DUE TO DECAY OR SINGLE ESF TRAIN OPERATION. REFER TO FSAR SECTION 12.2 FOR DECAY CORRECTIONS. NORMAL OPERATION SOURCES ARE NOT INCLUDED IN THESE ZONES.

NO	GENERAL DESCRIPTION
1	HATCH
2	PIPE CHASE
3	ELEC. CHASE
4	HVAC. CHASE
5	SUMP
6	ELEVATOR
7	MONORAIL
8	VALVE GALLERY
9	MCC
10	LOAD CENTER
11	SWITCHGEAR
12	EQUIP. OR ACCESS OPENING
13	TRANSMITTER RACK
14	VOLTAGE REG.
15	RELAY CAB.
16	LIGHTING PANEL
17	DISTRIBUTION PANEL
18	JUNCTION BOX
19	PIPE TRENCH

DENOTES UP U.N.O.
INDICATES ONE-WAY DOOR OR CONTROLLED FROM INSIDE OF BLDG
GATE
INDICATES CONT. ACCESS &/OR ROLLUP DRS.

LEGEND:
ZONE 1 $X < 0.5$ mR/hr
ZONE 2 $0.5 \leq X < 2.5$ mR/hr
ZONE 3 $2.5 \leq X < 15$ mR/hr
ZONE 4 $15 \leq X < 100$ mR/hr
ZONE 5 $100 \leq X < 1000$ mR/hr
ZONE 6 $1 \leq X < 5$ R/hr
ZONE 7 $5 \leq X < 10$ R/hr
ZONE 8 $10 \leq X < 100$ R/hr
ZONE 9 $100 \leq X$ R/hr

1

Palo Verde Nuclear Generating Station
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RADIATION ZONES - LOCA
DEGRADED CORE - INTACT PRIMARY
PLAN AT AUX. 88', CONTROL 160',
SECTS. F, G
Figure II.B.2-23

II.B.3 POSTACCIDENT SAMPLING CAPABILITY

Position

A design and operational review of the reactor coolant and containment atmosphere sampling line 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 rem 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 (in less than 2 hours) certain radionuclides 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 nonvolatile isotopes (which indicate fuel melting). The initial reactor coolant spectrum should correspond to a Regulatory Guide 1.3 or 1.4 release. The review should also consider the effects of direct radiation from piping and components in the auxiliary building and possible contamination and direct radiation from 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

FSAR section 9.3 has a detailed description of the PVNGS nuclear sampling system. This includes additions to perform functions specifically required by NUREG 0737 and Reg Guide 1.97, Rev 2. These additions will be complete prior to fuel load.

1 A review of the reactor coolant and containment atmosphere sampling systems and the radiological spectrum and chemical facilities has been conducted. Plant modifications are being made to permit personnel to obtain samples within 1 hour after an accident (without incurring an exposure to an individual

in excess of three rem whole-body or 18 3/4 rem to the extremities), to analyze samples within 2 hours for radioactive noble gases, iodines, cesium and non-volatile isotopes, and 1
to analyze samples within 1 hour for pH, boron, chlorides, dissolved oxygen, gaseous oxygen, and dissolved hydrogen. These 2
plant modifications, collectively referred to as the post accident sampling system (PASS) are illustrated in figures II.B.3-1 and II.B.3-2. Procedures will be developed for obtaining and analyzing these samples. 1

Design Capability

Modification of currently existing plant facilities allows plant personnel to obtain samples, in less than 1 hour, of pressurized and depressurized reactor coolant, safety injection, containment air, ESF and radwaste containment sumps, auxiliary 2
building sumps and volume control tank.

In-line analyses with automatic sequencing and computer control and readout will be backed up by a grab sample capability using manual controls. In-line analyses will include isotopic, gross gamma, pH, boron, dissolved hydrogen, chloride, dissolved oxygen and gaseous oxygen. Gaseous hydrogen will be sampled by the 1
containment hydrogen control system.

The onsite facility design and procedures provide the following capabilities:

A. Containment air may be sampled under positive or negative pressures.

B. Provisions are made to purge sample lines, for reducing plate-out, to insure proper mixing, for minimizing leakage, preventing blockage, to back-flush, to blowdown, for appropriate sample disposal, minimizing crud traps, and for passive flow restrictions (when sampling high pressure systems).

C. The PASS sampling lines and components conform to Quality Group D requirements. Isolation valves are used with appropriate automatic closure signals where PASS piping interconnects with higher quality group classification piping.

D. Provisions are included to measure a wide range of isotopes
1 for liquids from 10^{-3} μCi/ml to 10^{+7} μCi/ml and for gases from 10^{-7} μCi/ml to 10^{+5} μCi/ml. Sample dilution is available for grab samples but is not required for in-line analysis.

E. Provisions are included to measure dissolved hydrogen from 0 to 2000 cc/kg.

F. Provisions are included to restrict background levels of radiation at the PASS sample area such that the sample analysis provides results with an acceptably small error (approximately a factor of 2).

G. Provisions are included to facilitate plant procedures to identify the analyses required, measurement techniques and background level reduction methods.

H. Chemical analysis capability is provided which can meet all requirements when exposed to the specified source term.

I. Provisions are included to be able to draw a grab sample while maintaining ALARA radiation exposures and not exceeding General Design Criteria 19.

1. Shielding calculations were performed using source term specified in NUREG 0737 and Regulatory Guide 1.97 Rev 2 (Refer to section II.B.2).

2. A result of the shielding review, piping used for 1
backup grab sampling in the hot lab sample room area will be lead wrapped to keep operator doses ALARA.

3. High range portable and fixed survey instruments and personnel dosimeters are provided to permit rapid assessment of exposure rates and accumulated personnel exposure.

J. Design parameters are based on the full range of design pressure of the reactor coolant system and containment.

K. Testing will be conducted to demonstrate the ability to 2
obtain and analyze a sample by in-line and grab sample 1
techniques.

1 SAMPLE INLET PIPING

2 Samples can be drawn from the following nine points in each unit:

A. RCS hot leg - north side of RCS

B. Inlet to letdown system - south side of RCS

C. Safety injection mini flow line - Train A

D. Safety injection mini flow line - Train B

E. Containment recirculation sump

F. Containment radwaste sump

G. Auxiliary building sumps

H. Volume control tank

1 I. Containment atmosphere via the containment hydrogen control system

Isolation valves provide a quality and material classification break between the sample points and the PASS.

Sample Return Piping

Samples are returned to the following three points in each unit:

A. Containment Atmosphere (via the containment hydrogen control system) for gas samples only

B. Reactor drain tank - for post accident liquid samples

C. Equipment drain tank - for normal operation liquid samples

Isolation valves provide a quality and material classification 1
break between the sample return points and the PASS.

Sample Station

The PASS is a modular system. The valve module including pumps, 2
heat exchangers, valves and other sources of possible leakage 1
are located in the piping penetration room. The radiological 2
and chemical module is located in the auxiliary building. This
contains the in-line isotopic and chemical analysis equipment. 1
The two modules are connected via short pipes through a shield
wall that separates the remainder of the auxiliary building 2
from the piping penetration room.

This method of separation provides for:

A. Ease of routing sample piping to and from the PASS

B. Isolation of any leakage during normal and post accident operation 1

C. Ability to repair and maintain the spectroscopic and chemical instruments during normal operation

D. Meeting space requirements in an existing building.

The valve module provides the ability to draw, purge, cool, degas, circulate, dilute, and route samples in the PASS.

The radiological and chemical module provides isotopic, gross
gamma, pH, chloride, boron, dissolved oxygen and gaseous
1 oxygen analysis. (Hydrogen analysis is provided by the contain-
ment hydrogen control system.)

Isotopic analysis is provided with a collimated high-purity
germanium HP(Ge) detector with a 30-day supply of liquid
nitrogen (LN_2). The HP(Ge) detectors may operate under
2 unlimited temperature cycling between room and LN_2 temperatures
without performance degradation. The LN_2 storage container will
be supplied with a roller base to aid in refilling from a con-
venient location or container change-out.

The entire sampling sequence is controlled by a valve sequenc-
ing controller which in turn is run by a pulse height analyzer
(PHA) with extended capabilities. The PHA is operated by a
technician.

1 Grab samples are available from the hot lab in either a depres-
surized, diluted or pressurized sample device. Instrumentation
is provided for pressure, temperature and flow measurement.
The radiological and chemical module is provided with sufficient
2 shielding to maintain radiation exposure criteria ALARA during
normal and post accident operation.

1 Gas samples will be followed by a nitrogen purge to flush out
remaining gases.

Liquid samples will be followed by a demineralized water purge followed by a nitrogen gas purge to flush out remaining fluids. Nuclear cooling water will be provided for sample heat exchangers.

Instrument air will be provided for pneumatically controlled valves.

Control Panel

The PASS Control Panel is part of the hot lab PHA computer system. This is located in a shielded room, adjacent to the hot lab. This includes: 1

A. The valve control assembly with manual switching capability for all remote valves, pumps, and instruments

B. Recording devices

C. Mini status board

D. Microcomputer with CRT display

E. In-line instrumentation readout

PASS Operation

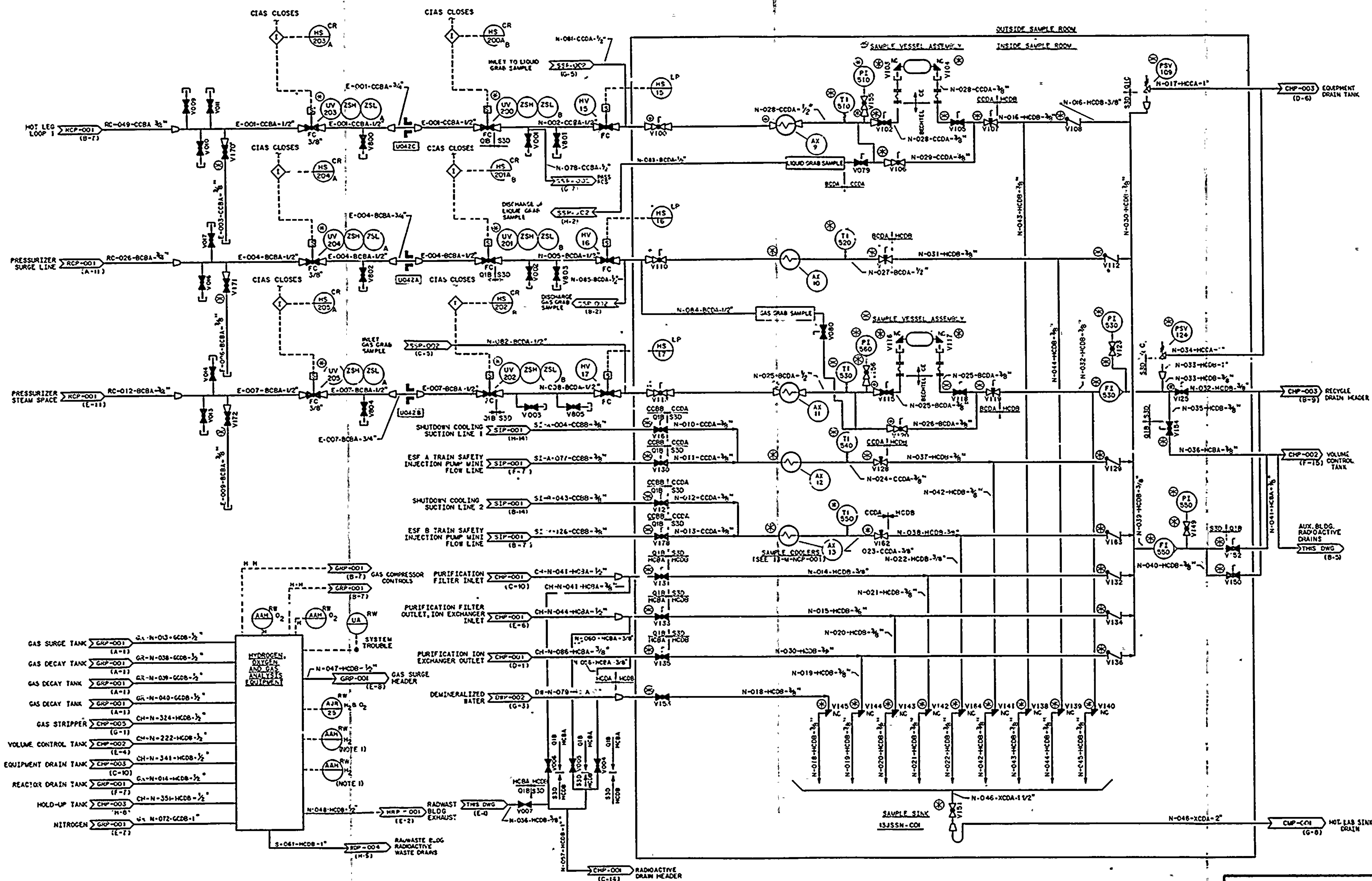
The operation of the system requires communication between 2
the chemistry technician in the hot lab and operators in the
control room. Prior to sampling a specific point, the hot 1
lab chemistry technician verifies with the control room 2
operator to ensure that the system isolation valve is open.
This may involve overriding a CIAS to re-open a valve. The 1
hot lab chemistry technician will then initialize the PHA 2

computer and perform radiological and chemical analysis, directing the sample to a predetermined discharge point. Results will be available on the PHA CRT and on the recorder printout.

Meeting Requirements

1

The process sampling system discussed in FSAR section 9.3.2 reflects the addition of the post accident sampling system. This fulfills the NRC requirements as outlined in NUREG 0578, NUREG 0737, and Reg Guide 1.97, Rev 2.



NOTE: 1. HYDROGEN ALARMS TO BE USED ONLY WHEN SAMPLING HOLD-UP TANK.

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

NUCLEAR SAFETY RELATED

1

13-N-SSP-001 Rev. 6

Palo Verde Nuclear Generating Station LLIR

NUCLEAR SAMPLING SYSTEM
Figure II.B.3-1

August 1981 Amendment 1

[illegible]

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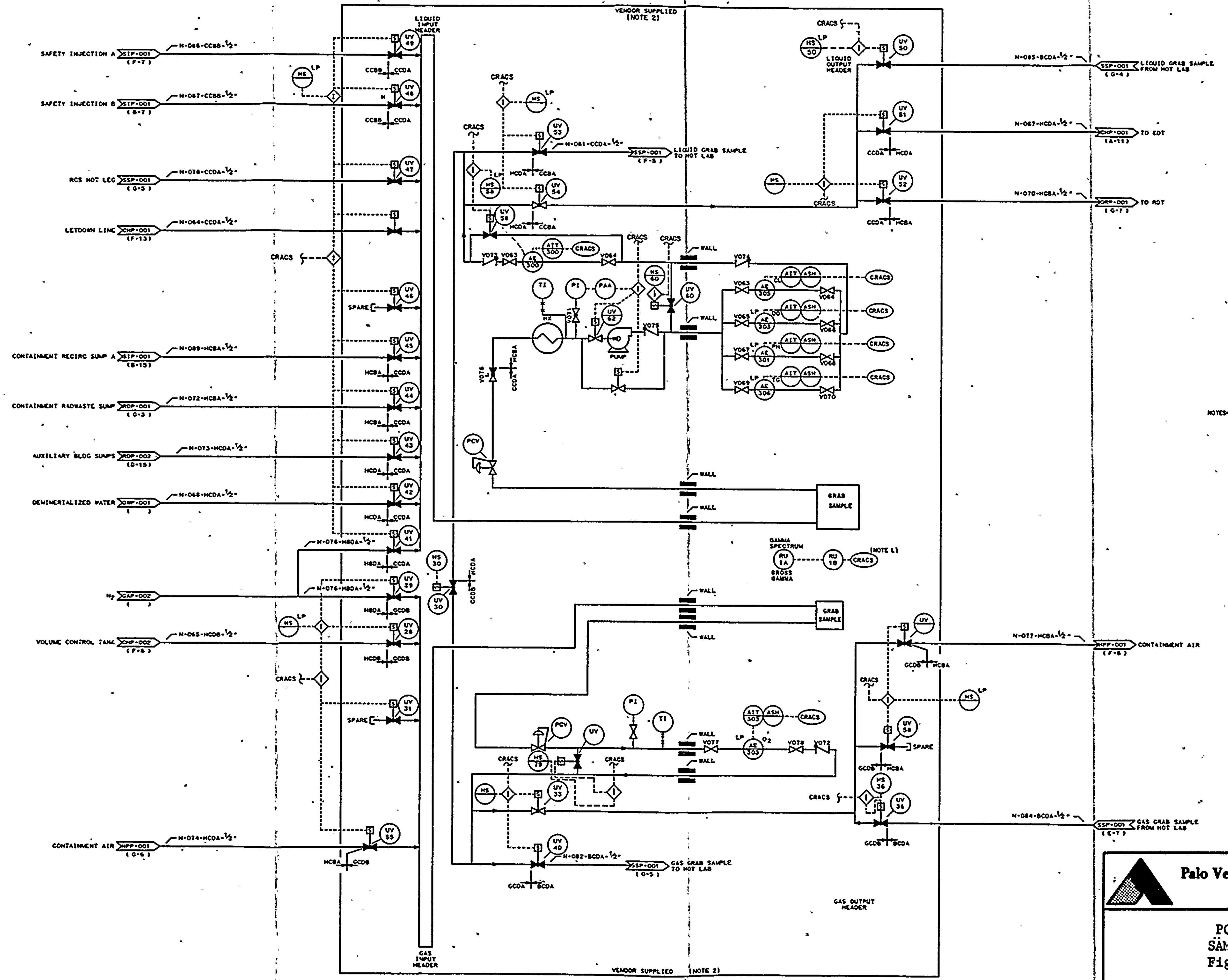

NOTES: 1. CRACS—CHEMICAL AND RADIOLOGICAL ANALYSIS COMPUTER SYSTEM—SUPPLIES VALVE & PUMP CONTROL AND DATA READOUT.

2. VENDOR SUPPLIED PORTION REPRESENTS A CONCEPTUAL FLOW DIAGRAM TO ACHIEVE DESIRED ANALYSES FROM BECHTEL SPECIFIED SAMPLE POINTS AND DISCHARGED TO BECHTEL SPECIFIED POINTS. VENDOR IS RESPONSIBLE FOR DETAILED DESIGN, INCLUDING BUT NOT LIMITED TO, PUMP SIZING, PRESSURE PROTECTION, NECESSARY OPERATOR INSTRUMENTATION AND ANY NEEDED VALVING.

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

THE SEPARATION GROUP DESIGNATOR N IS TO PRECEDE ALL LINE NUMBERS SHOWN ON THIS DRAWING UNLESS OTHERWISE INDICATED. 1

13-N-SSP-002 Rev. 0

Palo Verde Nuclear Generating Station
LLIR

POST ACCIDENT SAMPLING SYSTEM
Figure II.B.3-2

August 1981 Amendment 1

II.B.4 TRAINING FOR MITIGATING CORE DAMAGE

Position (NRC Letter, H.R. Denton to All Power Reactor Applicants and Licensees, dated March 28, 1980)

(1) Develop Training Program

A program is to be developed to ensure that all operating personnel are training in the use of installed plant systems to control or mitigate an accident in which the core is severely damaged. The training program should include the following topics:

A. Incore Instrumentation

1. Use of fixed or movable incore detectors to determine extent of core damage and geometry changes.

2. Use of thermocouples in determining peak temperatures; methods for extended range readings; methods for direct readings at terminal junctions.

3. Methods for calling up (printing) incore data from the plant computer.

B. Excore Nuclear Instrumentation (NISN)

Use of NIS for determination of void formation; void location basis for NIS response as a function of core temperatures and density changes.

C. Vital Instrumentation

1. Instrumentation response in an accident environment; failure sequence (time to failure, method of failure); indication reliability (actual vs. indicated level).

2. Alternative methods for measuring flows, pressures, levels, and temperatures.

    a. Determination of pressurizer level if all transmitters fail.

    b. Determination of letdown flow with a clogged filter (low flow).

    c. Determination of other Reactor Coolant System parameters if the primary method of measurement has failed.

D. Primary Chemistry

1. Expected chemistry results with severe core damage; consequences of transferring small quantities of liquid outside containment; importance of using leak tight systems.

2. Expected isotopic breakdown for core damage; for clad damage.

3. Corrosion effects of extended immersion in primary water; time to failure.

E. Radiation Monitoring

1. Response of Process and Area Monitors to severe damages; behavior of detectors when saturated; method for detecting radiation readings by direct measurement at detector output (overranged amplifier); expected accuracy of detectors at different locations; use of detectors to determine extent of core damage.

2. Methods of determining dose rate inside containment from measurements taken outside containment.

F. Gas Generation

1. Methods of H_2 generation during an accident; other sources of gas (Xe, Kr); techniques for venting or disposal of non-condensables.

2. H_2 flammability and explosive limit; sources of O_2 in containment or Reactor Coolant System.

(2) Complete Training

The course shall be developed and these personnel shall participate prior to fuel loading of unit one.

PVNGS Evaluation

(1) Develop Training Program

A course will be developed to train shift technical advisors and operating personnel through the operations chain to the licensed operators in the use of installed plant systems to control or mitigate an accident in which the core is severely damaged. The training program will include the topics suggested in the H.R. Denton letter of March 28, 1980.

Managers and technicians in the Instrumentation and Control (I & C), radiation protection and chemistry sections will receive training commensurate with their responsibilities that meets the requirements of the H.R. Denton letter of March 28, 1980. This training will consist of approximately 40 hours of formal classroom presentations by either a private consultant or the engineering staff. Operators
2 will acquire the theoretical basis for these actions in the academic programs and the practical application during the simulator training course. The total training time will exceed 80 hours.

(2) Complete Training:

The above training will be completed prior to full power operation.

II.D REACTOR COOLANT SYSTEM RELIEF AND SAFETY VALVES

II.D.1 PERFORMANCE TESTING OF BOILING-WATER REACTOR AND PRESSURIZED-WATER REACTOR RELIEF AND SAFETY VALVES

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.

PVNGS Evaluation

Refer to CESSAR Appendix B, Item II.D.1.

II.D.3 DIRECT INDICATION OF RELIEF AND SAFETY-VALVE POSITION

Position

Reactor coolant 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 primary code safety valves, located at the top of the pressurizer, are headered into the reactor drain tank (RDT) inside containment. Upstream of the common header each code safety valve is monitored for seal leakage by an in-line resistive-temperature device (RTD) (refer to FSAR Figure 5.1-1).

Indirect indication of code safety valve leakage is provided by an increase of RDT pressure and a decrease of pressurizer pressure and pressurizer level, monitored by safety-grade instrumentation. 1

Positive indication of safety valve position will be provided in the control room. Monitoring will be provided by an acoustic monitoring system consisting of an accelerometer (acoustic sensor) mounted downstream of each valve. The sensing instrumentation will be environmentally qualified to function in a post-LOCA environment in accordance with Regulatory Guide 1.89. A plant annunciator alarm will be provided to alarm valve opening. The acoustic monitoring system will be powered from a reliable

instrument bus with Class IE backup power. The system is
1 designed to meet the requirements of Revision 2 to Regulatory
Guide 1.97.

Installation of positive pressurizer safety valve position indication and development of emergency procedures will be completed prior to fuel loading of PVNGS Unit 1.

II.E SYSTEM DESIGN

II.E.1.1 AUXILIARY FEEDWATER SYSTEM EVALUATION

Position

The Office of Nuclear Reactor Regulation is requiring reevaluation of the auxiliary feedwater (AFW) systems for all PWR operating plant licensees and operating license applications. This action includes:

(1) Perform a simplified AFW system reliability analysis that uses event-tree and fault-tree logic techniques to determine the potential for AFW system failure under various loss-of-main-feedwater-transient conditions. Particular emphasis is given to determining potential failures that could result from human errors, common causes, single-point vulnerabilities and test and maintenance outages;

(2) Perform a deterministic review of the AFW system using the acceptance criteria of Standard Review Plan Section 10.4.9 and associated Branch Technical Position ASB 10-1 as principal guidance; and

(3) Reevaluate the AFW system flowrate design bases and criteria.

PVNGS Evaluation

An auxiliary feedwater system review was conducted on August 21 and 22, 1980, and included NRC participation.

This review included a review of the PVNGS Auxiliary Feedwater System Reliability Analysis. A transcript of this review was provided to NRC on October 17, 1980. Responses to open items of the review were provided to NRC on March 6, 1981.

The PVNGS Auxiliary Feedwater System Reliability Analysis was performed in response to the NRC letter of March 10, 1980 from D.F. Ross, Jr., to All Pending Operating License Applicants of Nuclear Steam Supply Systems Designed by Westinghouse and Combustion Engineering. The final analysis was submitted to the NRC by APS letter dated February 10, 1981 from E.E. Van Brunt, Jr., to Director of Nuclear Reactor Regulation
2 and was included as Appendix 10B of the PVNGS FSAR.

II.E.1.2 AUXILIARY FEEDWATER SYSTEM AUTOMATIC INITIATION AND FLOW INDICATION

PART 1: Auxiliary Feedwater System Automatic Initiation

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 (AFWS), the following requirements shall be implemented in the short term:

(1) The design shall provide for the automatic initiation of the AFWS.

(2) The automatic initiation signals and circuits shall be designed so that a single failure will not result in the loss of AFWS 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 AFWS from the control room shall be retained and shall be implemented so that a single failure in the manual circuits will not result in the loss of system function.

(6) The ac motor-driven pumps and valves in the AFWS shall be included in the automatic actuation (simultaneous and/or sequential) of the loads onto the emergency buses.

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

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

PART 2: Auxiliary Feedwater System Flowrate Indication

Position

Consistent with satisfying the requirements set forth in General Design Criterion 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

Refer to CESSAR Appendix B, Item II.E.1.2. In addition:

A. Automatic Initiation

1. The auxiliary feedwater system shown in PVNGS FSAR Figure 10.4-11 consists of two Seismic Category I pumps and their associated flow paths and valves and one non-Seismic Category I pump and its associated flow paths and valves.

   The emergency feedwater trains of the auxiliary feedwater system are 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 emergency feedwater trains of the auxiliary feedwater system are 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 CESSAR Figure 7.3-1d).

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

2. The emergency feedwater trains of the auxiliary feedwater system are composed of components in two separate and diverse load groups (i.e., load group 1 and load group 2). Each of the four emergency 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 isolation of auxiliary feedwater from a ruptured steam generator. Load group 2 powers the emergency 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 emergency turbine-driven auxiliary feedwater pump. No ac power is required for support of the turbine-driven emergency 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 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 ESFAS. 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 instrumentation channels, each of which is supplied by either offsite power or the diesel generator when offsite power is not available and is backed up by a Class IE battery.

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 and circuits 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.

The PVNGS auxiliary feedwater system complies with the recommendations of NUREG-0737, November 1980.

B. Flow Indication

1. The PVNGS design will include monitoring of auxiliary feedwater flow to both steam generators. These flow indicator channels are displayed on the main control boards. The flow indication system will be environmentally qualified. Class IE (safety grade) pressure indicators located up-stream of the manual block valves and Class IE steam generator level indicators are also provided. Pressure indicator PI-18A monitors the Train A turbine-driven auxiliary feedwater pump discharge pressure and flow indicators FI-40 and FI-41 will monitor flow to steam generator 1 (refer to FSAR Figure 10.4-11). Pressure indicator PI-17A monitors the Train B motor-driven auxiliary feedwater pump discharge pressure and flow indicators FI-40 and FI-41 will monitor the flow to steam generator 2. Four channels of Class IE steam generator level indication are provided for each steam generator.

2. The safety grade pressure and level and the added flow indication channels are powered from redundant Class IE buses.

The PVNGS FSAR Table 7.5-1 and Figure 10.4-11 will be amended to indicate this revision.

II.E.3.1 EMERGENCY POWER SUPPLY FOR PRESSURIZER HEATERS

<u>Position</u>

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

(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

Based upon calculated insulation losses and previous plant operating experience for C-E plants, heat losses from the PVNGS pressurizer are expected to range from 100,000 to 500,000 BTU/h. Preoperational testing will determine the actual PVNGS heat loss. Assuming a conservative heat loss of 500,000 BTU/h., a heater capacity of 150kW is required to maintain normal reactor coolant system operating pressure.

The PVNGS pressurizer heaters are configured as follows:

1. The PVNGS pressurizer heater (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 powered from the non-Class IE and Class IE distribution systems as shown in table II.E.3.1-1. Redundant heater capacity of 150 kW is available for manual loading on the diesel generators upon loss of normal (offsite) power. The

Table II.E.3.1-1

PVNGS PRESSURIZER HEATERS

Number of Heaters	Capacity (kW)	480V Bus	IE Power	IE Controls	SIAS Trip	Reset from Control Room
5-3 element groups	750	NGN-L11	No	No	No	N/A
5-3 element groups	750	NGN-L12	No	No	No	N/A
1-3 element groups	150	PGA-L33	Train A	Train A	Yes	No
1-3 element groups	150	PGB-L32	Train B	Train B	Yes	No

diesel generators are sized to accommodate this heater capacity concurrent with a forced shutdown or LOCA (refer to FSAR Table 8.3-3). In the event that heater capacity beyond that powered from a Class IE source (150 KW on each train) is required, heaters can be supplied from the non-Class IE power system that is fed from the offsite power system.

2. Heaters fed from the Class IE power system (300 kW) are automatically shed upon receipt of a loss of off-site power (LOP) or a safety injection actuation signal (SIAS). They may subsequently be manually reconnected to the ESF buses without shedding of any loads from the Class IE buses.

   Sufficient margin exists in each diesel generator for at least an additional 150 kW heater.

3. 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.

1 The PVNGS pressurizer heaters comply with the recommendations of NUREG-0737.

II.E.4.1 DEDICATED HYDROGEN PENETRATIONS

Position

Plants using external recombiners or purge systems for post-accident combustible gas control of the containment atmosphere should provide containment penetration 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 50, and that are sized to satisfy the flow requirements of the recombiner or purge system.

The procedures for the use of combustible gas control systems following an accident that results in a degraded core and release of radioactivity to the containment must be reviewed and revised, if necessary.

PVNGS Evaluation

Redundant independent combustible gas control systems are provided. Each system has dedicated containment penetrations, external hydrogen monitors, and connection points for an external hydrogen recombiner. Two portable hydrogen recombiners will be onsite and available for connection to the affected unit. Either recombiner is capable of reducing hydrogen levels as noted in FSAR Section 6.2.5. The two systems are completely independent and meet single failure criteria.

An additional hydrogen reduction capability is provided by a charcoal filtered purge exhaust unit. This non safety grade unit can be connected to either set of gas control containment penetrations. This capability would only be utilized in the event of separate failures in both recombiner units.

The arrangement of these combustible gas control methods is shown in FSAR Figure 6.2.5-1.

1 As noted in section II.B.2, the dose rate during installation of the hydrogen recombiners will be less than 5 rem/hr. This rate allows the requirements of GDC 19 to be met.

Therefore, the PVNGS containment hydrogen control system complies with the recommendations of NUREG-0737.

II.E.4.2 CONTAINMENT ISOLATION DEPENDABILITY

<u>Position</u>

(1) Containment isolation system designs shall comply with the recommendations of Standard Review Plan Section 6.2.4 (i.e., that there be diversity in the parameters sensed for the initiation of containment isolation).

(2) All plant personnel shall give careful consideration to the definition of essential and nonessential systems, identify each system determined to be essential, identify each system determined to be nonessential, describe the basis for selection of each essential system, modify their containment isolation designs accordingly, and report the results of the reevaluation to the NRC.

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

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

(5) The containment setpoint pressure that initiates containment isolation for nonessential penetrations must be reduced to the minimum compatible with normal operating conditions.

(6) Containment purge valves that do not satisfy the operability criteria set forth in Branch Technical Position CSB 6-4 or the Staff Interim Position of October 23, 1979 must be sealed closed as defined in SRP 6.2.4, Item II.3.f during operational conditions 1, 2, 3, and 4. Furthermore, these valves must be verified to be closed at least every 31 days.

(7) Containment purge and vent isolation valves must close on a high radiation signal.

PVNGS Evaluation

Refer to CESSAR Appendix B, Item II.E.4.2. In addition, 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 (1685 psig). The power access purge and refueling purge are additionally isolated by high containment purge radioactivity.

2. A generic review of fluid systems penetrating the containment for C-E designed plants was conducted for the C-E Owners Group on Post-TMI Efforts. The results of this review were used by Arizona Public Service to evaluate the PVNGS containment isolation system.

Essential systems are those systems critical to ensure the capability to mitigate consequences of accidents, to ensure the integrity of the reactor coolant pressure boundary and to ensure the capability to shut down the reactor and maintain it in a safe shutdown condition. Table II.E.4.2-1 lists essential systems penetrating the PVNGS containment. No essential systems are functionally isolated by a containment isolation actuation signal (CIAS) with the exception of the hydrogen control system, as classified in the table. This system is not immediately required for accident mitigation. The isolation valves can be manually opened from the control room as part of the hydrogen recombiner startup procedures.

The steam supply for turbine-driven auxiliary feedwater pump is from the main steam lines, upstream of the main steam isolation valves (MSIV). The steam supply valves on the main steam lines are normally closed. These valves automatically open on an auxiliary feedwater actuation signal (AFAS), and 2
can be manually opened from the main control room. A CIAS does not affect the operation of the isolation valves.

The atmospheric dump valves (ADV) are normally-closed and are manually operated from the main control room and remote shutdown panel. These valves are used to

2 release steam from the steam generators to the atmosphere for reactor heat removal. The ADVs are located upstream of the MSIVs and are not affected by a CIAS.

3. Non-essential systems penetrating the PVNGS containment are listed in table II.E.4.2-2. Non-essential systems are automatically isolated by a CIAS with the exception of the following systems:

   A. Those containing locked closed valves or flanged closed connections.

   B. Main steam and feedwater

   C. Steam generator blowdown and blowdown sample and safety injection drain

   D. RCP seal injection and CVCS charging

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The main steam and feedwater systems, while not essential, aid in heat removal during a small LOCA. These systems should not, therefore, be isolated by a CIAS generated on low pressurizer pressure. The steam and feedwater systems are isolated for a main steam line break by a main steam isolation actuation signal (MSIS) on high containment pressure (5 psig) or low steam generator pressure (870 psig).

The steam generator blowdown and blowdown sample systems are isolated by either a MSIS, auxiliary feedwater actuation signal (AFAS) or a safety injection actuation signal (SIAS). The safety injection drain is isolated on a SIAS. Plant parameters which generate SIAS also generate CIAS.

It is desirable to leave RCP seal injection and CVCS charging paths open to provide additional core protection after an accident in which offsite power is available. In addition, the charging pumps can be transferred to emergency power at the discretion of the operator in the event that offsite power is lost. Conversely, it is undesirable to lose charging or seal injection capability during normal operation due to an inadvertent CIAS. The potential release of fission products through the penetration is not a concern for the following reasons:

(1) Flow is into the containment and RCS.

(2) Check valves inside the containment prevent backflow out of the containment if the charging pumps stop.

(3) The connecting portions of the CVCS outside of containment are designed to Safety Class 2, Seismic Category I standards and have design pressure well in excess of containment design pressure.

(4) The operator has the capability of isolating these lines if continued charging or seal injection proves to be unnecessary.

4. Override of a CIAS signal is available for each containment isolation valve via the control switch for that valve. Resetting of a CIAS does not result in the automatic opening of containment isolation valves. Reopening requires operator action for each valve and does not compromise the containment isolation signal.

5. Item 1 above identifies 5 psig as the containment setpoint pressure that initiates containment isolation. The high containment pressure setpoint of 5 psig was established in CESSAR Table 7.2-4. Both 2

containment isolation (CIAS signal) and safety injection
(SIAS signal) are actuated at this setpoint. The set-
point was established to conservatively bound expected
fluctuations in containment pressure due to such factors
as instrument air leakage, containment air temperature
changes, and changes in differential pressure between
2 inside and outside containment. It also takes into
account instrument error.

Since the high containment pressure setpoint actuates both CIAS and SIAS, it was conservatively established to minimize spurious challenges to the safety injection system.

6. Containment power access purge isolation valves
satisfy the operability criteria set forth in Branch
Technical Position CSB 6-4. Containment refueling
purge isolation valves will be sealed closed except
2 during operational modes 5 and 6. These valves will
be verified closed once every 30 days. (Cold shutdown
and refueling respectively, see CESSAR Section 16.1.1).

7. As stated in item 1 above, both the power access purge and the refueling purge isolate on high containment purge radioactivity.

Table II.E.4.2-1
ESSENTIAL SYSTEMS PENETRATING THE PVNGS CONTAINMENT

System	Normal Position	Post CIAS Position	Notes	
HPSI	Closed	Closed	1	
LPSI	Closed	Closed	1	
Containment Spray	Closed	Closed	2	
Recirculation Sump Suction	Closed	Closed	3	
Long Term Recirculation	Closed	Closed	4	
Auxiliary Feedwater	Closed	Closed	5	
H_2 Control	Closed	Closed	4,7	
Containment Pressure Sensor	Open	Open	6	
Shutdown Cooling System	Closed	Closed	4	1

1. Opens on Safety Injection Actuation Signal
2. Opens on Containment Spray Actuation Signal
3. Opens on Recirculation Actuation Signal
4. Manually opened from control room
5. Opens on Auxiliary Feedwater Actuation Signal
6. Function requires it to remain open
7. Isolates on CIAS

Table II.E.4.2-2

NON-ESSENTIAL SYSTEMS PENETRATING THE PVNGS CONTAINMENT (Sheet 1 of 2)

System	Normal Position	Post CIAS Position	Notes
Demineralized Water	C	C	Locked
1 Fire Protection	C	C	Locked
Pool Cooling	C	C	Locked
Fuel Transfer	C	C	Flanged
Containment Test	C	C	Flanged
Service Air	C	C	Locked
Integrated Leak Rate Test	C	C	Flanged
Personnel Lock	C	C	-
Equipment Hatch	C	C	-
Emergency Lock	C	C	-
Pressurizer Sample - Water	C	C	1
Pressurizer Sample - Steam	C	C	1
Hot Leg Sample	C	C	1
High Pressure Nitrogen	C	C	1
Containment Purge (Refueling)	C	C	1,2
Radiation Monitor	O	C	1
Low Pressure Nitrogen	O	C	1
Instrument Air	O	C	1
Nuclear Cooling Water	O	C	1
CVCS Letdown	O	C	1
Reactor Drain Tank (RDT) Vent	O	C	1
CVCS RDT Drain/Fill	O	C	1
Chilled Water	O	C	1
Power Access Purge	O	C	1,2
Containment Normal Sump	O	C	1
Main Steam	O	O	3
Main Feedwater	O	O	3
Steam Generator Blowdown	O	O	4
Steam Generator Blowdown Sample 1	C	C	4

Table II.E.4.2-2

NON-ESSENTIAL SYSTEMS PENETRATING THE PVNGS CONTAINMENT (Sheet 2 of 2)

System	Normal Position	Post CIAS Position	Notes
Steam Generator Blowdown Sample 2	O	O	4
Safety Injection Drain	C	C	5
RCP Seal Injection	O	O	6
CVCS Charging	O	O	6

1. Closes on CIAS
2. Closes on Containment Purge Isolation Actuation Signal
3. Closes on MSIS
4. Closes on MSIS, AFAS or SIAS
5. Closes on SIAS
6. Seismic Category I check valve inside containment

II.F INSTRUMENTATION AND CONTROLS

II.F.1 ADDITIONAL ACCIDENT-MONITORING INSTRUMENTATION

A human factor analysis will be performed to ensure that the displays and controls added for additional-accident accomplish 1
this monitoring do not increase the potential for operator error (see section I.D.1). Installation will be completed 2
prior to fuel load.

II.F.1.1 ATTACHMENT 1, NOBLE GAS EFFLUENT MONITOR

Position

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 necessary to cover the ranges of interest.

(1) 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.

(2) Noble gas effluent monitoring shall be provided for the total range of concentration extending from normal condition (as low as reasonably achievable (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.

PVNGS Evaluation

FSAR section 11.5 provides detailed descriptions of the effluent
1 monitors installed at Palo Verde Units 1, 2 and 3. This includes
the additional monitors that have been added specifically to
address the NUREG-0737 and Reg Guide 1.97, Rev 2 requirements
for radiation monitoring. Installation and calibration will be
completed by September 30, 1982. A description of the calibration sources, frequency of calibration, and technique is provided in FSAR Table 11.5.1 and Sections 11.5.2.1.6.2 and
11.5.2.1.6 respectively. The instrumentation is described in
detail in FSAR Table 11.5-1. The outputs of the effluent
monitor and flow meter will be in μCi/cc and cc/h respectively
and when obtained need only be multiplied to obtain μCi/h.
2 Sampling of effluents will meet the criteria of ANSI NB.1 1969
as discussed in FSAR Sections 11.5.2.1.1.7.2.2 and 11.5.2.2.1.
Monitors are designed to meet a 90% efficient level for particulates and 90% efficiency for iodine as required by NUREG 0737
Table II.F.1-2. They are also designed to conform with design
basis shielding envelopes for sampling media as discussed in
FSAR Section 12.1.2.4 and item II.B.2. A human factor analysis
was performed as discussed in item I.D.1.

A. Wide-Range Effluent Monitor

1 In order to cover the dynamic range required, a normal range
monitor is used with a high range monitor. One decade is used
for overlap when switching between monitors. Both monitors

sample using controlled isokinetic flows. Their combined range is $10^{-6}\mu Ci/cc$ to $10^{+5}\mu Ci/cc$. These monitors are located on the plant vent, main condenser/gland seal exhaust, and the fuel
building vent. In addition to noble gas monitoring capability, 1
these instruments have separate particulate and iodine sample chambers for both the low and high range, and use charcoal
or Silver-Zeolite cartridges in conjunction with a portable 2
pump. All cartridges will be removed to the counting laboratory for gamma spectrum analysis. High range monitor iodine samples are provided with a lead shield. Procedures will be developed to define ALARA concepts for removal, transport and analysis. These monitors have complete digital readout and control from the Health Physics Office and the main control room. The high range monitors automatically switch to a new particulate/iodine cartridge pair when the current cartridge reaches a preset radiation level. Filter materials used minimize absorption of noble gases.
Samples are preconditioned as necessary to assure accurate 1
results without damaging the sample assemblies. Each monitor is controlled by a remote microprocessor. This microprocessor is linked by a "daisy chain" to a minicomputer which provides multiple informational displays on request by the operator. A dedicated alarm status line is maintained on the CRT display. This status line does not move with each change of CRT displays. Thus alarms are provided regardless of the status of the displays in the Health Physics Office and Main Control Room. Monitors are provided with an open structural construction

that provides for easy maintenance and good heat dissipation. Backup battery power is provided to assure continued microprocessor memory during a loss of external power sources. Multiple detectors are used to achieve the dynamic range required. Hard copy readouts are available from dedicated printers in the Health Physics office and the control room.

B. Main Steam Line Monitor

One area monitor with a collimating lead shield is mounted adjacent to each main steam line in the Main Steam Support Structure. These monitors measure direct dose rates from the main steam line to identify effluent from the atmospheric dump, main steam relief valves, and auxiliary feedwater pump discharge.
1 An extra 2 inches of shielding is placed on the containment side of the detector shield. There are a total of 4 detectors with one remote microprocessor for each 2 detectors. The ion chamber covers a range from 1mr/h to 10^7mr/h. The microprocessor is linked by a "daisy chain" to a minicomputer which provides multiple informational displays at the request of the operator. Alarms are provided regardless of the status of the displays in the Health Physics Office and the Control Room. Hardcopy readouts are provided from dedicated printers in the Health Physics Office and the Control Room. Backup battery power is provided to assure continued microprocessor memory during a loss of external power sources. The detector is designed to operate in a post-accident environmental condition with a background of 10 R/h.

C. Meeting the NRC Requirement

These monitors operate in conjunction with the other monitors as
discussed in FSAR section 11.5 and fulfill the requirements as 1.
outlined in NUREG 0737 and Regulatory Guide 1.97, Rev 2.
Installation will be completed prior to fuel load. 2

II.F.1.2 ATTACHMENT 2, SAMPLING AND ANALYSIS OF PLANT EFFLUENTS

<u>Position</u>

Because 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.

<u>PVNGS Evaluation</u>

PVNGS response to this item is included in the evaluation of section II.F.1.1 requirements.

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II.F.1.3 ATTACHMENT 3, CONTAINMENT HIGH-RANGE RADIATION MONITOR

Position

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 developed and qualified to function in an accident environment.

This requirement was revised in the October 30, 1979 letter from H.R. Denton to All Operating Nuclear Power Plants to provide for a photon-only measurement with an upper range of 10^7 R/hr.

PVNGS Evaluation

PVNGS response to this item is included in the evaluation of section II.F.1.1 requirements. 1

II.F.1.4 ATTACHMENT 4, CONTAINMENT PRESSURE MONITOR

Position

A continuous indication of containment pressure shall be provided in the control room of each operating reactor. Measurement and indication capability shall include three times the design pressure of the containment for concrete, four times the design pressure for steel, and -5 psig for all containments.

PVNGS Evaluation

Wide range containment pressure measurement will be provided consisting of redundant pressure transmitters whose signals of 1

containment pressure will be continuously displayed within the control room. Continuous recording will be provided for one channel over the entire range of pressure measurement. The transmitters will be located outside of the containment structure and will measure the containment pressure through sensing lines penetrating the containment structure. The range of the system will be from -5 to 180 psig, three times the containment
1 pressure.

The transmitters will be physically separated, redundant, environmentally qualified to function in a post-LOCA environment in accordance with Regulatory Guide 1.89, and seismically qualified to function during and following an SSE in accordance with Regulatory Guide 1.100. The safety grade pressure instrumentation is powered from redundant Class 1E buses. The instrumentation is designed to meet Regulatory Guide 1.97, Rev. 2.

II.F.1.5 ATTACHMENT 5, CONTAINMENT WATER LEVEL MONITOR

Position

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

PVNGS Evaluation

(1) Narrow Range Level Measurement

Continuous indication of the radwaste sumps (containment normal sumps) water level is provided in the control room. Each sump will be monitored from 6 inches above the bottom of the sump to 6 inches above the top of the sump by a sensor environmentally qualified to function in a post-LOCA environment in accordance with Regulatory Guide 1.89. The instrumentation will be powered from a reliable instrument bus with Class IE backup power. The instrumentation is designed to meet the requirements of Regulatory Guide 1.97, Rev. 2.

(2) Wide Range Level Measurement 1

Continuous control room indication is provided for containment water level from 6 inches above the top of the radwaste sump to 6 inches above the maximum expected flood level, providing a total range of 11 feet. The sensors will be physically separated, redundant, environmentally qualified to function in a post-LOCA environment in accordance with Regulatory Guide 1.89, and seismically qualified to function during and following an SSE in accordance with Regulatory Guide 1.100. The safety grade level instrumentation is powered from redundant Class 1E buses. Recording for one channel is provided in the control room. The instrumentation is designed to meet Revision 2 to Regulatory Guide 1.97, Rev. 2.

II.F.1.6 ATTACHMENT 6, CONTAINMENT HYDROGEN MONITOR

Position

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.

PVNGS Evaluation

Containment post-LOCA hydrogen monitoring is provided by two redundant and physically separated hydrogen analyzers. Each analyzer is of the thermal conductivity type, supplied by Comsip, Inc., Delphi Systems Division. A remote control panel
1 mounted on the main control board provides control of each analyzer. Indication is provided in the control room of hydrogen concentration over a range of 0 to 10%. Recording for one channel is also provided in the control room. The analyzers will be in standby during normal operation and can provide continuous indication of hydrogen concentration in less than 30 minutes after activation from the control room. The analyzers will operate under containment design conditions from -5 to 60 psig, the containment design pressure. The analyzers are environmentally qualified to function in a post-LOCA environment in accordance with Regulatory Guide 1.89 and seismically qualified to function during and following an SSE in accordance with Regulatory Guide 1.100. The safety grade hydrogen analyzer instrument channels are powered from redundant Class 1E buses.

The instrumentation is designed to meet Revision 2 to Regulatory Guide 1.97. 1

II.F.2 INSTRUMENTATION FOR DETECTION OF INADEQUATE CORE COOLING

Position

Licensees shall provide a description of any additional instrumentation or controls (primary or backup) proposed for the plant to supplement existing instrumentation (including primary coolant saturation monitors) in order to provide an unambiguous, easy-to-interpret indication of inadequate core cooling (ICC). 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

(1) Procedures to Detect and Recover from Conditions Leading to Inadequate Core Cooling 2

Emergency procedures are being developed which will pro-
vide the reactor operators with directions to detect and 1
recover from conditions that could lead to inadequate
core cooling and will be in place prior to fuel load.
The NSSS vendor will prepare emergency operating 2
procedure guidelines.

(2) Inadequate Core Cooling Instrumentation 2

Inadequate core cooling monitoring requirements can be met 1
by appropriately measuring and displaying margin to saturation,

reactor vessel water level above the core, and reactor core exit temperatures. In order to accomplish these functions, the PVNGS Unit 1, 2, and 3 design will utilize three instruments: (1) The Subcooled Margin Monitor to measure saturation/superheat margin, (2) The Heated Junction Thermocouple System to monitor level/temperature in the region of the upper portion of the reactor vessel, and (3) Core Exit Thermocouples to measure the temperature of the reactor coolant as it leaves the reactor core.

The Subcooled Margin Monitor is an instrument which indicates the degree of subcooling and superheat in the reactor coolant either in terms of temperature or pressure. The system is
1 designed as a two-channel system, each channel measuring four points within the reactor coolant system as follows:

Reactor Coolant Piping Hot Leg Temperature	(1)
Reactor Coolant Piping Cold Leg Temperature	(2)
Pressurizer Pressure	(1)

The range of the Subcooled Margin Monitor is increased by incorporation of signals from the Heated Junction Thermocouple System and the Core Exit Thermocouples such that the instrument can also calculate and display degrees superheat.

The Heated Junction Thermocouple System monitors liquid in the region above the core. Redundant strings of heated junction thermocouples are arranged in the reactor vessel head area to provide indication of conditions at eight levels. The system is a two-channel system each consisting of eight sensor outputs.

The Core Exit Thermocouples utilize a Type K (Chrome-Alumel) thermocouple. The thermocouples are located so as to monitor the temperature of the reactor coolant as it leaves the fuel assemblies. The system consists of 61 thermo-couples as follows: fifteen sensors in channel A plus 16 sensors in channel C appropriately isolated to the channel A system; and fifteen sensors in channel B plus 15 sensors in channel D appropriately isolated to the channel B system. 1

Location of the inadequate core cooling sensors is schemat-ically shown in Figure II.F.2-1.

SIGNAL PROCESSING AND DISPLAY

Signal processing and display in the control room of the inade-quate core cooling instrumentation is performed by the Qualified Safety Parameter Display System (QSPDS) (refer to I.D.2-1). The QSPDS consists of a microprocessor-based design for the signal processing equipment in conjunction with a display having alpha-numeric representation and associated keyboard for each of the two channels. Each channel accepts and processes input parameter signals and transmits its output to the alphanumeric display. The inadequate core cooling processing and display parameters are identified in Table II.F.2-1. The alphanumeric display,

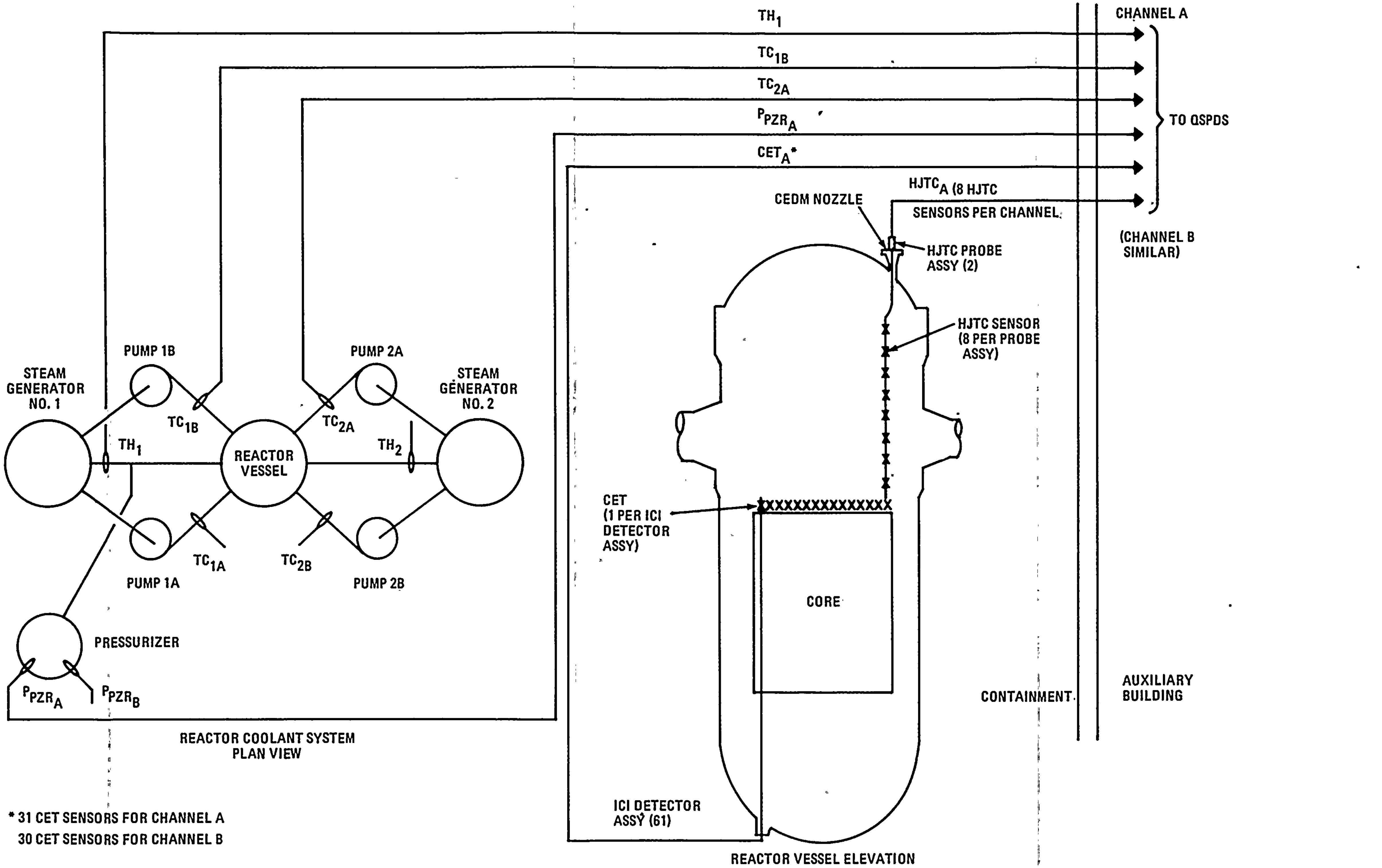
located in the control room, provides display of the inadequate
1 core cooling parameters. The redundant QSPDS is powered from
redundant Class IE buses.

TABLE II.F.2-1
INADEQUATE CORE COOLING PROCESSING AND DISPLAY

FUNCTION	SENSORS	PROCESSING RANGE	NO/CHANNEL	OUTPUTS
1. Saturation Margin	Hot Leg RTD Temp Cold Leg RTD Temp Max HJTC Unheated Junct. Temp *Rep. CET Temp Pressurizer Pressure	0-750°F 0-750°F 100-1800°F 200-2300 0-3200 psia	Up to 12	1. Temperature Margin sub-cooled to superheat) 2. Pressure margin
2. Reactor Vessel Liquid Level/ Temperature (Top of core to top of head)	HJTC Probe including: Heated Junction T/C Temp Unheated Junction T/C Temp Differential Temp (Level)	 100-1800°F 100-1800°F 100-1800°F	24	1. Liquid Level (Top of core to top of head) 2. HJTC sensor temperatures 3. Max HJTC un-heated junc-tion Temp
3. Core Exit Temperature	Core Exit Thermocouple (CET) Temperature	200-2300°F	Up to 31	1. CET temps. 2. *Rep. CET temp. 1

* Representative core exit thermocouple temperature is not yet determined but expected to be the maximum of the CET temperatures or average of the highest CET temperatures.

** Maximum superheat indication limited by input ranges.



Palo Verde Nuclear Generating Station
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SENSOR LOCATIONS FOR INADEQUATE CORE COOLING INSTRUMENTATION

Figure II.F.2-1

II.G . . ELECTRICAL POWER

II.G.1 EMERGENCY POWER FOR PRESSURIZER EQUIPMENT

Position

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

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

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

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

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

(4) The pressurizer level indication instrument channels shall be powered from the vital instrument buses. The buses shall have the capability of being supplied from

either the offsite power source or the emergency power source when offsite power is not available.

PVNGS Evaluation

PVNGS does not use PORVs or block valves.

Two channels of Class IE level instrumentation are provided for PVNGS. Pressurizer level channels L-110X and L-110Y 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 FSAR Figure 8.3-4, sheet 1 of 2). 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 comply with the recommendations of NUREG-0737, November 1980.

II.K MEASURES TO MITIGATE SMALL-BREAK LOCAs AND LOSS OF FEEDWATER ACCIDENTS

II.K.1 IE BULLETINS ON MEASURES TO MITIGATE SMALL-BREAK LOCAs AND LOSS OF FEEDWATER ACCIDENTS

II.K.1.5 Review of ESF Valves

Position (IE Bulletin No. 79-06B)

Review all safety-related valve positions, positioning requirements and positive controls to assure that valves remain positioned (open or closed) in a manner to ensure the proper operation of engineered safety features. Also review related procedures, such as those for maintenance, testing, plant and system startup, and supervisory periodic (e.g., daily/shift checks,) surveillance to ensure that such valves are returned to their correct positions following necessary manipulations and are maintained in their proper positions during all operational modes.

PVNGS Evaluation

Refer to CESSAR Appendix B, Item II.K.1.5. In addition, PVNGS will have tagout procedures and surveillance test procedures that will control safety system status. They will provide appropriate logs and checklists to ensure control of plant systems. Additionally, reviews will be conducted to verify that procedures for safety-related systems return those systems to service after having been tagged out for repair or surveillance testing. Refer to item I.C.2 for a discussion of procedures for shift relief and turnovers to ensure current plant conditions and system status is conveyed to the oncoming shift. Periodic audits will also be conducted to verify that tagouts are removed and systems returned to normal when the repair/testing has been completed.

II.K.1.10 Operability Status

<u>Position</u> (IE Bulletin No. 79-06B)

Review and modify as necessary your maintenance and test procedures to ensure that they require:

Verification, by test or inspection, of the operability of redundant safety-related systems prior to the removal of any safety-related system from service.

Verification of the operability of all safety-related systems when they are returned to service following maintenance or testing.

Explicit notification of involved reactor operational personnel whenever a safety-related system is removed from and returned to service.

<u>PVNGS Evaluation</u>

The PVNGS evaluation of item I.C.6 adequately addresses the concerns of this item.

II.K.2 COMMISSION ORDERS ON BABCOCK & WILCOX PLANTS (selected items applicable to PVNGS)

II.K.2.13 THERMAL MECHANICAL REPORT -- EFFECT OF HIGH-PRESSURE INJECTION ON VESSEL INTEGRITY FOR SMALL-BREAK LOSS-OF-COOLANT ACCIDENT WITH NO AUXILIARY FEEDWATER

Position

A detailed analysis shall be performed of the thermal-mechanical conditions in the reactor vessel during recovery from small breaks with an extended loss of all feedwater.

PVNGS Evaluation

Refer to CESSAR Appendix B, Item II.K.2.13.

II.K.2.17 POTENTIAL FOR VOIDING IN THE REACTOR COOLANT SYSTEM DURING TRANSIENTS

Position

Analyze the potential for voiding in the reactor coolant system (RCS) during anticipated transients.

PVNGS Evaluation

Refer to CESSAR Appendix B, Item II.K.2.17.

II.K.3 FINAL RECOMMENDATIONS OF B&O TASK FORCE

II.K.3.2 REPORT ON OVERALL SAFETY EFFECT OF POWER-OPERATED RELIEF VALVE ISOLATION SYSTEM

Position

(1) The licensee should submit a report for staff review documenting the various actions taken to decrease the probability of a small-break loss-of-coolant accident (LOCA) caused by a stuck-open power-operated relief valve (PORV) and show how those actions constitute sufficient improvements in reactor safety.

(2) Safety-valve failure rates based on past history of the operating plants designed by the specific nuclear steam supply system (NSSS) vendor should be included in the report submitted in response to (1) above.

PVNGS Evaluation

PVNGS does not use PORVs.

II.K.3.3 REPORTING SAFETY VALVE AND RELIEF VALVE FAILURES AND CHALLENGES

Position (NUREG-0694)

Assure that any failure of a PORV or safety valve to close will be reported to the NRC promptly. All challenges to the PORVs or safety valves should be documented in the annual report.

PVNGS Evaluation

PVNGS does not use PORVs. The Station Manual procedure governing reports to the NRC will require reporting any failure of a reactor coolant system valve to reseat following actuation, and challenges to RCS safety valves, to be documented in the annual report, commencing with full power operation.

II.K.3.5 AUTOMATIC TRIP OF REACTOR COOLANT PUMPS DURING LOSS-OF-COOLANT ACCIDENT

Position

Tripping of the reactor coolant pumps in case of a loss-of-coolant accident (LOCA) is not an ideal solution. Licensees should consider other solutions to the small-break LOCA problem (for example, an increase in safety injection flow rate). In the meantime, until a better solution is found, the reactor coolant pumps should be tripped automatically in case of a small-break LOCA. The signals designated to initiate the pump trip are discussed in NUREG-0623.

PVNGS Evaluation

Refer to CESSAR Appendix B, Item II.K.3.5.

II.K.3.17 REPORT ON OUTAGES OF EMERGENCY CORE COOLING SYSTEMS LICENSEE REPORT AND PROPOSED TECHNICAL SPECIFICATION CHANGES

Position

Several components of the emergency core cooling (ECC) systems are permitted by technical specifications to have substantial outage times (e.g., 72 hours for one diesel-generator; 14 days for the HPCI system). In addition, there are no cumulative outage time limitations for ECC systems. Licensees should submit a report detailing outage dates and lengths of outages for all ECC systems for the last 5 years of operation. The report should also include the causes of the outages (e.g., controller failures, spurious isolation).

PVNGS Evaluation

Arizona Public Service Company will establish a program for data collection of outage dates and lengths of outages, cause of the outage, ECCS systems or components involved in the outage, and the corrective action taken for ECC systems prior to fuel load of Unit 1. 2

II.K.3.25 EFFECT OF LOSS OF ALTERNATING-CURRENT POWER ON PUMP SEALS

Position

The licensees should determine, on a plant-specific basis, by analysis or experiment, the consequences of a loss of cooling water to the reactor recirculation pump seal coolers. The pump seals should be designed to withstand a complete loss of alternating-current (ac) power for at least 2 hours. Adequacy of the seal design should be demonstrated.

PVNGS Evaluation

Refer to CESSAR FSAR Appendix B, Item II.K.3.25. In addition, 2
the reactor coolant pump normal cooling water system (nuclear
cooling water system) is backed up by the essential cooling
water system to supply cooling water to the seals during loss
of ac power. On loss of ac power the nuclear cooling water
pumps will stop, but the redundant essential cooling water
system, powered from the Class IE redundant power supplies, 2
will automatically supply the reactor coolant pump seals with
cooling water. The essential cooling water system is
described in the PVNGS FSAR Section 9.2.2.

A C-E test program was conducted to determine the effect on the
reactor coolant pump seals due to a loss of component cooling
water to the pump seal assembly. The test results demonstrate 1
that the pumps will continue to operate for extended periods
without exceeding design seal leakage limits and seal

temperature limits. It has been shown that the System 80 reactor
1 coolant pumps are capable of operating without component cooling
water for 30 minutes without sustaining damage to the pump
thrust bearings.

II.K.3.30 REVISED SMALL-BREAK LOSS-OF-COOLANT-ACCIDENT METHODS TO SHOW COMPLIANCE WITH 10 CFR PART 50, APPENDIX K

Position

The analysis methods used by nuclear steam supply system (NSSS) vendors and/or fuel suppliers for small-break loss-of-coolant accident (LOCA) analysis for compliance with Appendix K to 10 CFR Part 50 should be revised, documented, and submitted for NRC approval. The revisions should account for comparisons with experimental data, including data from the LOFT Test and Semi-scale Test facilities.

PVNGS Evaluation

Refer to CESSAR Appendix B, Item II.K.3.30.

II.K.3.31 PLANT-SPECIFIC CALCULATIONS TO SHOW COMPLIANCE WITH 10 CFR PART 50.46

Position

Plant-specific calculations using NRC-approved models for small-break loss-of-coolant accidents (LOCAs) as described in Item II.K.3.30 to show compliance with 10 CFR 50.46 should be submitted for NRC approval by all licensees.

PVNGS Evaluation

Refer to CESSAR Appendix B, Item II.K.3.31.

III. EMERGENCY PREPAREDNESS AND RADIATION EFFECTS

III.A NRC AND LICENSEE PREPAREDNESS

III.A.1.1 UPGRADE EMERGENCY PREPAREDNESS

Position (NUREG-0694)

(1) Comply with 10CFR50, Appendix E

Comply with Appendix E, "Emergency Facilities," to 10 CFR Part 50, Regulatory Guide 1.101, "Emergency Planning for Nuclear Power Plants," and for the offsite plans, meet essential elements of NUREG-75/111 or have a favorable finding from FEMA.

(2) Comply with NUREG-0654

Provide an emergency response plan in substantial compliance with NUREG-0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants" (which may be modified as a result of public comments solicited in early 1980) except that only a description of and completion schedule for the means for providing prompt notification to the population (App. 3), the staffing for emergencies in addition to that already required (Table B.1), and an upgraded meteorological program (App. 2) need be provided (Ref. 10). NRC will give substantial weight [to FEMA] findings on offsite plans in judging the adequacy against NUREG-0654.

(3) Conduct Exercise

Perform an emergency response exercise to test the integrated capability and a major portion of the basic elements existing within emergency preparedness plans and organizations.

(4) Meteorological Data (NUREG 0737 Item III.A.2)

Revision 1 to NUREG-0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," provides meteorological criteria to fulfill, in part, the standard that "Adequate methods, systems, and equipment for assessing and monitoring actual or potential offsite consequences of a radiological emergency condition are in use" (see 10 CFR §50.47). The position in Appendix 2 to NUREG-0654 outlines four essential elements that can be categorized into three functions measurements, assessment, and communications.

(a) Adequate operational meteorological measurement program

(b) Backup system/procedures to obtain real-time local meteorological data

(c) System for making real-time, site specific estimates and predictions of atmospheric effluent transport and diffusion during and immediately following an accidental airborne radioactivity release

(d) Remote interrogation of systems producing meteorological data and effluent transport and diffusion elements.

PVNGS Evaluation

(1) Comply with 10CFR50 Appendix E

The PVNGS Emergency Plan submitted with the FSAR complies with Appendix E, "Emergency Facilities" to 10 CFR Part 50 and to Regulation Guide 1.101 as discussed in FSAR Section 1.8.

The Arizona Division of Emergency Services is currently preparing a comprehensive emergency response plan for fixed nuclear facilities which will incorporate actions to be taken by levels of government within the State of Arizona. This plan is expected to meet the essential elements of NUREG-75/111 and is expected to be forwarded to FEMA for review prior to fuel loading of Unit 1.

(2) Comply with NUREG-0654

The PVNGS Emergency Plan addresses the emergency planning criteria contained in NUREG-0654, including the means for providing prompt notification to the public, staffing for emergencies, and an upgraded meteorological program. The comprehensive emergency response plan for fixed nuclear facilities currently being developed by the Arizona Division 2

Services is expected to address the emergency planning criteria contained in NUREG-0654.

(3) Conduct Exercise

Prior to fuel loading of Unit 1, an emergency response exercise will be performed. The Arizona Division of Emergency Services will be invited to participate and major elements of both the PVNGS Emergency Plan and the state emergency response plan for fixed nuclear facilities will be tested.

(4) Meteorological Data

The PVNGS meteorological system is designed in accordance
2 with draft Regulatory Guide 1.23, Rev. 1 and satisfies the
requirements set forth in Regulatory Guide 1.97, Rev. 2
and NUREG's-0696, 0654, and 0737.

It will consist of two independent towers, sensors, signal
conditioning, amplifiers and processing systems with backup
power source. The system includes a highly reliable digital
data processor system to develop real time dispersion and
transport estimates. Additionally, auxiliary analog
1 information is recorded for all parameters monitored in
each independent system locally.

Real time validation of digital meteorological data will identify suspect data so that backup data will be used based on data reasonability by the comparison percentage method.

Data acquisition of the two independent meteorological system signals may be accomplished for projected dose calculations, CRT displays and remote data transmission. 1
The same capability is provided for the nuclear data link. As independent, redundant and validated signals are used, the system can achieve a non-availability of 0.01. 2

Below is a summary table of the sensors used in each independent meteorological system to monitor the environmental 1
parameters, and the common data processing available.

Sensor / Location	200' Wind Speed and Wind Direction Sensor	35' Wind Speed and Wind Direction Sensor	200' & 35' Aspirated Temperature Sensor	35' Aspirated Dewpoint Sensor	Rainfall Monitor Located at Ground Level	
Tower System A	Provided	Provided	Provided	Provided	Provided	2
Tower System B	Provided	Provided	Provided	Not Provided	Not Provided	1
Digital & Analog Processing	Provided	Provided	35' Ambient Only 200'-35' ΔT	Provided	Provided	2

The environmental parameters monitored by each independent tower system permits highly accurate and reliable meteorological data necessary to cover all data for the Pasquill 1
stability classes and transport protections needed for the PVNGS site.

The auxiliary analog information for each tower system is
provided on analog recorders located at the meteorological
station in the respective tower system equipment trailers.
Analog information for each tower system is converted to
digital data and transmitted by two serial links to the
plant site where the data is reduced to fifteen-minute
1 and hourly average meteorological parameters, and where all
effluent and environmental parameters are recorded and
available for time history displays in the control room,
emergency response facilities, and at external locations.
Serial data is also provided to the radiation exposure
management system to meet the reporting requirements of
2 Regulatory Guide 1.21. In addition, the signal is avail-
able to the multi-channel processing system for offsite
and projected dose calculation, technical support center
and the emergency operations facility CRT displays.

The models used for providing the estimates of offsite
exposure are:

- Plume exposure - Class A model will be used with
1 15 minute average meteorology data for initial
transport and diffusion estimates within
15 minutes following the classification of the
incident.

- Ingestion Zone - Class B model will be used for
relative concentration of the plume emergency
planning zone (EPZ) and ingestion EPZ.

The output of the processor provides graphic displays of 2, 10, and 50 mile graphics with terrain, major population zones, and a display of environs for radiation monitors located in the plume EPZ. Capability is provided to view this display in the Technical Support Center (TSC), 1 alternate TSC's, and offsite Emergency Operations Facility (EOF). Additional capabilities exist to view this display at selected locations. Supplemental information displayed will be rain meteorological data, numerical outputs, etc.

III.A.1.2 UPGRADE EMERGENCY SUPPORT FACILITIES

<u>Position</u> (NRC letter, D.G. Eisenhut to All Licensees of Operating Plants and Holders of Construction Permits, dated February 18, 1981)

Each operating nuclear power plant shall maintain an onsite technical support center (TSC) 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 TSC. Records that pertain to the as-built conditions and layout of structures, systems, and components shall be readily available to personnel in the TSC.

An operational support center (OSC) shall be established separate from the control room and other emergency response facilities as a place where operations support personnel can assemble and report in an emergency situation to receive instructions from the operating staff. Communications shall be provided between the OSC, TSC, EOF, and control room.

An emergency operating facility (EOF) will be operated by the licensee for continued evaluation and coordination of all licensee activities related to an emergency having or potentially having environmental consequences.

PVNGS Evaluation

A. Operational Support Center (OSC)

An area separate from the unit control rooms has been designated as the PVNGS onsite Operational Support Center (OSC). This is an assembly area of approximately 600 sq. ft. where plant operations support personnel will report in an emergency situation for further orders or assignment. The OSC is located in each unit at Elevation 140'-0" of the Auxiliary Building, adjacent to the main access corridor into the access control area (see figures III.A.1.2-1 and III.A.1.2-2). The OSC has communication with the control room and the onsite Technical Support Center (TSC). The Emergency Plan will reflect the existence of the OSC and will establish the methods and lines of communication and management.

1

B. Onsite TSC and Satellite TSCs

An onsite TSC and the satellite TSCs for Palo Verde Nuclear Generating Station have been established and will be fully operational prior to fuel load. The TSC is an underground structure of approximately 10,000 sq. ft. of area located within the site protected area adjacent to the main entry into the plant protected area (see figures III.A.1.2-1 and III.A.1.2-3). The TSC provides a work area for supervisory and technical personnel from Arizona Public Service Company and the NRC. It has communications with the control room and the other response centers. Satellite TSCs have been established adjacent to the control room in each unit to facilitate face to face communications between control room personnel and support personnel.

The satellite TSC has the same display capabilities as the main TSC (see figures III.A.1.2-1 and III.A.1.2-2). An analysis will be performed to determine equipment to be incorporated into the TSC to monitor and display the status of the affected unit. 1
Display of data at the TSC will be in accordance with NUREG 0696. The TSC will be habitable to permit occupancy following a loss of coolant accident. Monitoring equipment will be provided for direct and airborne radioactive contaminants to provide warning if radiation levels in the TSC are reaching dangerous levels. The TSC will have permanently installed radiation monitoring and filtered ventilation systems. The nonredundant systems in conjunction with building shielding will ensure that the combined doses from airborne radioactivity and direct radiation 1
meet GDC 19. As-built drawings and other appropriate records will be available in readily retrievable form at the site and will be accessible to the TSC.

The Palo Verde Nuclear Generating Station Emergency Plan will be revised to describe the TSC and its function.

C. <u>Emergency Operations Facility (EOF)</u>

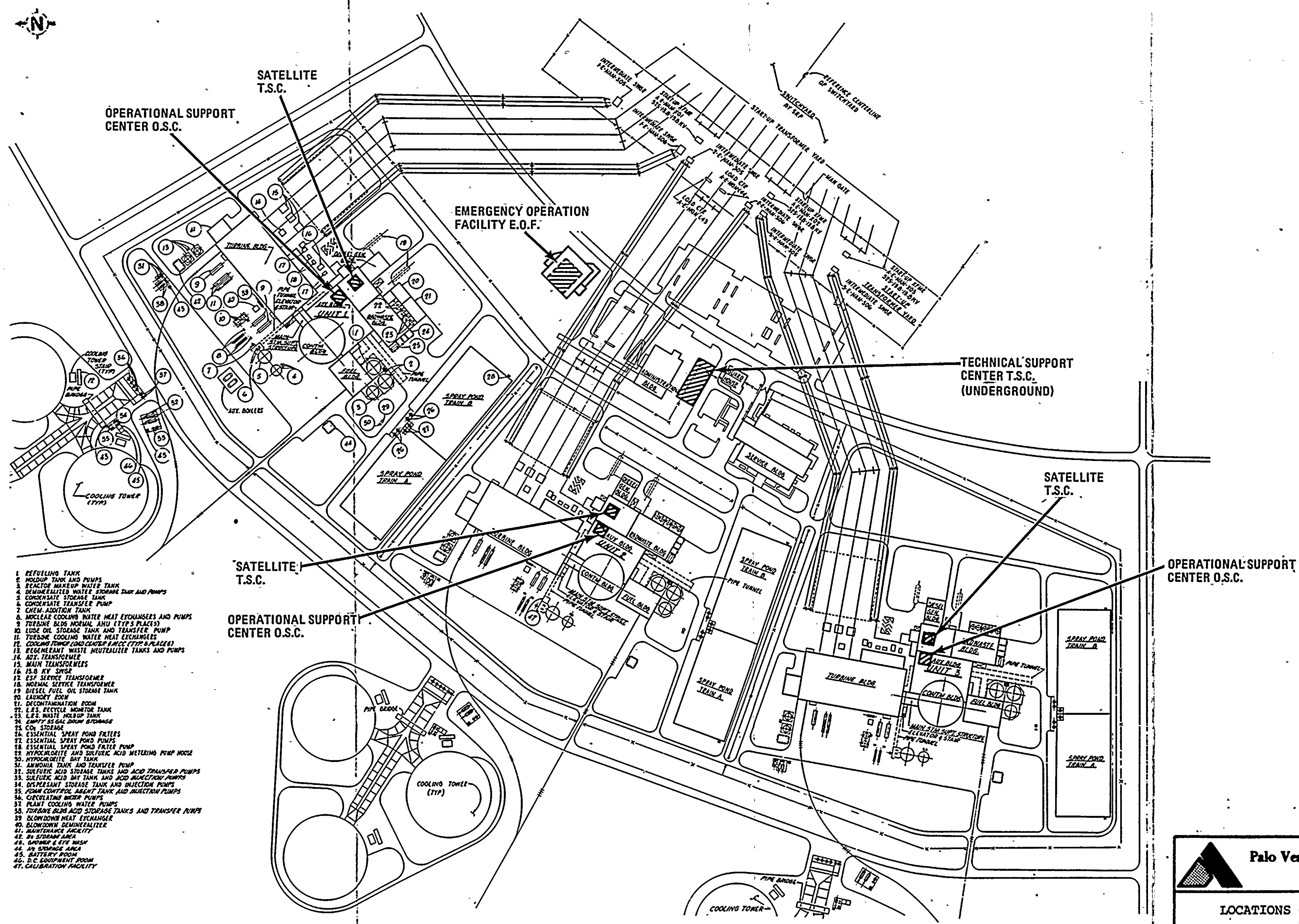
An Emergency Operations Facility (EOF) for PVNGS has been established and will be fully operational prior to PVNGS Unit 1 operating license. 1
This facility is located outside of the protected area and occupies approximately 6000 sq. ft. of the basement floor in the Administration Annex Building (see figures III.A.1.2-1 and III.A.1.2-4). The EOF provides space for operations personnel in support of the TSC. Plant personnel will be able to evaluate the magnitude and effect of radioactive

releases from the plant and recommend appropriate protective
1 measures. The EOF will have ventilation isolation capability
and be shielded against direct radiation. The facility is
habitable following a loss of coolant accident. A study will
be conducted to determine analysis and monitoring equipment
1 required. Display of data at the EOF will be in accordance
with NUREG 0696. Communications with the TSC will be provided.

1 The EOF provides space for recovery operations following an
accident as well as training facilities for the site.

The PVNGS Emergency Plan will be revised to describe the EOF
2 and its functions.

Implementation of these items will be complete before PVNGS
Unit 1 fuel load.



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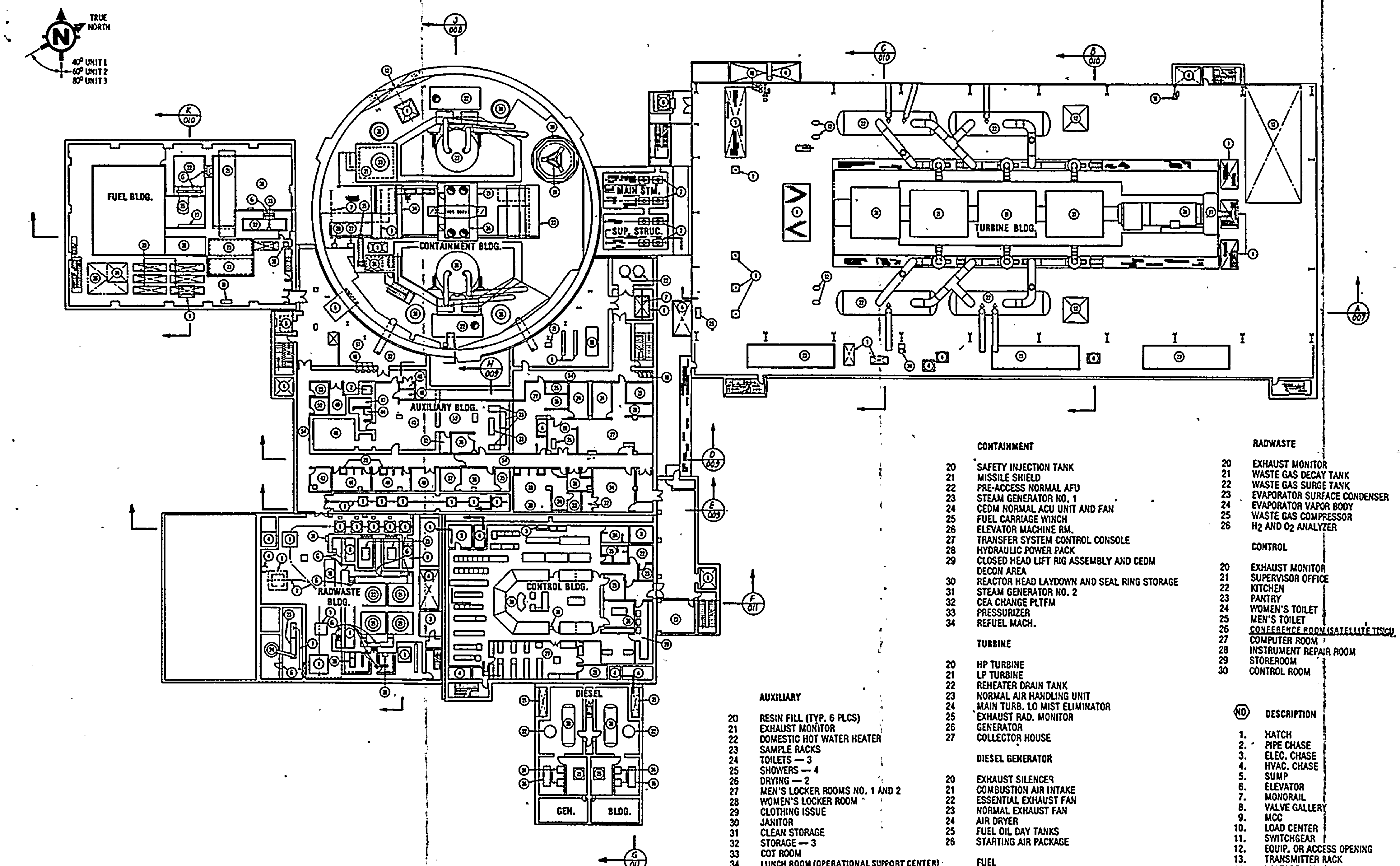
LOCATIONS OF EMERGENCY RESPONSE FACILITIES

Figure III.A.1.2-1



AUXILIARY

20 RESIN FILL (TYP. 6 PLCS)
21 EXHAUST MONITOR
22 DOMESTIC HOT WATER HEATER
23 SAMPLE RACKS
24 TOILETS — 3
25 SHOWERS — 4
26 DRYING — 2
27 MEN'S LOCKER ROOMS NO. 1 AND 2
28 WOMEN'S LOCKER ROOM
29 CLOTHING ISSUE
30 JANITOR
31 CLEAN STORAGE
32 STORAGE — 3
33 COT ROOM
34 LUNCH ROOM (OPERATIONAL SUPPORT CENTER)
35 H. P. OFFICE
36 HEALTH PHYSICS
37 INST. RESP. MAINT. ISSUE
38 OFFICE
39 SAMPLING AREA
40 FIRST AID
41 PERSONNEL DECON AREA — 2
42 H. P. SAMPLE COUNT. ROOM
43 HOT LAB
44 SAMPLE ROOM HOOD AND SINK
45 FLAMMABLE STORAGE
46 CYLINDER STORAGE
47 PIPING AREA
48 COUNTING ROOM
49 TOOL AND EQUIP. ROOM
50 SPECIAL DECON.
51 DECON. ROOM
52 EMERGENCY SHOWER AND EYE WASH
53 COLD LAB
54 CORRIDOR

CONTAINMENT

20 SAFETY INJECTION TANK
21 MISSILE SHIELD
22 PRE-ACCESS NORMAL AFU
23 STEAM GENERATOR NO. 1
24 CEDM NORMAL ACU UNIT AND FAN
25 FUEL CARRIAGE WINCH
26 ELEVATOR MACHINE RM.
27 TRANSFER SYSTEM CONTROL CONSOLE
28 HYDRAULIC POWER PACK
29 CLOSED HEAD LIFT RIG ASSEMBLY AND CEDM DECON AREA
30 REACTOR HEAD LAYDOWN AND SEAL RING STORAGE
31 STEAM GENERATOR NO. 2
32 CEA CHANGE PLTFM
33 PRESSURIZER
34 REFUEL MACH.

TURBINE

20 HP TURBINE
21 LP TURBINE
22 REHEATER DRAIN TANK
23 NORMAL AIR HANDLING UNIT
24 MAIN TURB. LO MIST ELIMINATOR
25 EXHAUST RAD. MONITOR
26 GENERATOR
27 COLLECTOR HOUSE

DIESEL GENERATOR

20 EXHAUST SILENCER
21 COMBUSTION AIR INTAKE
22 ESSENTIAL EXHAUST FAN
23 NORMAL EXHAUST FAN
24 AIR DRYER
25 FUEL OIL DAY TANKS
26 STARTING AIR PACKAGE

FUEL

20 SPENT FUEL POOL
21 SPENT FUEL HANDLING MACHINE
22 CASK LOADING PIT
23 NEW FUEL STORAGE COVERS
24 CASK LIFTING RIG LAYDOWN AREA
25 DECON PIT
26 DECON PIT COVER
27 GATE STORAGE
28 NEW FUEL INSPECTION PIT
29 NEW FUEL CONTAINERS LAYDOWN AREA
30 UPENDER HYD. DRIVE UNIT
31 POOL COOLING PURIFICATION PANEL
32 CANAL
33 NEW FUEL ELEVATOR

RADWASTE

20 EXHAUST MONITOR
21 WASTE GAS DECAY TANK
22 WASTE GAS SURGE TANK
23 EVAPORATOR SURFACE CONDENSER
24 EVAPORATOR VAPOR BODY
25 WASTE GAS COMPRESSOR
26 H_2 AND O_2 ANALYZER

CONTROL

20 EXHAUST MONITOR
21 SUPERVISOR OFFICE
22 KITCHEN
23 PANTRY
24 WOMEN'S TOILET
25 MEN'S TOILET
26 CONFERENCE ROOM (SATELLITE TSC)
27 COMPUTER ROOM
28 INSTRUMENT REPAIR ROOM
29 STOREROOM
30 CONTROL ROOM

NO	DESCRIPTION
1.	HATCH
2.	PIPE CHASE
3.	ELEC. CHASE
4.	HVAC. CHASE
5.	SUMP
6.	ELEVATOR
7.	MONORAIL
8.	VALVE GALLERY
9.	MCC
10.	LOAD CENTER
11.	SWITCHGEAR
12.	EQUIP. OR ACCESS OPENING
13.	TRANSMITTER RACK
14.	VOLTAGE REG.
15.	RELAY CAB.
16.	LIGHTING PANEL
17.	DISTRIBUTION PANEL
18.	JUNCTION BOX
19.	PIPE TRENCH

→ DENOTES UP U.N.O.
INDICATES ONE-WAY DOOR OR CONTROLLED FROM INSIDE OF BLDG

G-GATE

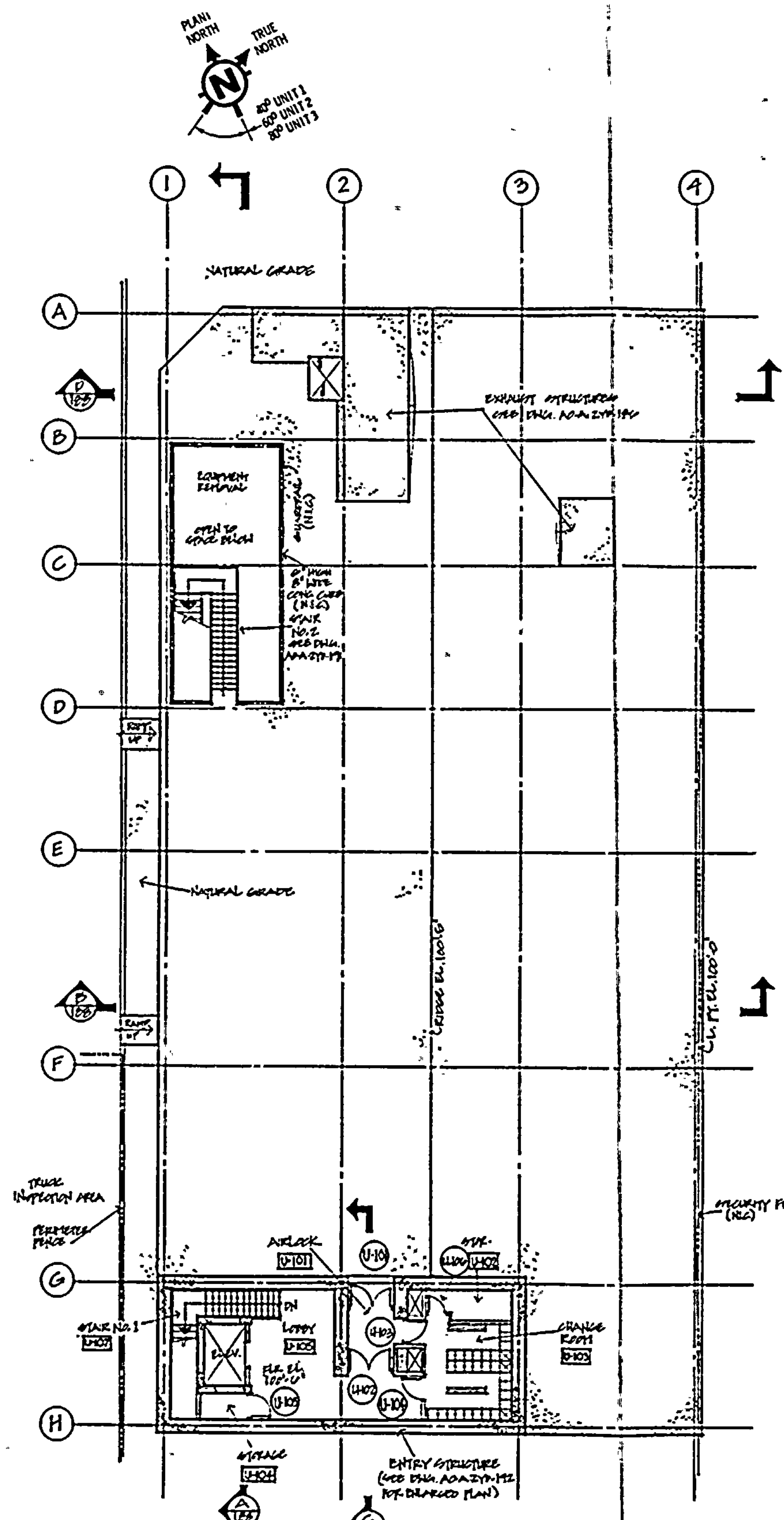
△ INDICATES CONTROLLED ACCESS AND/OR ROLL UP DOOR

13-P-00B-005, REV 3

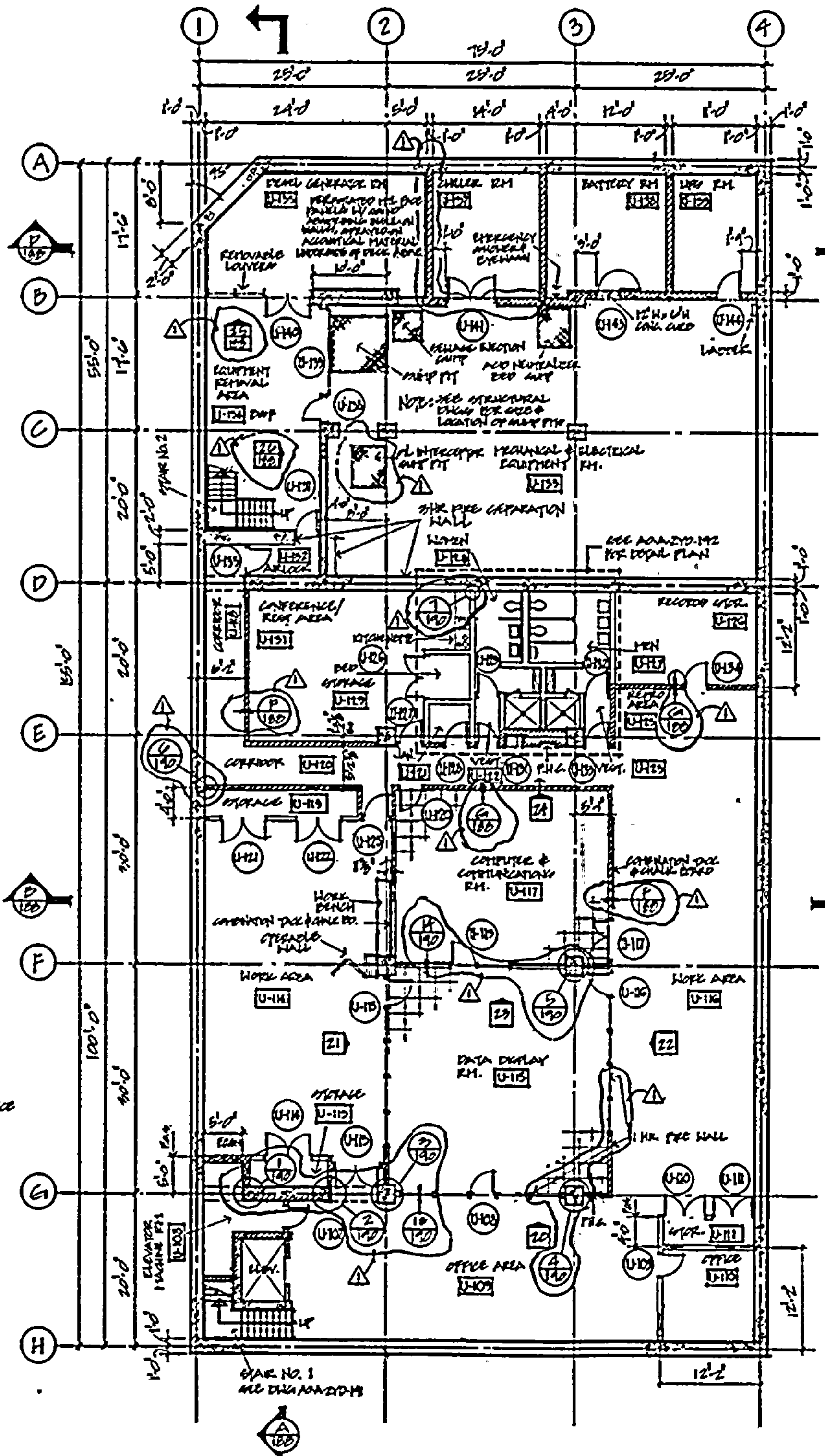
Palo Verde Nuclear Generating Station
LLIR

SATELLITE T.S.C. & OPERATIONAL SUPPORT CENTER

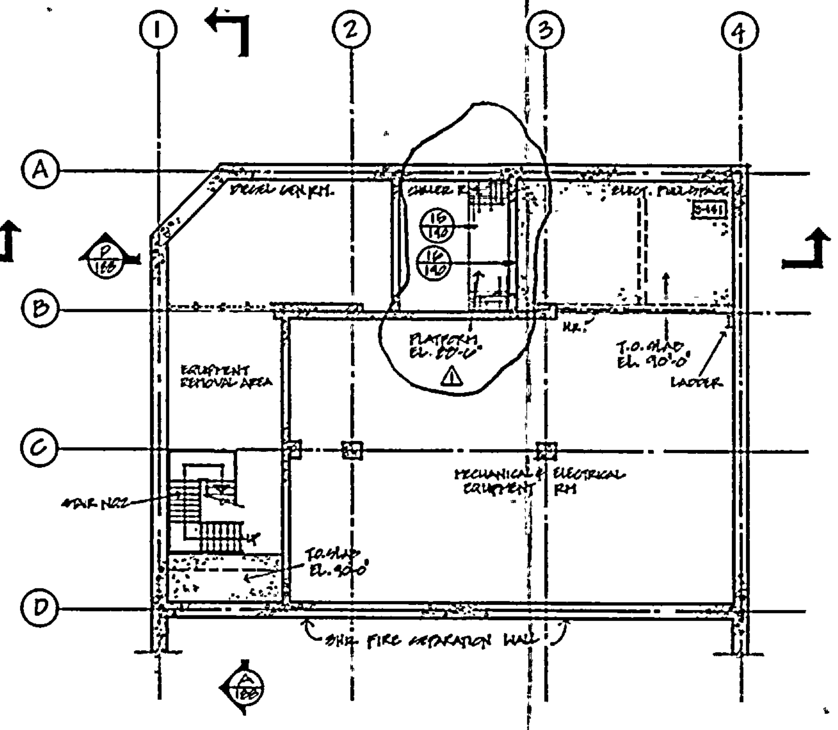
Figure III.A.1.2-2



PLAN @ GRADE



PLAN-BELOW GRADE @ EL.80'-0"



PARTIAL PLAN-MEZZ. @ EL. 90'-0"

NOTE:
UNLESS OTHERWISE NOTED ALL EXCAVATION, BACKFILL, & CONCRETE FOR THE BUILDING IS EXCLUDED FROM THE SUBCONTRACTORS SCOPE OF WORK.

EXCEPTIONS ARE: 1'-0" THICK 2ND STAGE REINFORCED CONC. SLAB & EQUIPMENT PADS IN ROOM U-133 & REINFORCED CONC. VAULT FOR DIESEL OIL STORAGE TANK SHOWN ON DWG. AO-P-ZYA-9[illegible]. (STL. TROWEL FINISH FOR SLAB - NAT. CONC. FINISH FOR VAULT)

CONCRETE NOTES:

1. ALL CONCRETE HEREIN REQUIRED SHALL BE DONE IN ACCORDANCE WITH ACI STANDARD 301-72, WITH THE FOLLOWING MODIFICATIONS:
   - CEMENT SHALL COMPLY WITH ASTM C150, TYPE II.
   - F'C (28 DAYS) SHALL BE 4,000 PSI FOR ALL CONCRETE.
   - REINFORCING SHALL BE #11 DEFORMED BILLET STEEL COMPLYING WITH ASTM A615, GRADE 60.
2. WORK INCLUDES:
   - A. FURNISHING & DELIVERING CONCRETE
   - B. FURNISHING & DELIVERING REINFORCING STEEL
   - C. PLACING REINFORCEMENT
   - D. FURNISHING, LOCATING & PLACING EMBEDMENTS, ANCHOR BOLTS & INSERTS
   - E. PLACING, FINISHING & CURING OF CONCRETE
3. SLEEVES SHALL BE PLACED WITHIN 1/2" OF THE LOCATION SPECIFIED.
4. THE COVER OVER REINFORCEMENT SHALL BE MINIMUM 3/4" CONCRETE.
5. (FIELD BENDING OF REINFORCING BARS)

   BARS PROJECTING FROM EXISTING CONCRETE MAY BE BENT, IF NECESSARY, FOR THE CONVENIENCE OF CONSTRUCTION PROVIDING THE BENDING AND STRAIGHTENING ARE IN ACCORDANCE WITH BECHTEL STANDARDS.
6. (INSERTS)

   CONDUITS AND OUTLET BOXES, UNISTRUTS, AND OTHER INSERTS AND EMBEDMENTS, SHALL BE SECURED AND PROTECTED DURING PLACEMENT OF CONCRETE.
7. (ANCHOR BOLTS)

   ANCHOR BOLTS WILL BE FURNISHED ONLY UNDER SPECIFICATIONS APPROVED BY BECHTEL. ANCHOR BOLT SLEEVES NOT SHOWN ON THE DESIGN DRAWINGS SHALL NOT BE USED EXCEPT AS APPROVED BY BECHTEL. ANCHOR BOLTS SHALL NOT BE WELDED TO REINFORCEMENT.
8. FOR TYPICAL REINFORCED CONCRETE DETAILS, SEE DWGS 13-C-00A-004 & 13-C-00A-005.

1

AO-A-ZYD-187, REV 1

Palo Verde Nuclear Generating Station
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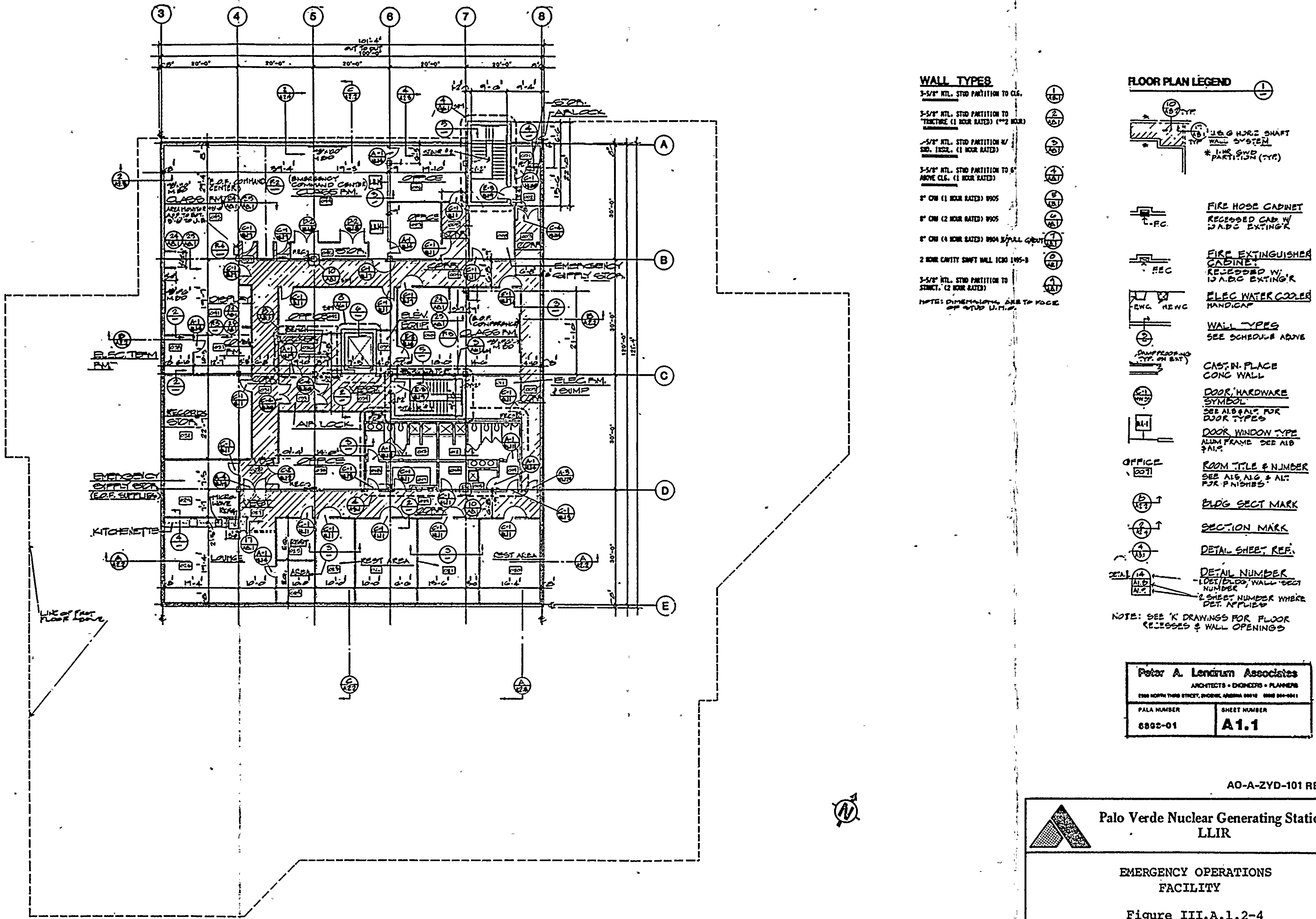
TECHNICAL SUPPORT CENTER

Figure III.A.1.2-3



Peter A. Lendrum Associates
ARCHITECTS • ENGINEERS • PLANNERS

PALA NUMBER	SHEET NUMBER
6802-01	A1.1

AO-A-ZYD-101 REV O

Palo Verde Nuclear Generating Station
LLIR

EMERGENCY OPERATIONS
FACILITY

Figure III.A.1.2-4

III.A.2 IMPROVING LICENSEE EMERGENCY PREPAREDNESS -- LONG-TERM

Position

Each nuclear facility shall upgrade its emergency plans to provide reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency. Specific criteria to meet this requirement is delineated in NUREG-0654 (FEMA-REP-1), "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparation in Support of Nuclear Power Plants."

PVNGS Evaluation

PVNGS response to this item is included in the evaluation of section III.A.1.1 requirements.

III.D RADIATION PROTECTION

III.D.1.1 INTEGRITY OF SYSTEMS OUTSIDE CONTAINMENT LIKELY TO CONTAIN RADIOACTIVE MATERIAL FOR PRESSURIZED-WATER REACTORS AND BOILING-WATER REACTORS

Position

Applicants shall 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 fluid 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. This program shall include periodic integrated leak tests at intervals not to exceed each refueling cycle.

PVNGS Evaluation

Refer to CESSAR Appendix B, Item III.D.1.1. In addition, a PVNGS design review was performed on the system below to assure that potential radioactive release paths following a serious transient or accident is reduced to as-low-as-reasonably achievable (ALARA) levels.

1 A. Shutdown Cooling System (SCS)

The existing design incorporates all-welded piping. Vent and drain lines throughout the system are capped when not in use. Relief valves on the system relieve to the equipment drain tank (a tank designed to accept radioactive fluids). The leakage from the LPSI pump seals and system valve stems is ALARA. Potential leakage from the SCS into the essential cooling water system (through the shutdown heat exchanger) can be detected during normal operation by installed radiation monitoring.

1 B. Containment Spray Recirculation System (CS)

The existing design incorporates all-welded piping. Vent and drain lines throughout the system are capped when not in use. Relief valves on the system (external to the containment) relieve to the equipment drain tank. The leakage from the CS pump seals and system valve stems is ALARA. Potential leakage during normal operation from the CS into the essential cooling water system (through the shutdown heat exchanger) can be detected by installed radiation monitoring.

C. <u>CVCS Charging and Letdown System</u> 1

The existing design incorporates all-welded piping. The letdown system is isolated upon CIAS and SIAS. Relief valves on the system relieve to the equipment drain tank.

The leakage from the CVCS charging pumps (positive displacement pumps) and other system equipment is ALARA as they are hard-piped to drains. The nuclear cooling water system is monitored for potential leakage from the CVCS through the letdown heat exchanger during normal operation.

D. <u>Sampling System</u> 1

The existing design of the normal sampling system incorporates "Swagelok" connections, however, the design will be upgraded to all-welded piping for sections which would come into contact with highly radioactive fluids. The system is isolated upon CIAS and SIAS. Relief valves relieve to the equipment drain tank. Leakage from the system is also minimized by the small size of the lines.

The post-accident sampling system will also be constructed of all-welded piping.

E. <u>High-Pressure Injection Recirculation (HPSI)</u>

The design incorporates all-welded piping. Relief valves on the system (external to the containment) relieve to the equipment drain tank. The vent and drain lines throughout the system are capped when not in use. Leakage from the HPSI pump seals and system valve stems is as-low-as reasonably achievable. Miniflow connections to the refueling water tank (RWT) are isolated upon Recirculation Actuation Signal (RAS). Manual cross over valves to the CVCS are normally locked shut.

1 F. <u>Waste Gas System</u>

The waste gas system is isolated from the containment upon CIAS. (The normal vent path from the reactor drain tank (RDT) and the reactor head vent system is isolated.) By design, the introduction of highly radioactive fluids to the system is precluded.

2 As part of the system testing program, each of the above systems is hydrostatically tested to 150% normal operating pressure per the requirements of ANSI B31.1, Summer 1976 Addendum for ANSI B31.1 piping systems, and to 125% normal operating pressure per the requirements of ASME Boiler & Pressure Vessel Code, Section III, 1977 Edition, for ASME piping systems.

PVNGS will institute a program to maintain leakage rates of systems outside containment to as low as practical which consists of the following:

1. High pressure safety injection system (recirculation portion only).

2. Low pressure safety injection system (shutdown cooling portion only).

3. Reactor coolant sampling system (post-accident sampling piping only).

4. Containment spray system.

5. Radioactive waste gas system (post-accident sampling return piping only). 2

6. Liquid radwaste system (post-accident sampling return piping only).

Systems excluded from the program: (They will not preclude any option of cooling the reactor core nor will they prevent the use of needed safety systems).

1. Radioactive liquid waste system, except as discussed above.

2. Radioactive waste gas system, except as discussed above. (The system is not required for use post-accident.)

3. Reactor coolant letdown system. (The system is not required to function post-accident. The plant can be

brought to a cold shutdown condition without the letdown system. The letdown system is isolated on SIAS and CIAS.)

4. Reactor coolant pump seal bleed-off system. (The system is not required to function outside containment post-accident. The seal bleed-off system is isolated outside containment on CIAS. The system remains isolated post-accident. If seal bleed-off is required post-accident, pressure in the seal bleed-off header will increase and the header relief valve will lift providing a flow path to the reactor drain tank.)

2 5. Charging System. (The charging system under post-accident conditions does not contain radioactive fluid since the letdown system is isolated as discussed in item 3 above. The charging system takes suction from the refueling water tank.)

6. Fuel Pool Cladding system (FPC). (The FPC is normally isolated from potentially highly contaminated systems by double, locked shut isolation valves.)

Immediate leak reduction measures: The program will consist of periodic monitoring of the systems during operation. Leaks will be identified and corrective maintenance performed.

1. All vent and drain lines will be capped to prevent release due to seal leakage.

2. The packing of all valves (except Kerotest which is a packless, stainless steel diaphragm valve) in the scoped liquid systems will be inspected for leakage or evidence of leakage such as boric acid accumulation. Maintenance will be performed on the packing of liquid system valves identified as requiring work.

3. The seals and packing on all pumps in the scoped liquid systems will be inspected for leakage or signs of leakage.

4. Valves, fittings, and compressor seals in the scoped gaseous systems will be checked for leakage. Maintenance will be performed on gas system valves and instrument fittings identified during leak tests as requiring work. 2

Leakage Measurement

System leakage will be periodically measured during the operational inservice program for the following systems:

1. High pressure safety injection system
2. Low pressure safety injection system
3. Charging system
4. Reactor coolant sampling system
5. Containment spray system
6. Radioactive waste gas system
7. Liquid radwaste system

Records of leakage measurement will be retained in station.

III.D.3.3 IMPROVED INPLANT IODINE INSTRUMENTATION UNDER ACCIDENT CONDITIONS

Position

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

(2) Each applicant for a fuel-loading license to be issued prior to January 1, 1981 shall provide the equipment, training, and procedures necessary to accurately determine the presence of airborne radioiodine in areas within the plant where plant personnel may be present during an accident.

PVNGS Evaluation

Prior to fuel load, procedures will be developed for determining airborne iodine concentration. Silver-Zeolite or charcoal
cartridges will be used in conjunction with a portable pump. 2
The cartridges will be removed to the counting laboratory for
gamma spectrum analysis. Charcoal cartridges will be purged
with clean bottled compressed air prior to counting. A low
background counting facility is available on the site. The 2
results of accurate airborne concentration can be obtained
within 15 minutes after collection of iodine on filtered
cartridges. Procedures will also define ALARA concepts for

removal, transport, and analysis of filter cartridges. There
2 will be at least three such portable airborne monitors avail-
able at each unit which meet the NUREG-0737 recommendations.

1 PVNGS response to this item is included in the evaluation of
section II.F.1 requirements.

III.D.3.4 CONTROL ROOM HABITABILITY REQUIREMENTS

Position

In accordance with Task Action Plan item III.D.3.4 and control room habitability, licensees shall assure that control room operators will be adequately protected against the effects of accidental release of toxic and radioactive gases and that the nuclear power plant can be safely operated or shut down under design basis accident conditions (Criterion 19, "Control Room," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50).

PVNGS Evaluation

Potential hazards in the vicinity of the site are discussed in FSAR Section 2.2. The operators in the control room are adequately protected from these hazards and the release of radioactive gases as discussed in FSAR Section 6.4. The required information provided below is in the format suggested by Attachment 1 to NUREG 0737 Section III.D.3.4.

INFORMATION REQUIRED FOR CONTROL-ROOM HABITABILITY EVALUATION

(1) Control-room mode of operation: automatic filtered recirculation with filtered makeup for pressurization for radiological accident isolation. Automatic filtered recirculation without makeup for chemical accident isolation. Manual smoke removal mode (operators alerted by smoke detector).

(2) Control-room characteristics:

(a) air volume control room:

1.61×10^5 ft^3

(b) control-room emergency zone (control room, critical files, kitchen, washroom, computer room, etc.):

140 ft. elevation, Control Bldg.

(c) control-room ventilation system schematic with normal and emergency air-flow rates:

see FSAR Figures 9.4-1 and 9.4-2

normal rate = 30,000 ft^3/min

emergency rate = 28,600 ft^3/min

(d) infiltration leakage rate:

170 ft^3/min

(e) high efficiency particulate air (HEPA) filter and charcoal adsorber efficiencies:

HEPA = 99.97% of 0.3 micron particles

charcoal > 95% of particulates and Iodines

(f) closest distance between containment and air intake:

150 ft, approximately

(g) layout of control room, air intakes, containment building, and chlorine, or other chemical storage facility with dimensions:

see FSAR Figures 1.2-7 and 6.4-1

(h) control-room shielding including radiation streaming from penetrations, doors, ducts, stairways, etc:

shielding study in progress

(i) automatic isolation capability-damper closing time, damper leakage and area:

closing time = 15 sec.

leakage = zero leakage (by bubble test)

area = 12.6 sq ft

(j) chlorine detectors or toxic gas (local or remote):

redundant in C.R. ventilation outside air intake plenum; smoke detectors alert operators for manual operation

(k) self-contained breathing apparatus availability (number), refer to FSAR Section 6.4.2.2.2 Item M. 2

(l) bottled air supply (hours supply), refer to FSAR Section 6.4.2.2.2 Item M. 2

(n) control-room personnel capacity (normal and emergency):

PVNGS response to items k, l, and n are undergoing review in light of the personnel manning requirements of section III.A.1.2.

(m) emergency food and potable water supply (how many days and how many people):

Presently, the PVNGS control room design has emergency food and water supply for 6 people for 7 days (within the closed control room). The commitment is being reviewed in response to NUREG 0696 TSC requirements.

(o) potassium iodide drug supply:

2 Sufficient potassium iodine will be maintained in a central location at the station to supply 6 persons for 7 days, as noted in FSAR Section 6.4.4.3.D.

(3) Onsite storage of chlorine and other hazardous chemicals:

NOTE: No onsite storage of liquid or gaseous chlorine. It is stored as sodium hypochlorite

2 (liquid).

(a) total amount and size of container:

hydrogen:

125,000 SCF @ 2200-2450 psi is stored in fourteen steel cylinders sulfuric acid:

55,000 gal. in five 11,000-gal. tanks

8,000 gal. in two 4,000-gal. tanks

50,000 gal. in two 25,000-gal. tanks

22,000 gal. in two 11,000-gal. tanks

carbon dioxide:

180 tons in four 7 ft. diameter tanks

(b) closest distance from control room air intake:

hydrogen: >600 ft (and obstructed)

sulfuric acid: closest is >500 ft.
most are >3000 ft.

carbon dioxide: >3000 ft.

(4) Offsite manufacturing, storage, or transportation facilities of hazardous chemicals

(a) identify facilities within a 5-mile radius; None

(b) distance from control room; N/A

(c) quantity of hazardous chemicals in one container; N/A

(d) frequency of hazardous chemical transportation traffic (truck, rail, and barge); N/A, >5 miles from site.

(5) Technical specifications (refer to standard technical specifications)

(a) chlorine detection system:

See FSAR Sections 7.4 and 7.5.

(b) control-room emergency filtration system including the capability to maintain the control-room pressurization at 1/8-in. water gauge, verification of isolation by test signals and damper closure times, and filter testing requirements:

Required procedures, tests, and administrative controls will be developed prior to PVNGS Unit 1 fuel load.