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SUBJECT: Forwards changes to FSAR Section 18 & other ref sections updating info in response to TMI Items II. B. 2, II. D. 3, II. F. 1 & III. D. 3. 4. Changes update descriptions to represent current plant conditions & will be included in next FSAR amend. S66 RG21

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December 19, 1986
ANPP-39455-JGH/JKR/98.05

Director of Nuclear Reactor Regulation
Attention: Mr. George W. Knighton, Project Director
PWR Project Directorate #7
Division of Pressurized Water Reactor Licensing - B
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: Palo Verde Nuclear Generating Station (PVNGS)
Unit 3
Docket No. STN 530
Changes to the FSAR Concerning TMI Related Sections
File: 86-D-005-419.05; 86-G-056-026

Dear Mr. Knighton:

Attached for your review on PVNGS Unit 3 are FSAR changes to Section 18 and other referenced sections. These changes involve updating information in response to TMI items II.B.2, II.D.3, II.F.1 and III.D.3.4.

These changes are justified because they update our descriptions to more accurately represent the current condition of the plant. Many of the changes are editorial in nature.

For PVNGS Units 1 and 2, safety evaluations have been completed for implementation of these changes in accordance with the requirements of 10 CFR 50.59. The safety reviews and evaluations have determined that there are no unreviewed safety questions involved with the changes. These changes will be included in the next FSAR amendment.

If you have any questions, please contact Mr. W. F. Quinn of my staff.

Very truly yours

J G Haynes

J. G. Haynes
Vice President
Nuclear Production

JGH/JKR/rw
Attachment

cc: O. M. De Michele
E. E. Van Brunt, Jr.
E. A. Licitra
R. P. Zimmerman
A. C. Gehr

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18.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

A radiation and shielding-design review of the spaces around systems that may, as a result of an accident, contain highly radioactive materials has been conducted.

General design considerations to keep post-accident exposures ALARA are described in section 12.1.2.4. A summary of the shielding design review results is provided in section ~~12.3.2.1~~ 12.3.2.3, and a description of the source terms used

in the post-accident shielding review is provided in section 12.2.3. Post-accident radiation zones are discussed in section 12.3.1.3, and presented as figures 12.3-25 through 12.3-45.

The qualifications of safety-related equipment is provided in section 3.11.5.2.

The function, operation and design of the radiation monitoring system are described in section 11.5.

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would be the radwaste building ventilation systems which are provided with gaseous process monitors to monitor for leakage from the waste gas compressors and waste gas decay tank valves.

B. Areas in which the new and spent fuel is received and stored, specifically the containment and fuel building, are provided with detectors which indicate and alarm in the presence of abnormal radiation levels.

C. The location of each ~~area radiation detector is indicated on the radiation zoning and access control~~ monitor location drawings, figures 11.5-2 ~~12.3-1~~ through 11.5-5. ~~12.3-20.~~

11.5.1.2 Postulated Accidents

The process, effluent, and area monitoring systems, collectively referred to as the radiation monitoring system (RMS), are designed to perform the following functions in order to meet the requirements of 10CFR50, 10CFR100, and follow the recommendations of NUREG 0737 and NRC Regulatory Guides 1.13, 1.97 and 8.12 for postulated accidents: 5

A. Provide the capability to alarm and initiate containment purge isolation in the presence of high airborne radioactivity within the containment which could potentially cause an offsite dose in excess of 10CFR100 limits.

B. Provide the capability to alarm and initiate isolation of the fuel building from the normal ventilation system and actuation of fuel building essential ventilation in the unlikely event of a fuel handling accident in the fuel building.

C. Provide the capability to alarm and initiate isolation of the control room normal ventilation system and actuation of control room essential filtration in the

as figures 12.3-25 through 12.3-35.

in figures 12.3-36 through 12.3-45

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12.1.2.4 General Design Considerations to Keep Post-Accident Exposures ALARA

The facility layout will assist in keeping occupational 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 ~~in Section II.B.2 of the LLIR~~. Zone maps for the hypothetical condition of a LOCA with an intact primary but with a degraded core are ~~also~~ provided ~~in Section II.B.2 of the LLIR~~. 5 A discussion of the source terms for these events is provided in section 12.2.3. The dose rates projected for these two sets of drawings do not assume decay beyond that corresponding to the onset of recirculation. Even so, virtually unrestricted access will be permitted within the control and diesel generator buildings, as well as portions of the upper floor of the auxiliary building (such as the area of the operational support center).

To provide sampling capability with exposures kept ALARA, PVNGS will incorporate a post-accident sampling system that meets the requirements of NUREG 0737 and Regulatory Guide 1.97, 15 Revision 2 as described in section 9.3.2 and 18.II.B.3 ~~LLIR Item II.B.3~~.

The only other area where post accident access will ~~might~~ be required is to the hydrogen monitors/recombiners. Projected dose rates without the recombiners in operation but at the onset of recirculation 5 are expected to be approximately 10 to 30 rem/hr (sump recirculation). As the recombiners ~~do not~~ have to be installed ~~until~~

Refer to section 9.3 for a detailed description of the PVNGS nuclear sampling system.

ENSURING THAT OCCUPATIONAL RADIATION EXPOSURES ARE AS LOW AS IS REASONABLE ACHIEVABLE (ALARA)

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within ~~at least~~ 72 hours ~~but no earlier than 48 hours~~ the area will not be accessed until ~~one day~~ after the DBA, dose rates ~~will drop~~ have dropped 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 would not be consistent with hydrogen generation inside the containment. If hydrogen generation were postulated, 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 feet (refer to section 9.4).
5 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 feet in the auxiliary building (refer to section 9.4). The use of non-ESF filtration is acceptable since there are no recirculation components above elevation 100 feet. 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.

Therefore, considering direct and airborne sources, access can be provided to those vital areas necessary for control of the plant and personnel exposures will meet GDC 19 and NUREG 0737 limits.

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V = free volume of the region in which the leak occurs in cm^3

$C_i(t)$ = airborne concentration of the i^{th} radioisotope at time t in $\mu Ci/cm^3$ in the applicable region

From the above equation, it is evident that the peak or equilibrium concentration, $C_{Eq,i}$ of the i^{th} radioisotope in the applicable region will be given by the following expression:

$$C_{Eq,i} = (L\ R)_i\ A_i\ (P\ F)_i\ /\ (V\ \lambda_{Ti}) \quad (2)$$

With high exhaust rates, this peak concentration will be reached within a few hours.

12.2.3 SOURCES USED IN NUREG 0737 POST-ACCIDENT SHIELDING REVIEW

The post-accident shielding review described in section 12.1.2.4 considered two LOCA events. 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: 5 12

A. Source A: Containment airborne: 100% noble gases, 25% iodines - see table 12.2-8.

B. Source B: Reactor coolant: 100% noble gases, 50% halogens, 1% solids - see table 12.2-9.

C. Source C: Containment sump: 50% halogens, 1% solids - see table 12.2-10. 5

Volumes used for each source were:

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

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

used initial core releases equivalent to those recommended in Regulatory Guides 1.4 and 1.7, and Standard Review Plan 15.6.5, and

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C. 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 12.2-11. The results of the shield review are presented in section 12.3.1.3.

Decay curves, normalized to initial time equals zero, were developed for the sources as an aid in developing post-accident access plans. These curves are presented as figures 12.2-1 (source A), 12.2-2 (source B), and 12.2-3 (source C).

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TABLE 12.2-9

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 12.2-11 ~~II.B.2-4~~

Systems ~~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.

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12.3.1.2 Design Radiation Zoning and Access Control 12

Access into the plant structures and plant yard areas is regulated and controlled. 12

Plant areas are categorized as design radiation zones according to expected maximum radiation levels and anticipated personnel occupancy with consideration given toward maintaining personnel exposures as low as is reasonably achievable and within the standards of 10CFR20. Each design radiation zone defines the radiation level range to which the aggregate of contributing sources must be attenuated by shielding. Each room, corridor, and pipeway of every plant building is evaluated for potential radiation sources during normal, shutdown, spent resin transfer, and emergency operations; for maintenance occupancy requirements; for general access requirements; and for material exposure limits to determine appropriate zoning. The design radiation zone categories employed, and their descriptions are given in table 12.1-1. The specific design zoning for each plant area is shown in figures 12.3-1 through 12.3-20. Frequently accessed areas, e.g., corridors, are shielded for design radiation Zone 1 or Zone 2 access. 12

The control of entry or exit of plant operating personnel to controlled access areas, and procedures employed to ensure that radiation levels and allowable working time are within the limits prescribed by 10CFR20 is described in section 12.5.

12.3.1.3 Radiation Zones - Post-Accident 12

Radiation zone maps were developed in accordance with NUREG 0737 to review potential access throughout the plant post accident. (Refer to PVNGS LLIR Section II.B.2.) Two events were considered as noted in section 12.1.2.4 using the sources described in section 12.2.3. The events were a LOCA with sump recirculation and an intact primary-degraded core LOCA. Estimated radiation levels in vital areas were based on radiation sources from the post-accident

INSERT (A)

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INSERT (A) TO SECTION 12.3.1.3 (Pg 12.3-11)

¶ Radiation zone maps were developed in accordance with the NUREG 0737 to review potential access throughout the plant post accident period. The facility layout ~~will~~ assists in keeping occupational 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 12.3-25 ~~II.B.2-4~~ through 12.3-35. ~~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 12.3-36 ~~II.B.2-14~~ through 12.3-45 ~~II.B.2-23.~~ The source terms correspond to those noted in section 12.2.3. ~~II.B.2A.~~ The dose rates projected for these two sets of drawings do 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 support center) and the lower levels of the control building. Continuous occupancy will be permitted in the control 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.

operation of the following systems: containment, safety injection/shutdown cooling/containment spray, chemical and volume control system (up to purification filter inlet), post-accident sampling, and hydrogen recombiners. The gaseous
5 radwaste system will not be used post-accident. Palo Verde does not have a standby gas treatment system or equivalent.

As a result of this review, piping used for ~~backup~~ grab sampling in the hot lab sample room area ~~will be~~ lead ~~wrapped~~ to keep operator doses ALARA. ~~Note that the primary post-accident sampling method is a remote, automatic system.~~

12.3.2 SHIELDING

The bases for the nuclear radiation shielding and the shielding configurations are discussed in this section.

12.3.2.1 Design Objectives

The basic objective of the plant radiation shielding, in conjunction with a program of controlled personnel access to, and
12 occupancy of, radiation areas, is to reduce personnel and population exposures to levels that are within the dose regulations of 10CFR20 and 10CFR50 and are as low as is reasonably achievable (ALARA). Shielding and equipment layout and design are considered in ensuring that exposures are kept ALARA during anticipated personnel activities in areas of the plant containing radioactive materials, utilizing the design recommendations given in Regulatory Guide 8.8, Paragraph C.2, where practical.

Four plant conditions are considered in the nuclear radiation shielding design: normal, full-power operation; shutdown; spent resin transfer; and emergency operations (for required access to safety-related equipment).

The shielding design objectives for the plant during normal operation, including anticipated operational occurrences; for shutdown operations; and emergency operations are:

A. To ensure that radiation exposure to plant operating personnel, contractors, administrators, visitors, and proximate site boundary occupants are ALARA and within the limits of 10CFR20.

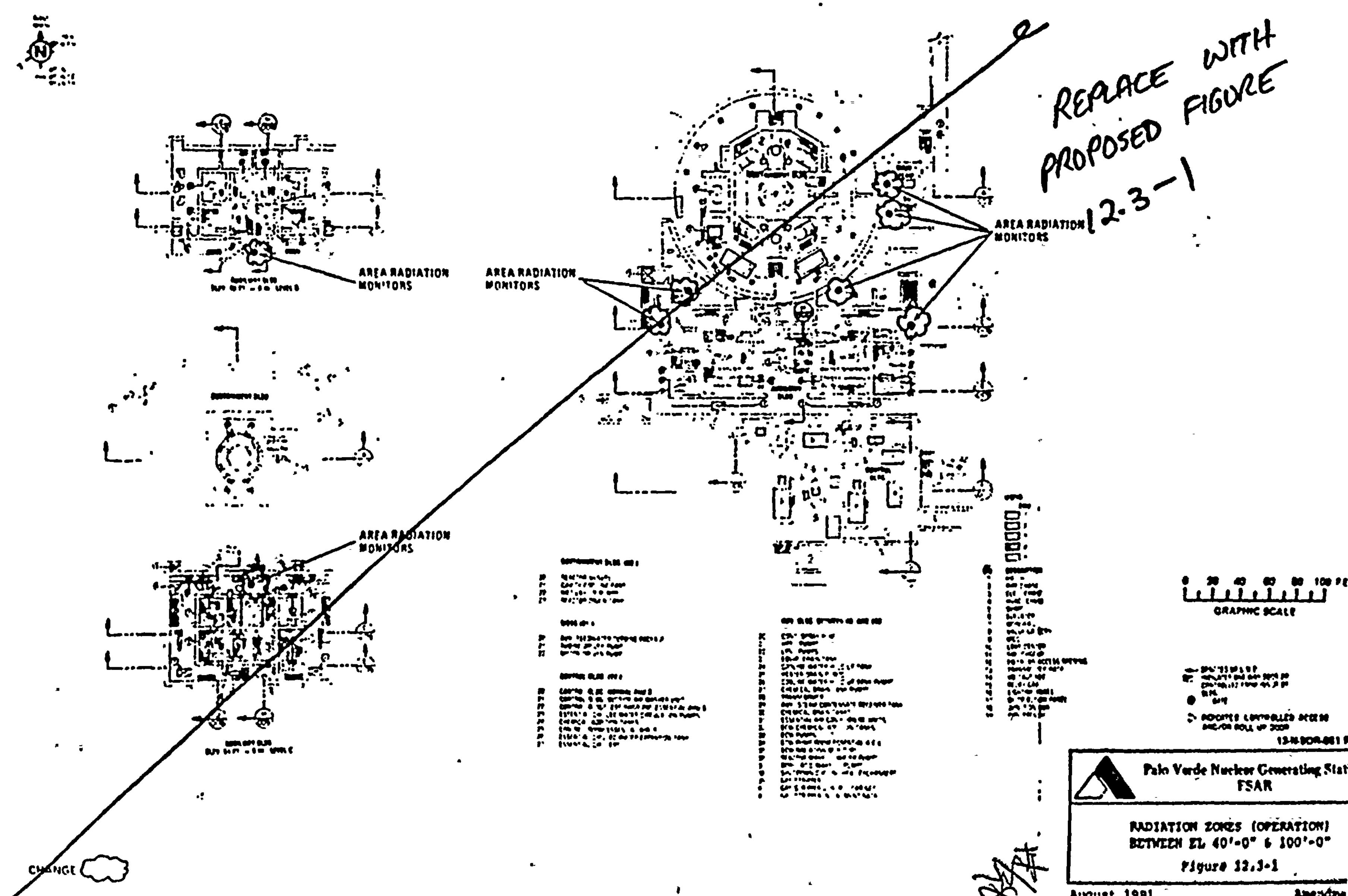
B. To assure sufficient personnel access and occupancy time to allow normal anticipated maintenance, inspection, and safety-related operations required for each plant equipment and instrumentation area.

C. To reduce potential equipment neutron activation and mitigate the possibility of radiation damage to materials.

INSERT (B) TO SECTION 12.3.2.1, (PAGE 12.3-12)

¶ An analysis of the PVNGS shielding design was performed to determine if TMI level source strengths would inhibit maintenance access or violate 10 CFR 50, Appendix A, General Design Criterion 19 (GDC 19). The review demonstrated that personnel radiation exposures in vital areas, during post-accident activities will meet the criteria of NUREG 0737 and the GDC-19 design basis.

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



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RADIATION ZONES (OPERATION)
BETWEEN EL 40'-0" & 100'-0"

Figure 12.3-1

August 1981 Amendme

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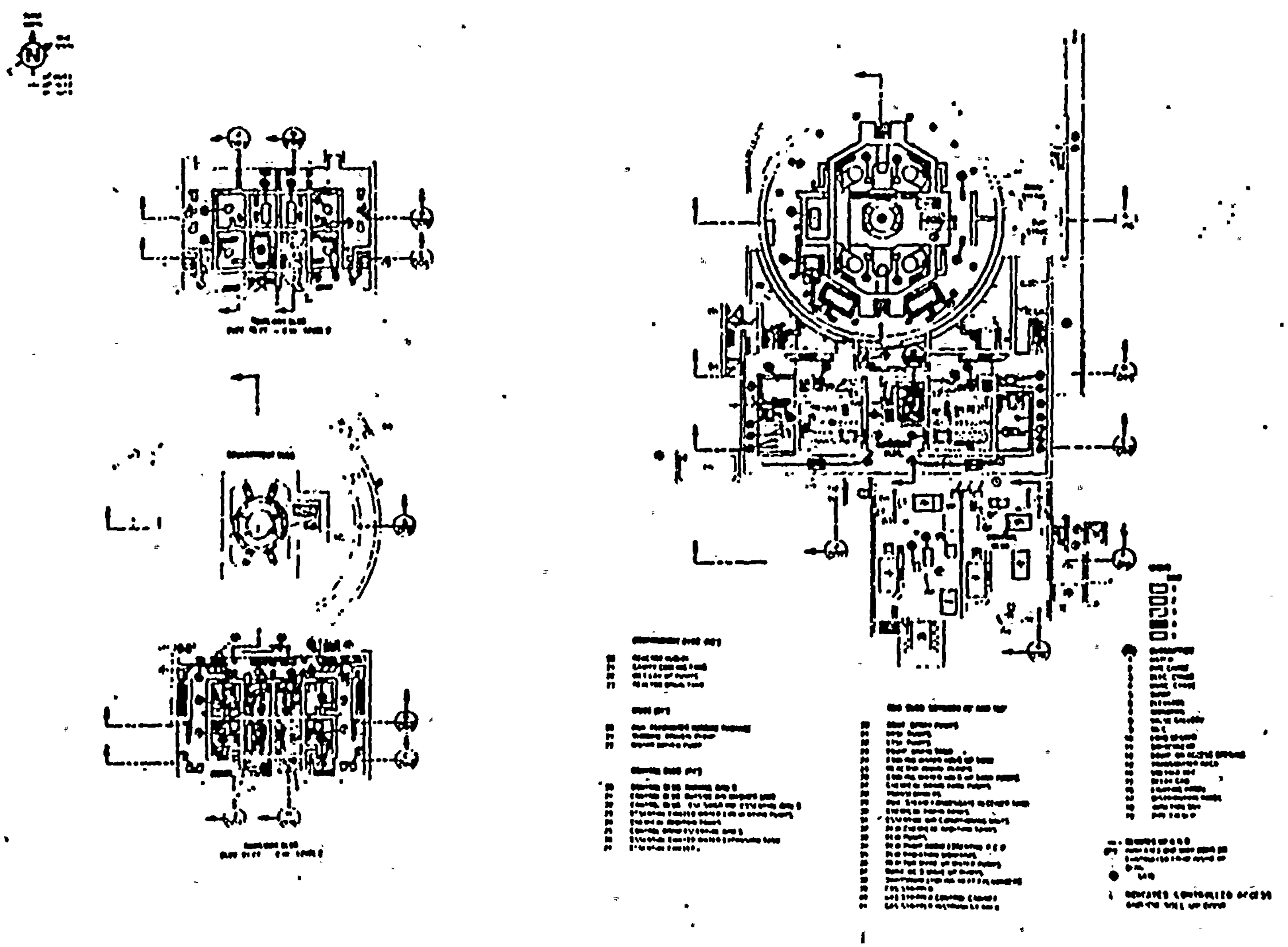

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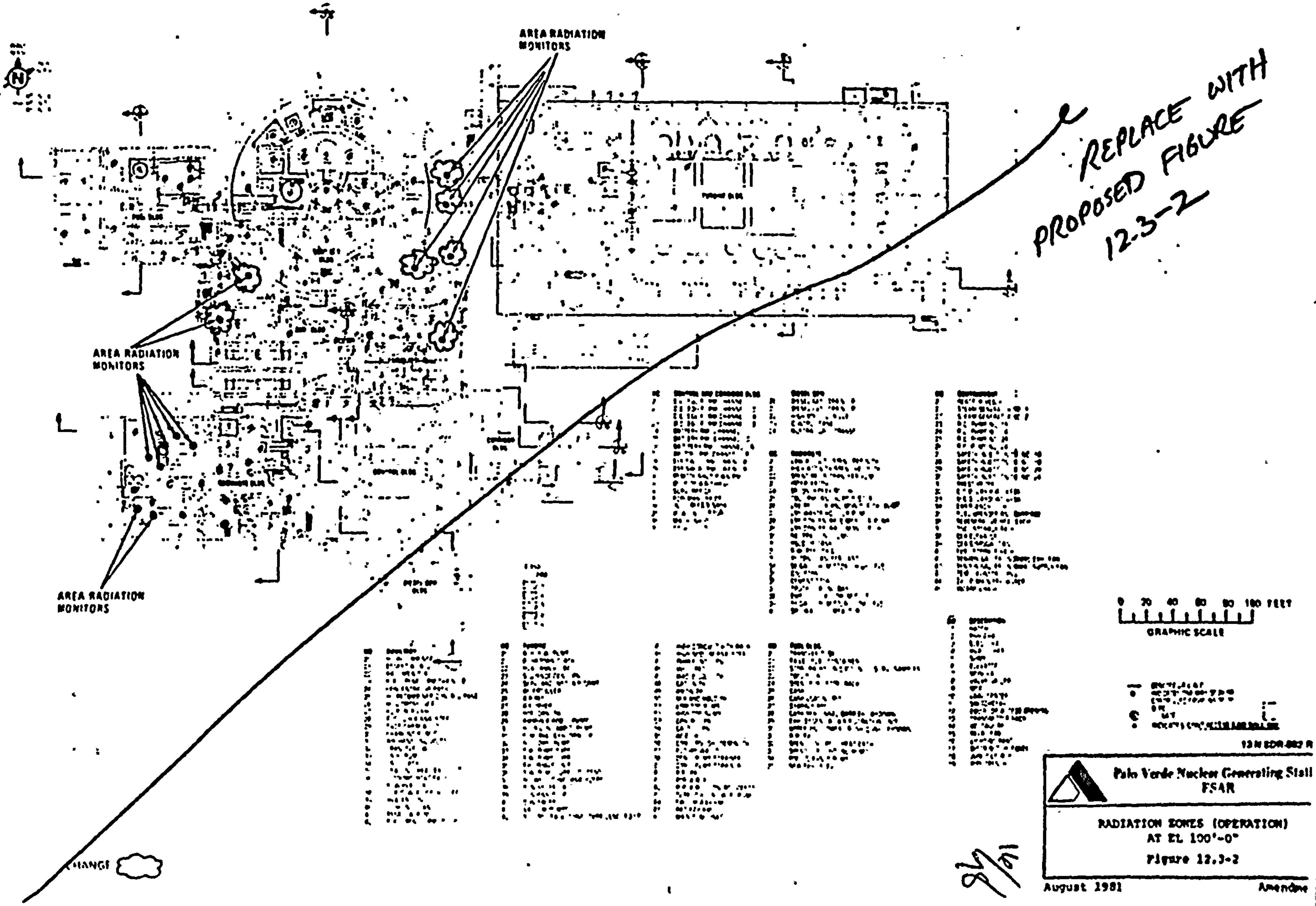
RADIATION ZONES (OPERATION)
BETWEEN EL. 40'-0" AND EL. 100'-0"

Figure 12.3-1



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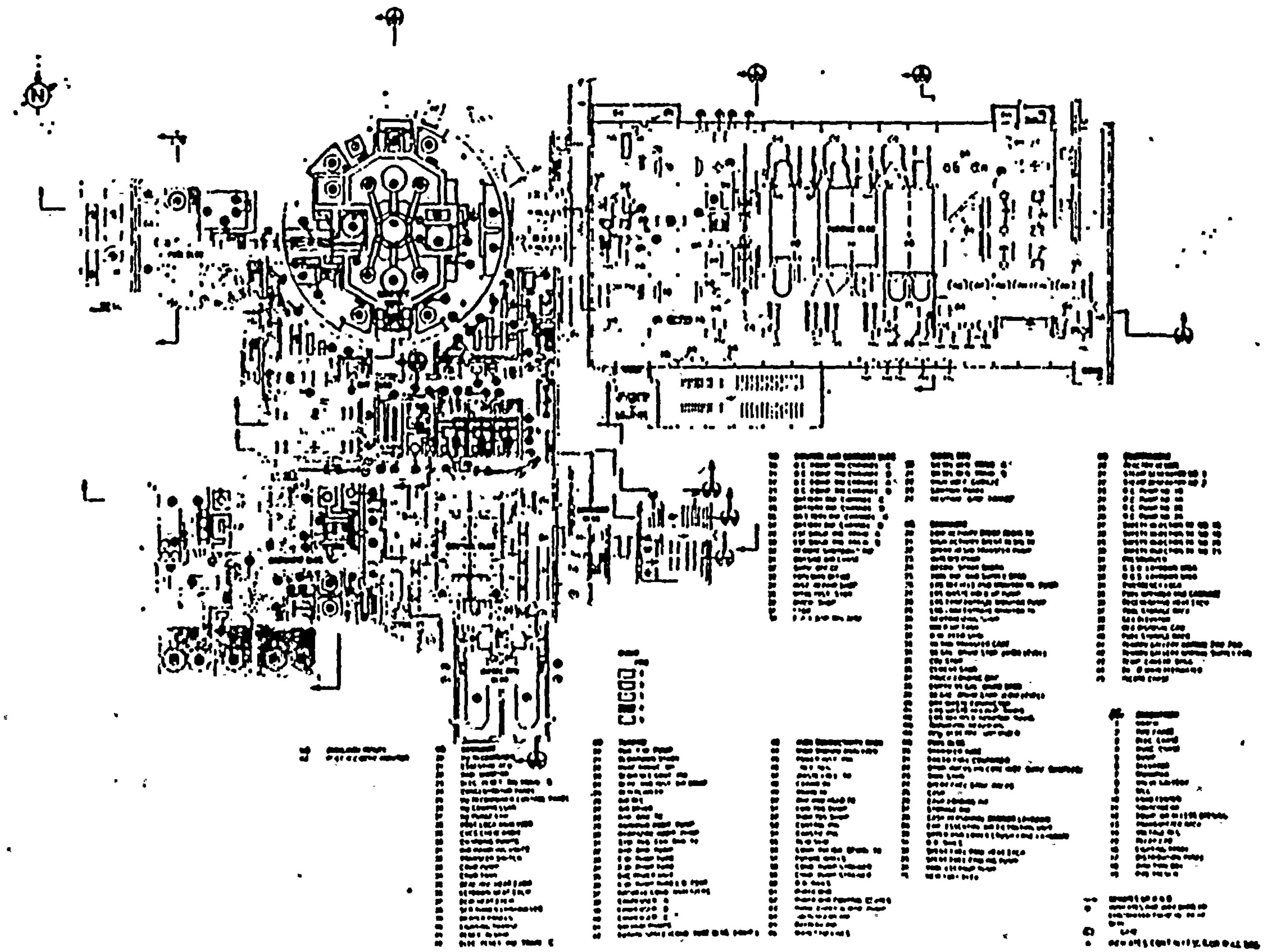
RADIATION ZONES (OPERATION)
AT EL 100'-0"

Figure 12.3-2

August 1981 Amendme



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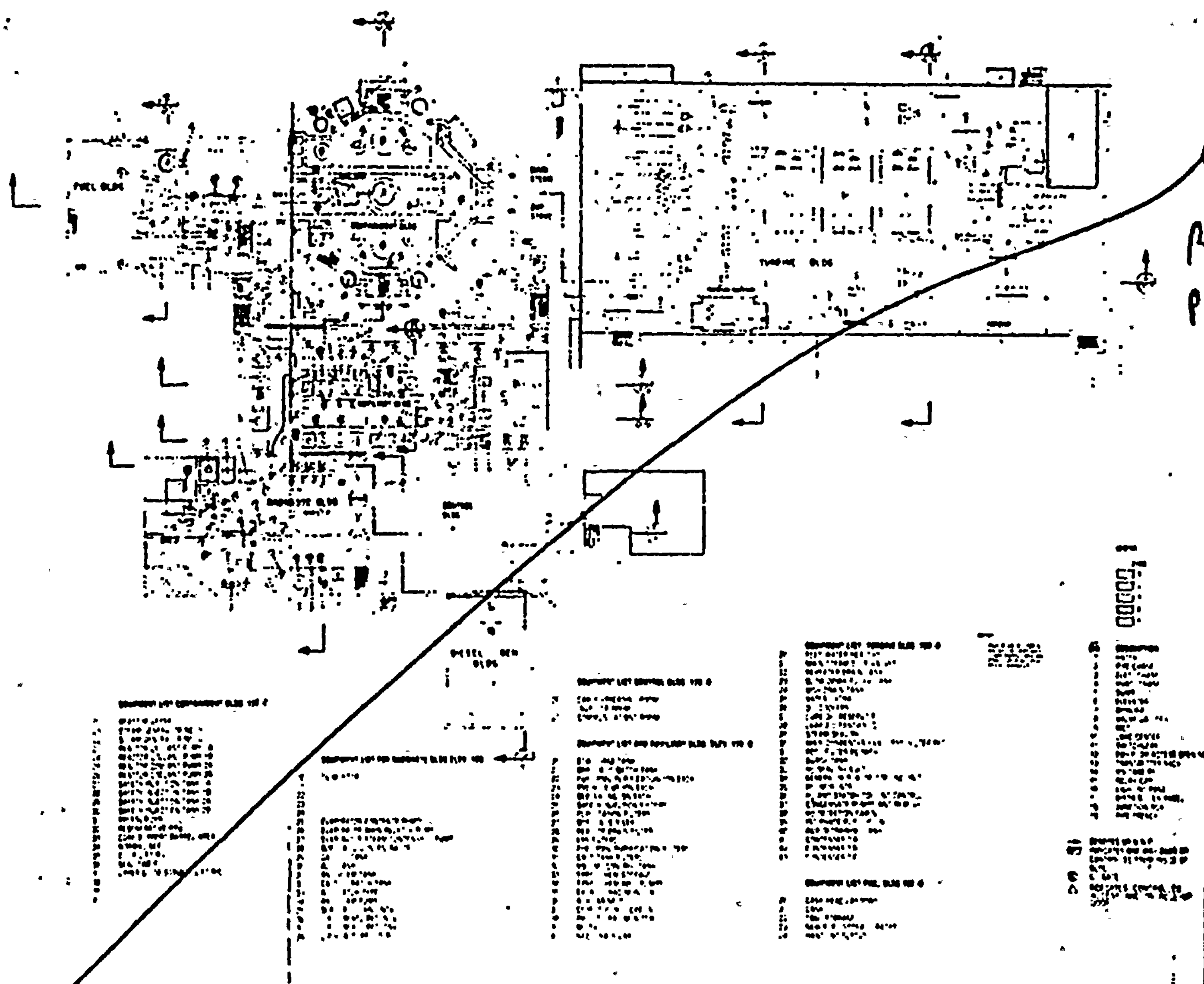
RADIATION ZONES (OPERATION)
AT EL. 100'-0"

Figure 12.3.1-2



REPLACE WITH PROPOSED FIGURE 12.3-3

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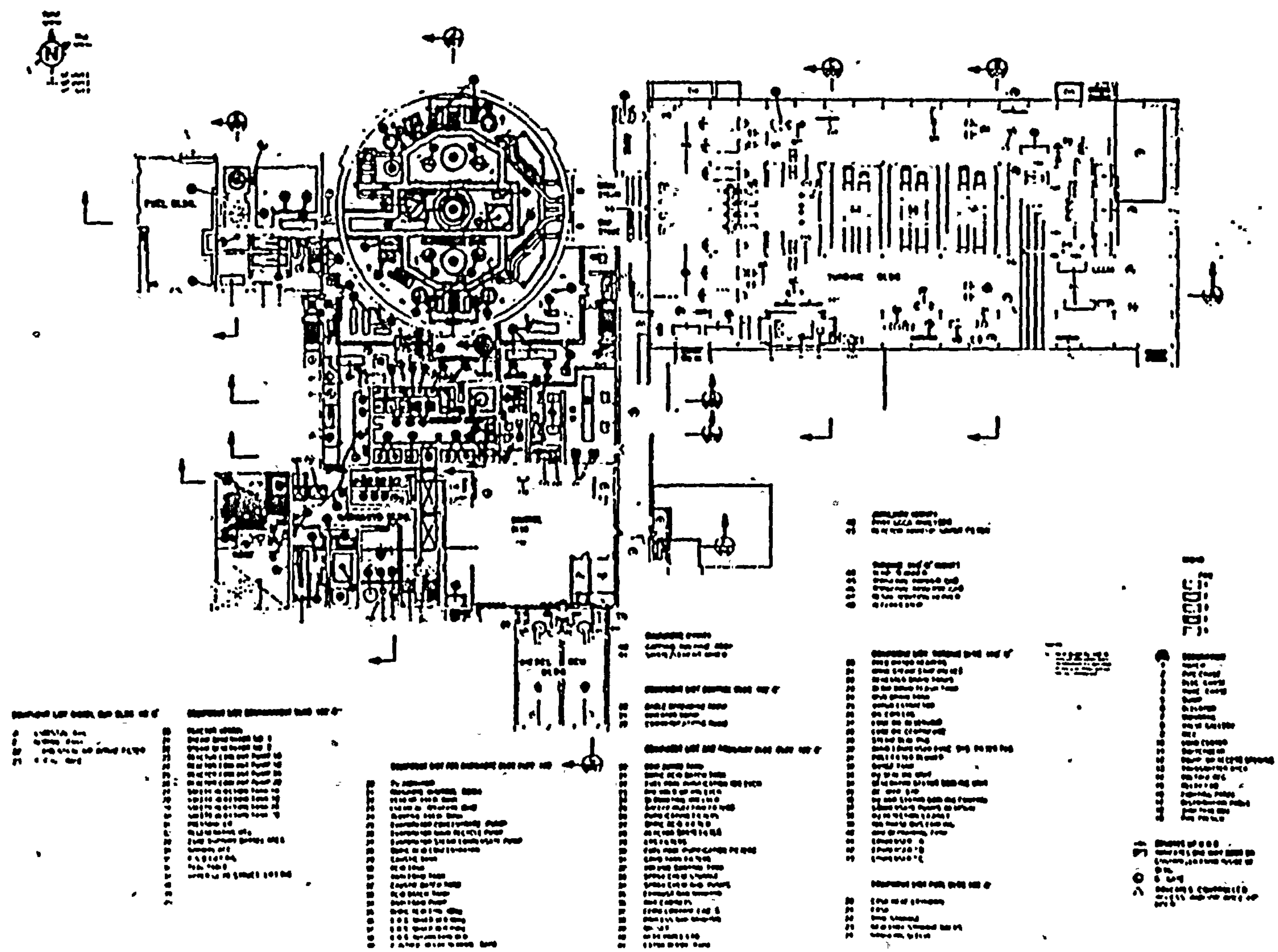
RADIATION ZONES (OPERATION)
BETWEEN EL 120'-0" & 140'-0"

Figure 12.3-3

April 1983

Amendment



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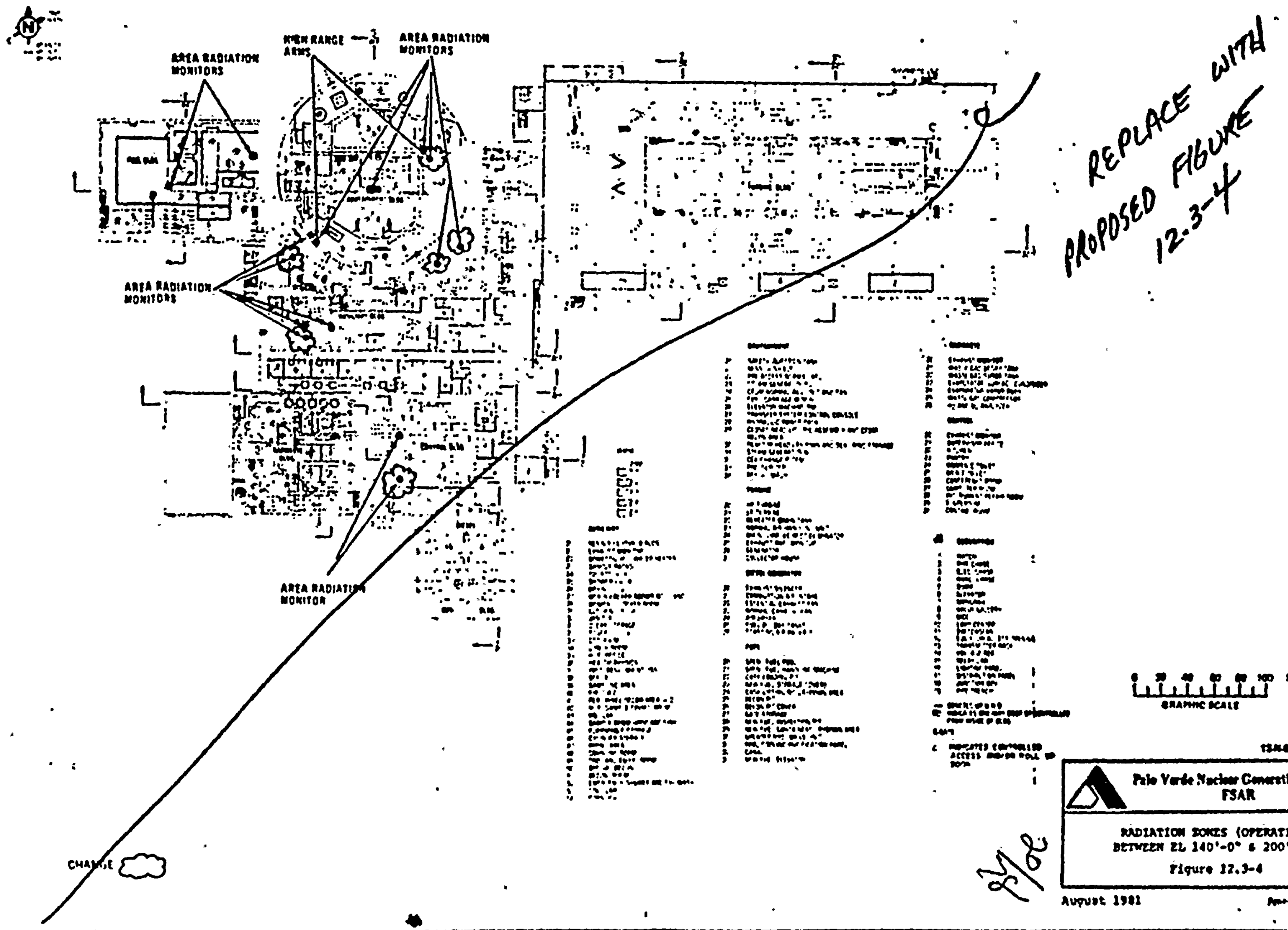
RADIATION ZONES (OPERATION)
BETWEEN EL. 120'-0" AND EL. 140'-0"

Figure 12.1-1



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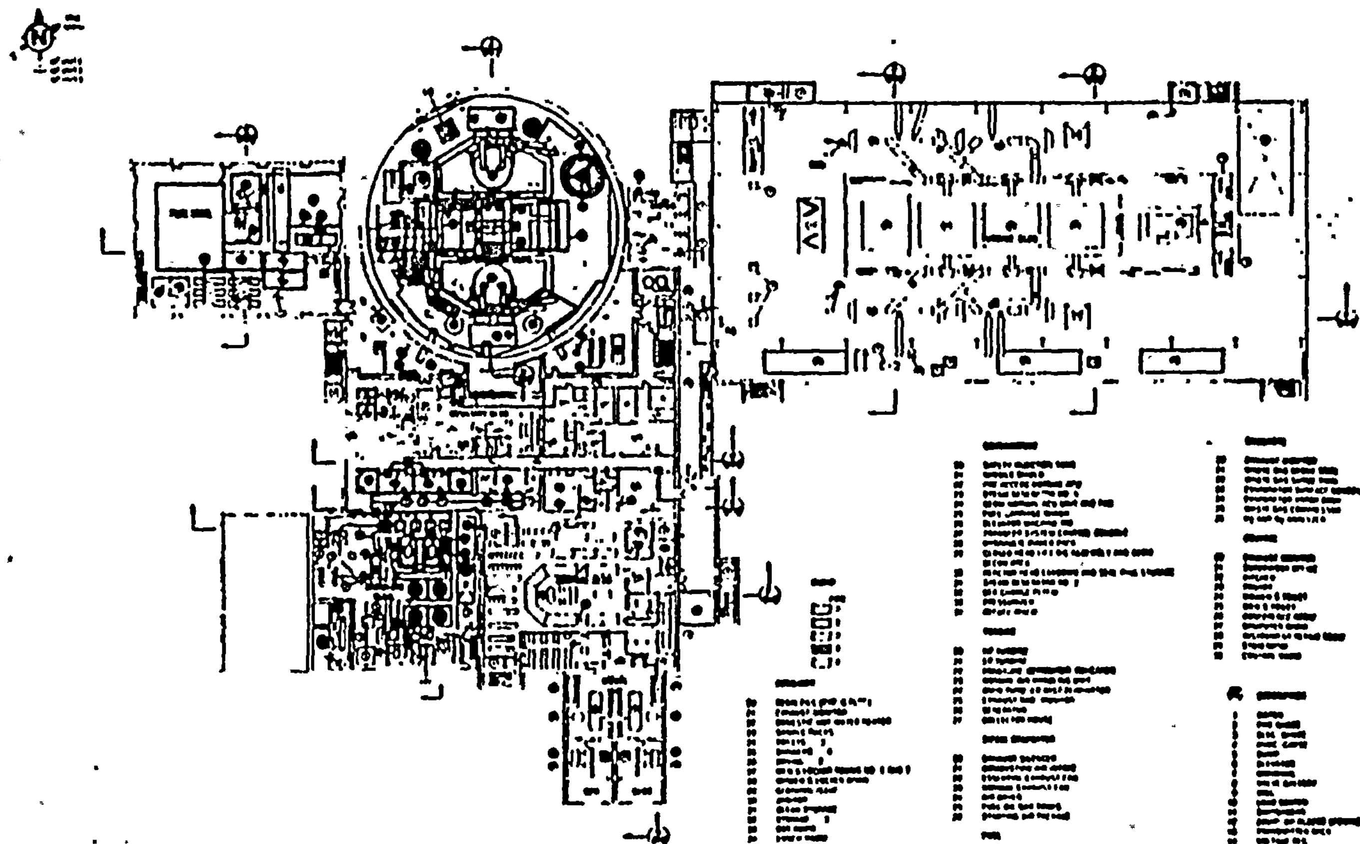
RADIATION ZONES (OPERATI
BETWEEN EL 140'-0" & 200'

Figure 12.3-4

August 1981



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Palo Verde Nuclear Generating Statio[n]
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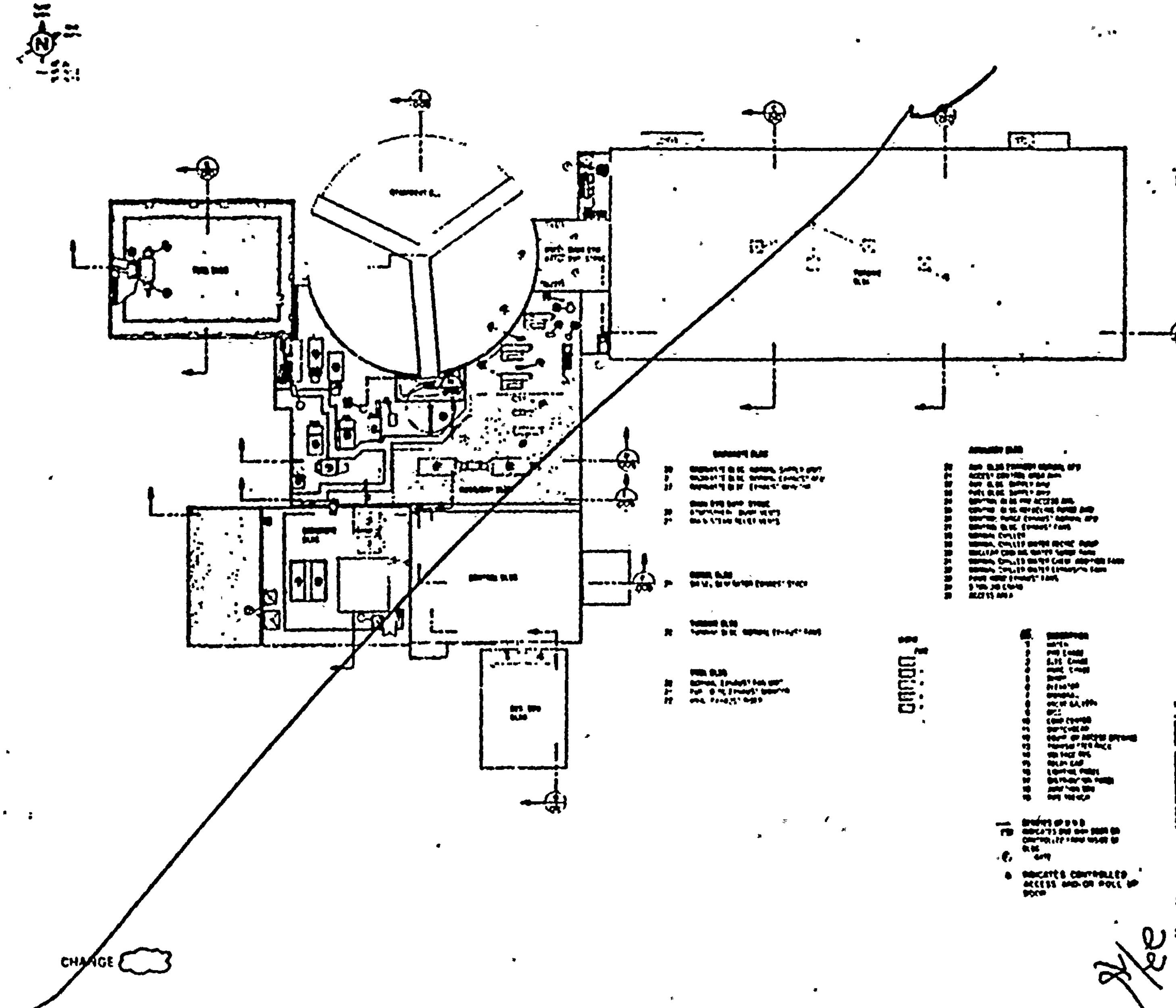
RADIATION ZONES (OPERATION)
BETWEEN EL. 140'-0" AND EL. 200'-0"

Figure 12.3-4

REPLACE WITH PROPOSED FIGURE 12.3-5



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RADIATION ZONES (OP
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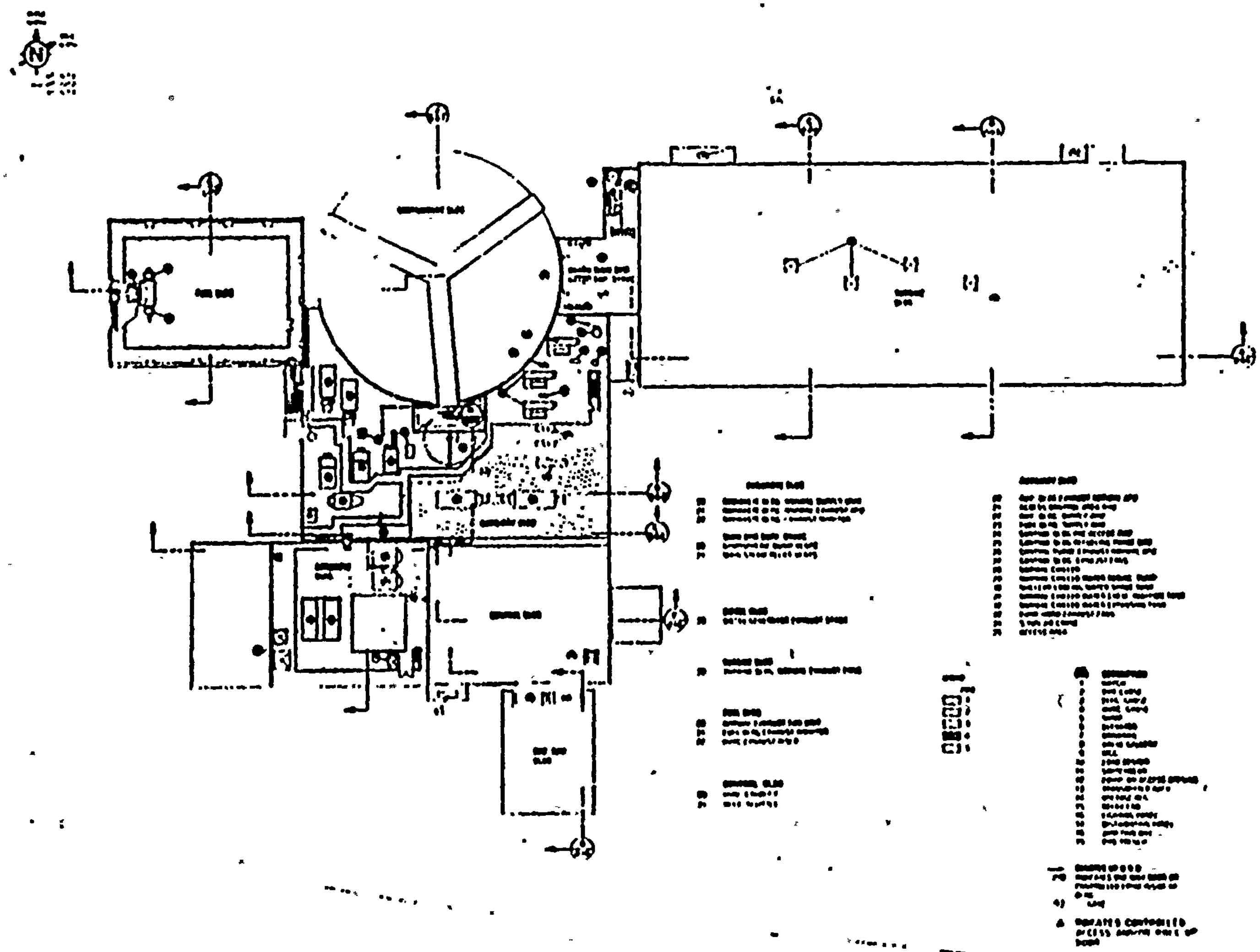
Figure 12.3-

March 1980



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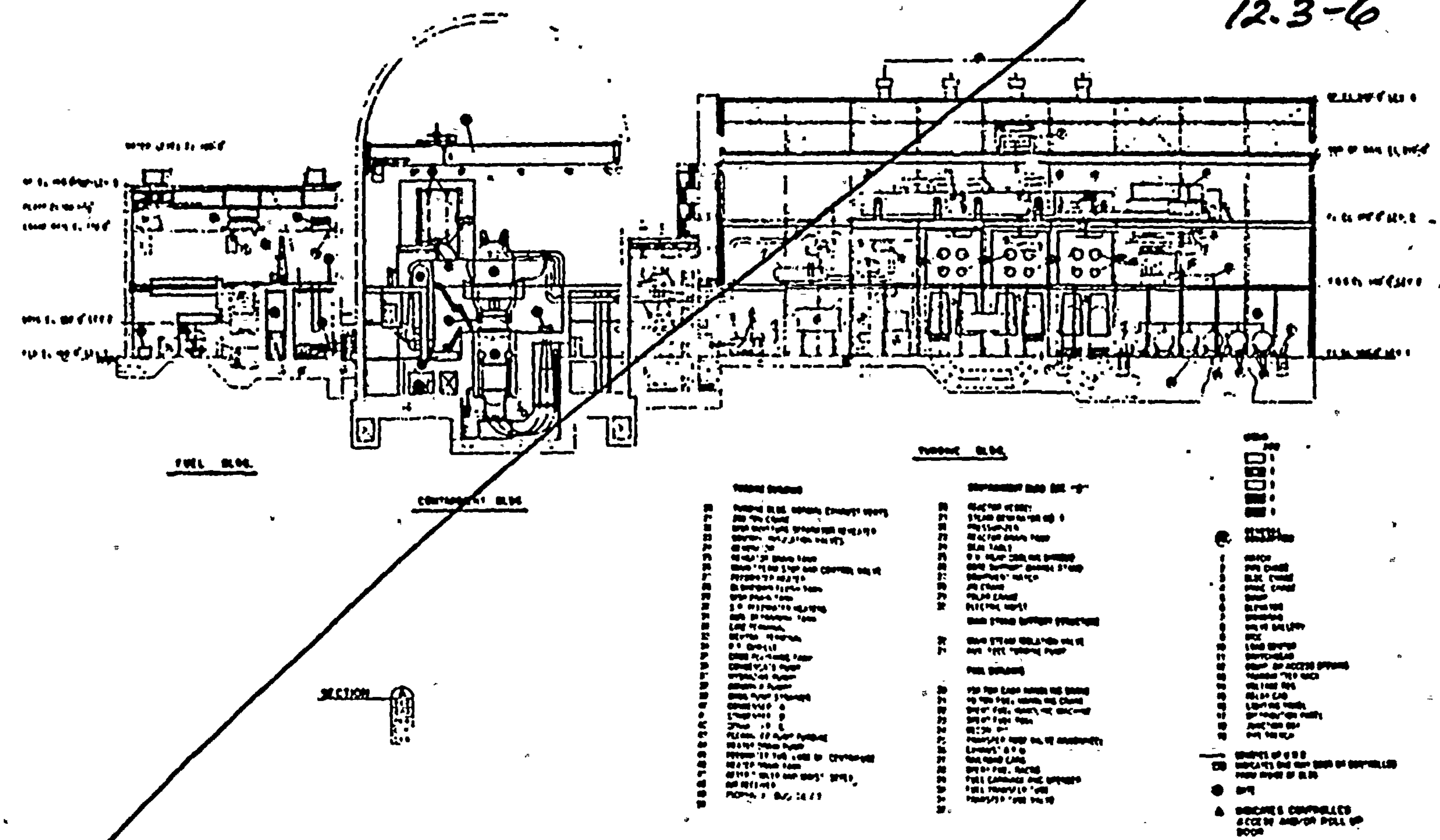
RADIATION ZONES (OPERATION)
AT ROOF ELEVATION

Figure 12.3-5

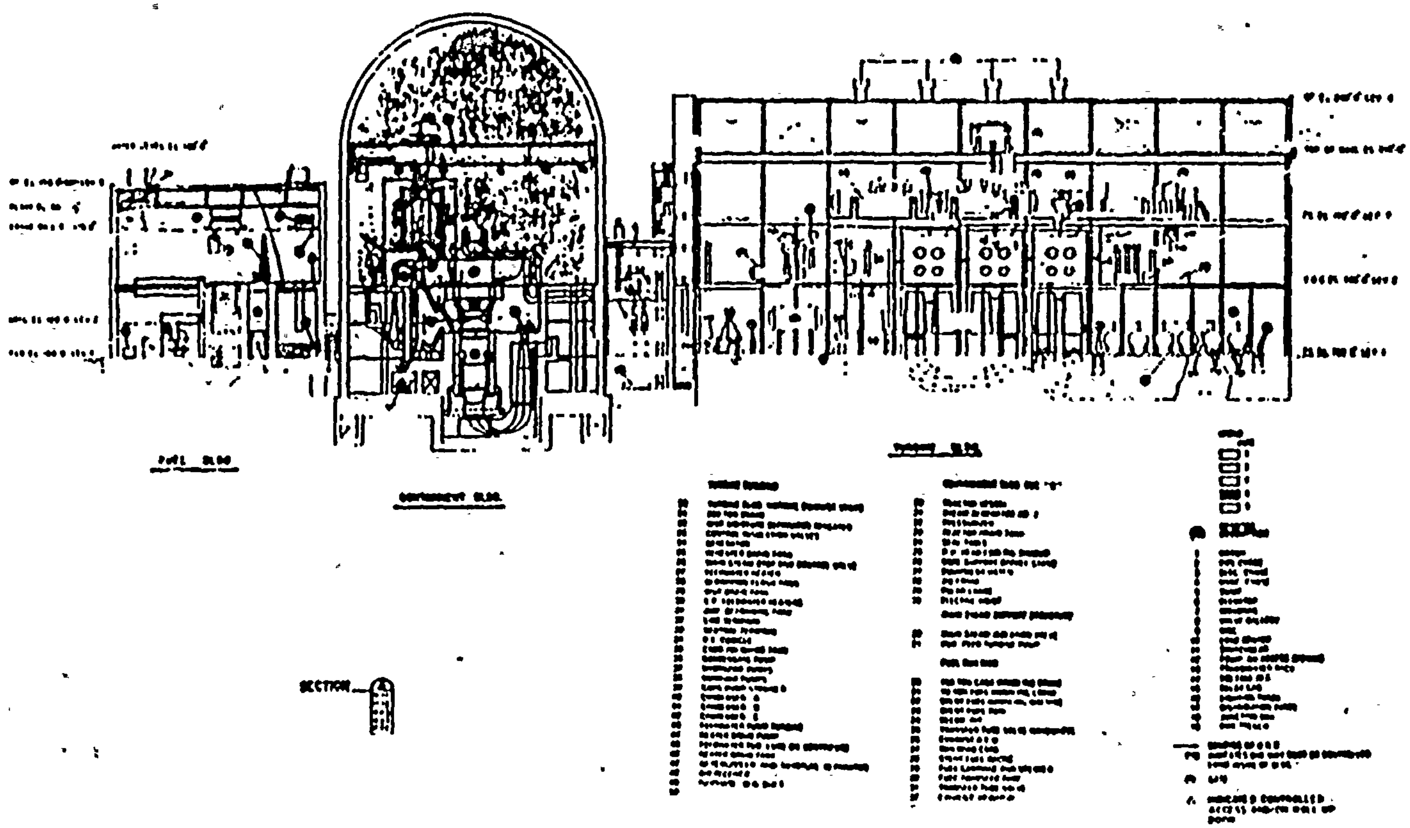
REPLACE WITH PROPOSED FIGURE 12.3-6



Palo Verde Nuclear Generating Station
FSAR

RADIATION ZONES (OPERATION)
SECTION A

Figure 12.3-6



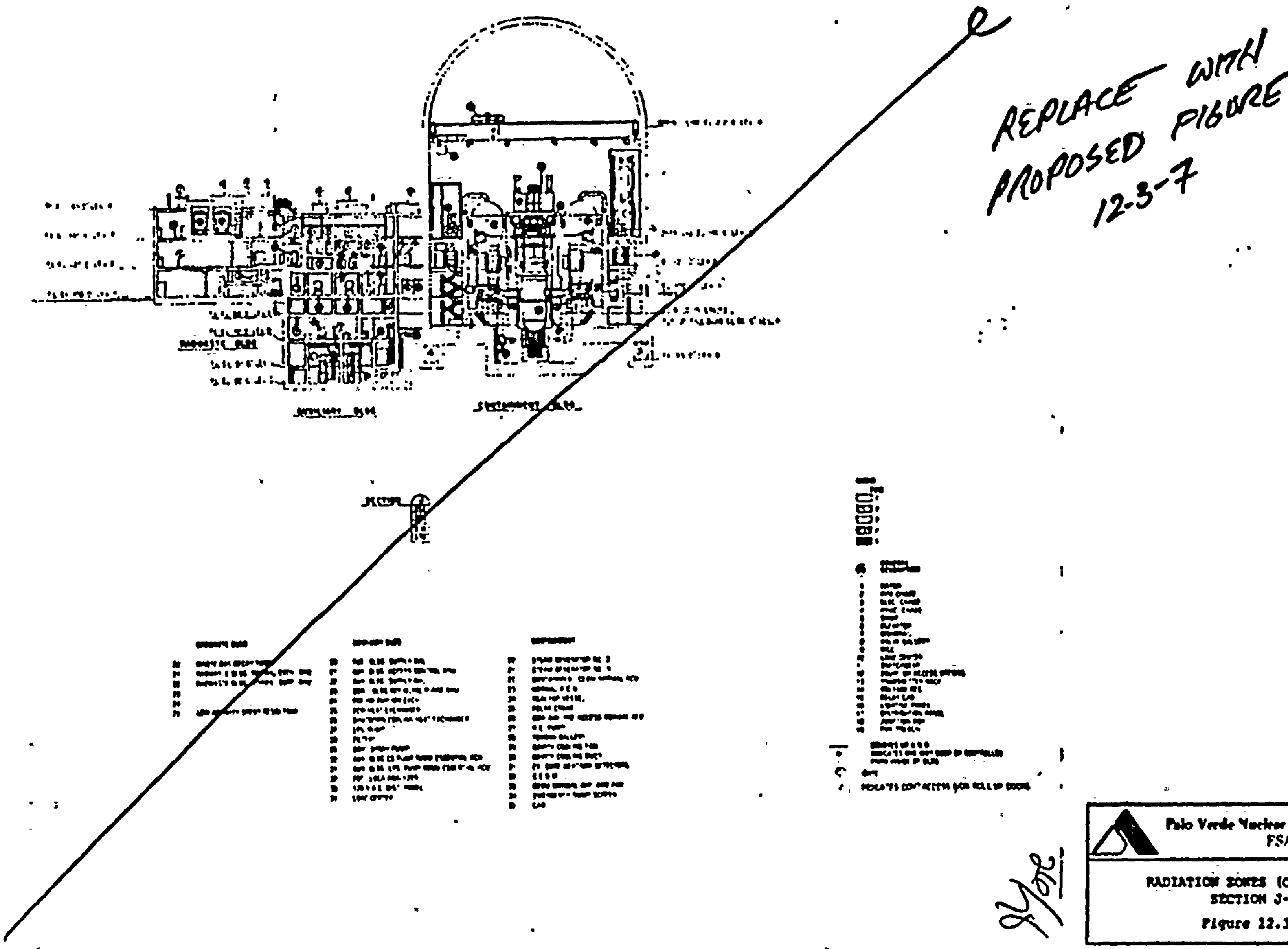
PROPOSED

13-N-RAR-006, REV

Palo Verde Nuclear Generating Station
FSAR

RADIATION ZONES (OPERATION)
SECTION A-A

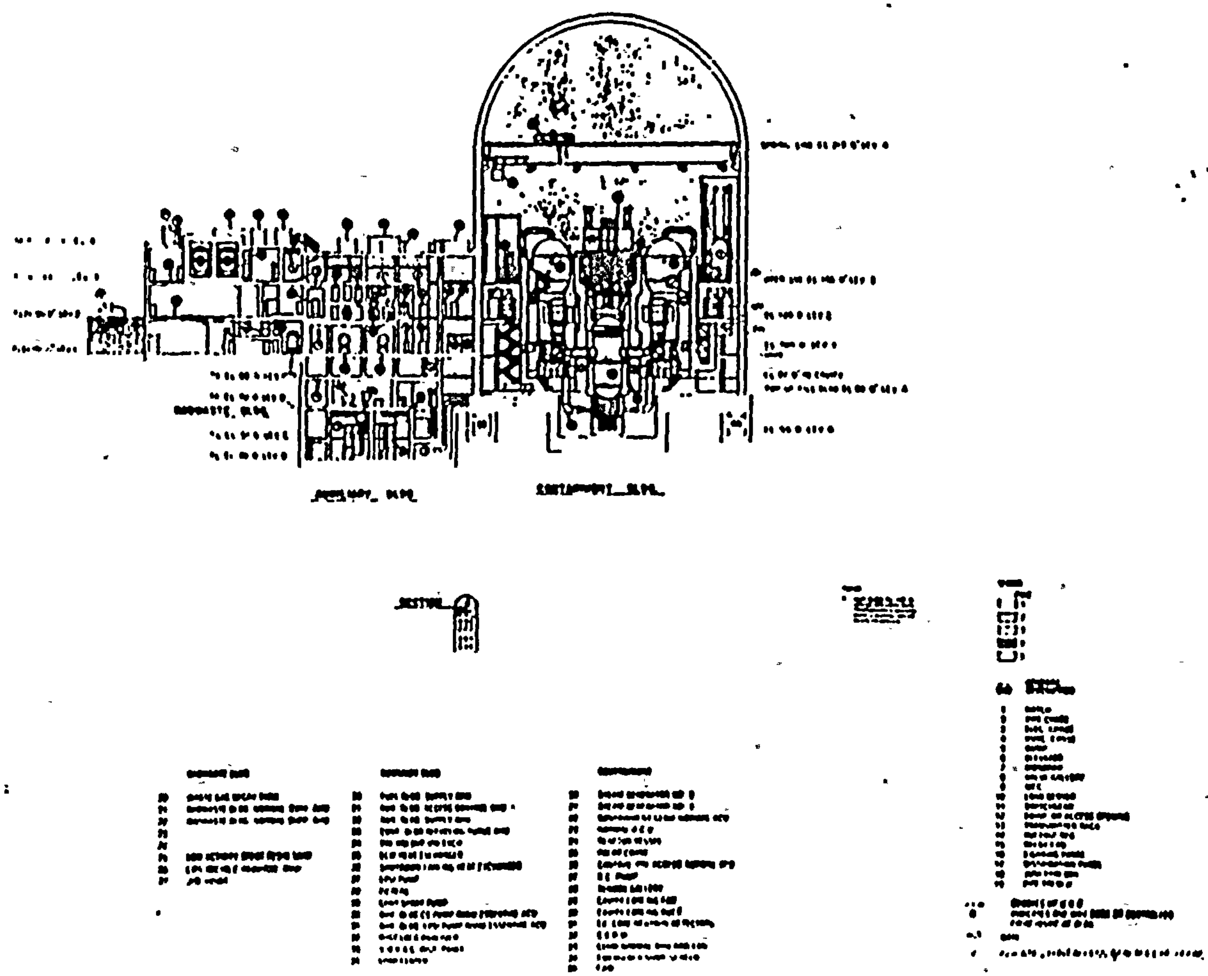
Figure 12.1-6



Palo Verde Nuclear Generating Station
FSAR

RADIATION ZONES (OPERATION)
SECTION J-J

Figure 12.3-7



PROPOSED

13-N-RAR-007, REV. 0

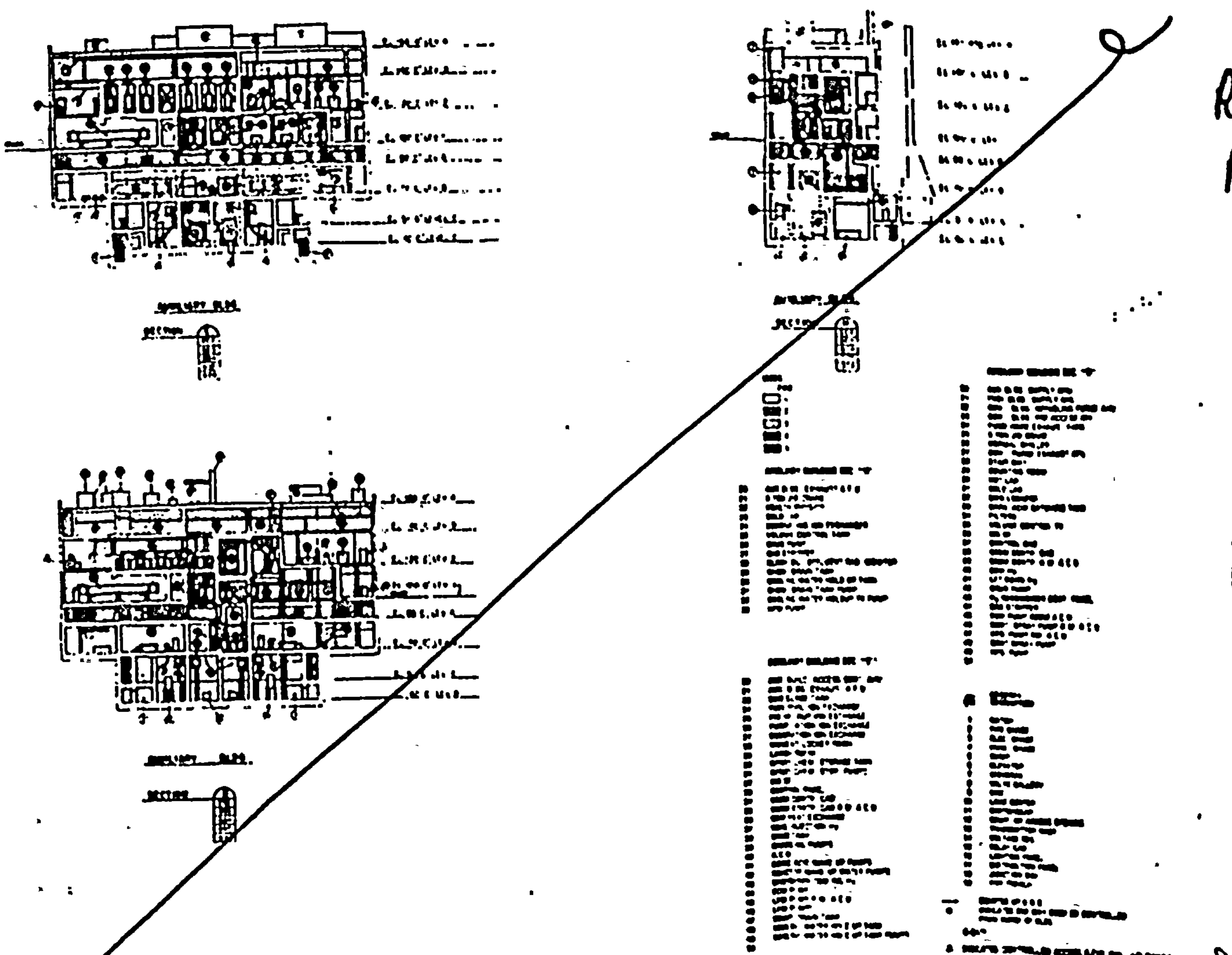
Palo Verde Nuclear Generating Station
FSAR

RADIATION ZONES (OPERATION)-
SECTION J-J

Figure 12.3-7

SARCN# 2088

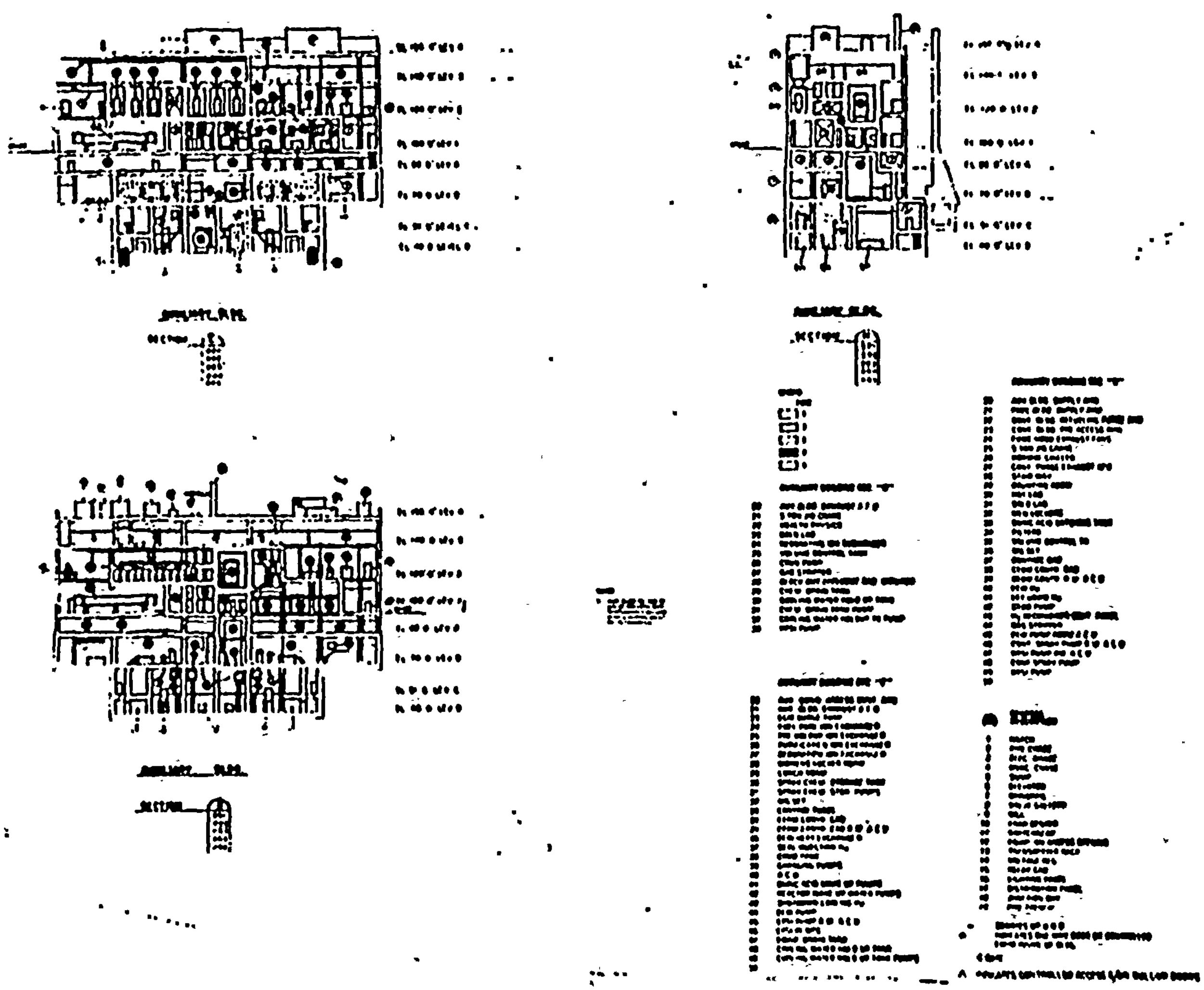
REPLACE WITH PROPOSED FIGURE 12.3-8



Palo Verde Nuclear Generating Stati[on]
FSAR

RADIATION ZONES (OPERATION)
SECTIONS D, E, N

Figure 12.3-8



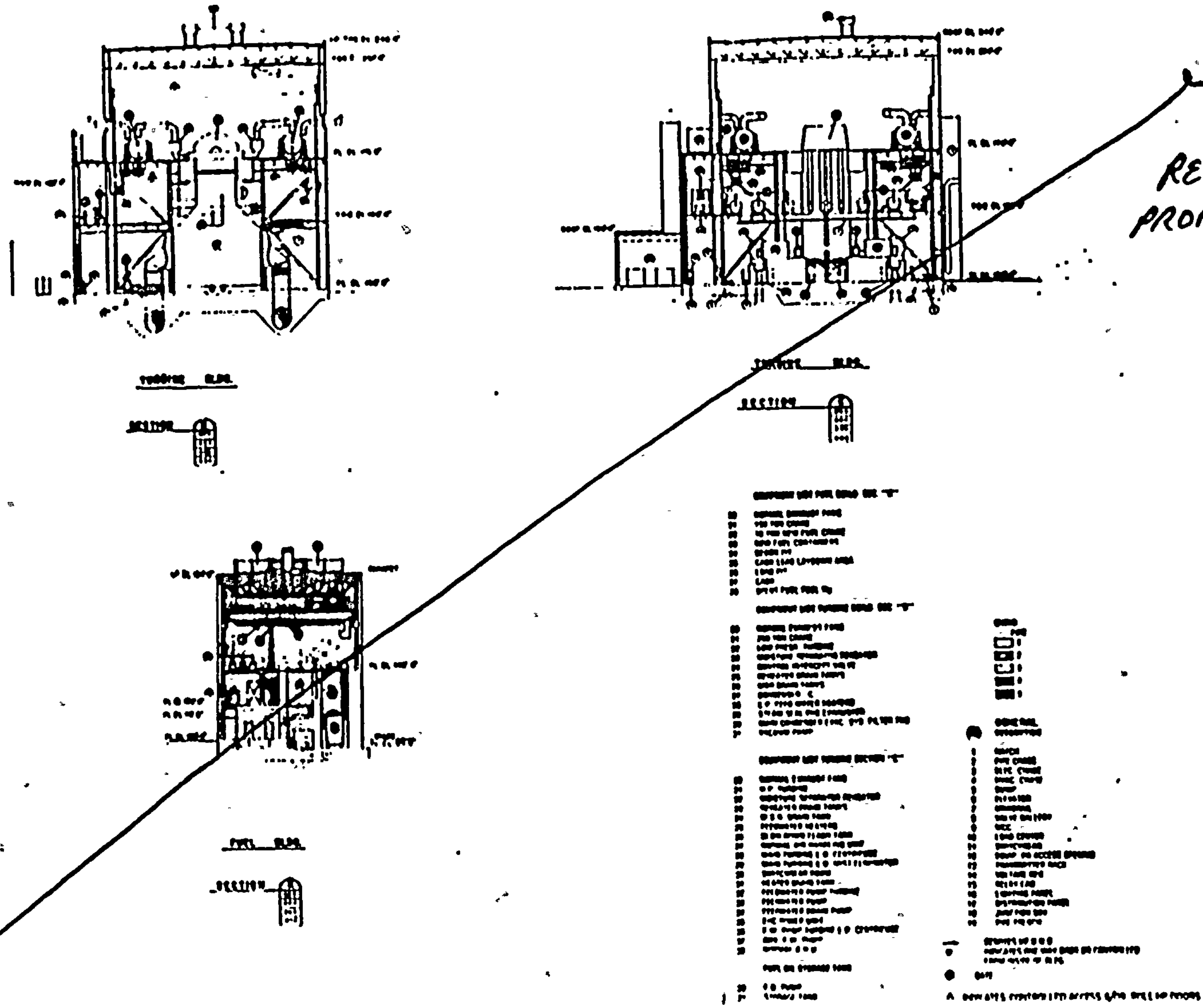
PROPOSED

13-N-RAR-008, REV.

Palo Verde Nuclear Generating Station
FSAR

RADIATION ZONES (OPERATION)
SECTIONS D, E, AND H

Figure 12.3-8

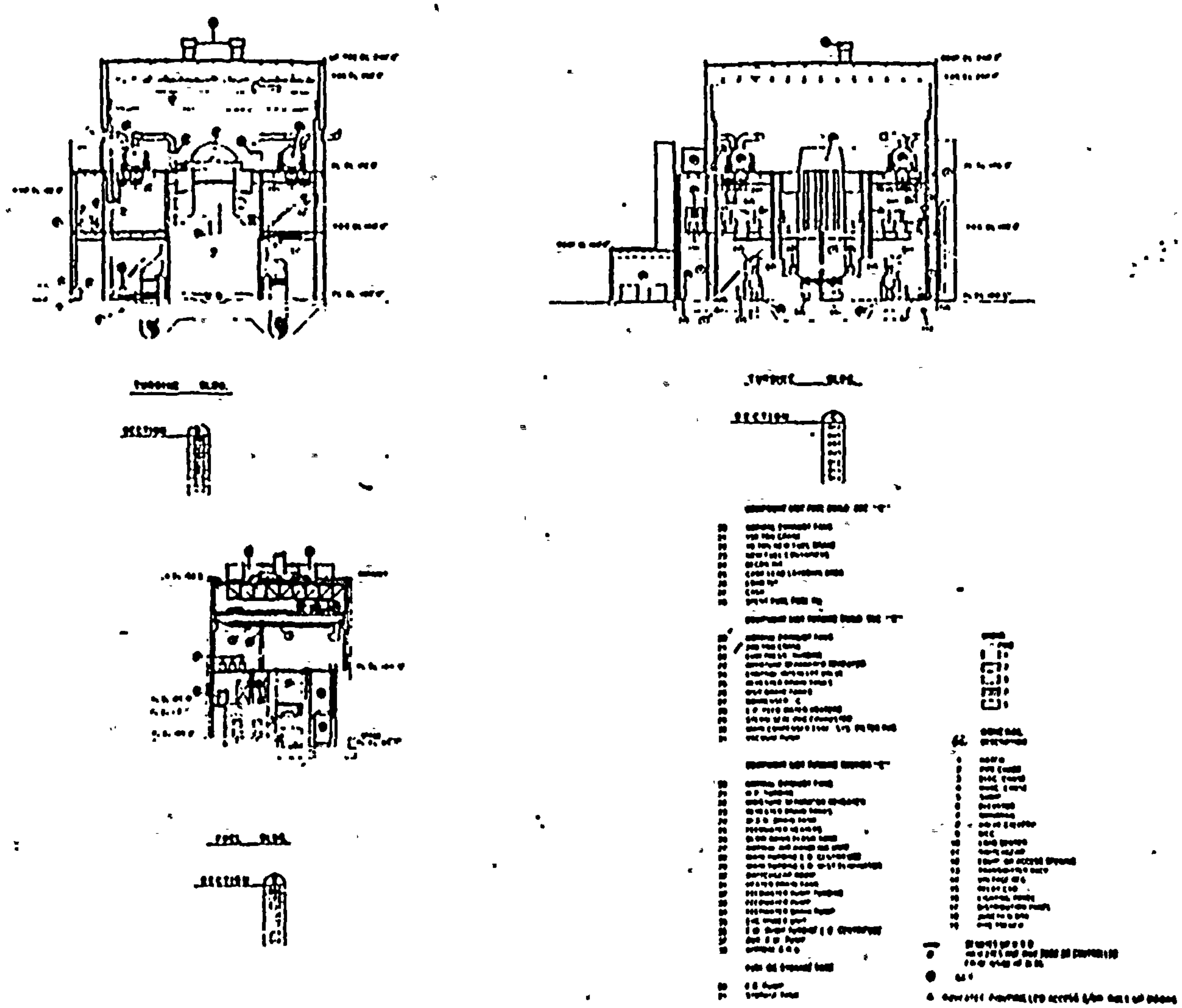


REPLACE WITH PROPOSED FIGURE 12-3-9

Palo Verde Nuclear Generating S
FSAR

RADIATION ZONES (OPERATION)
SECTIONS B, C, K

Figure 12.3-9

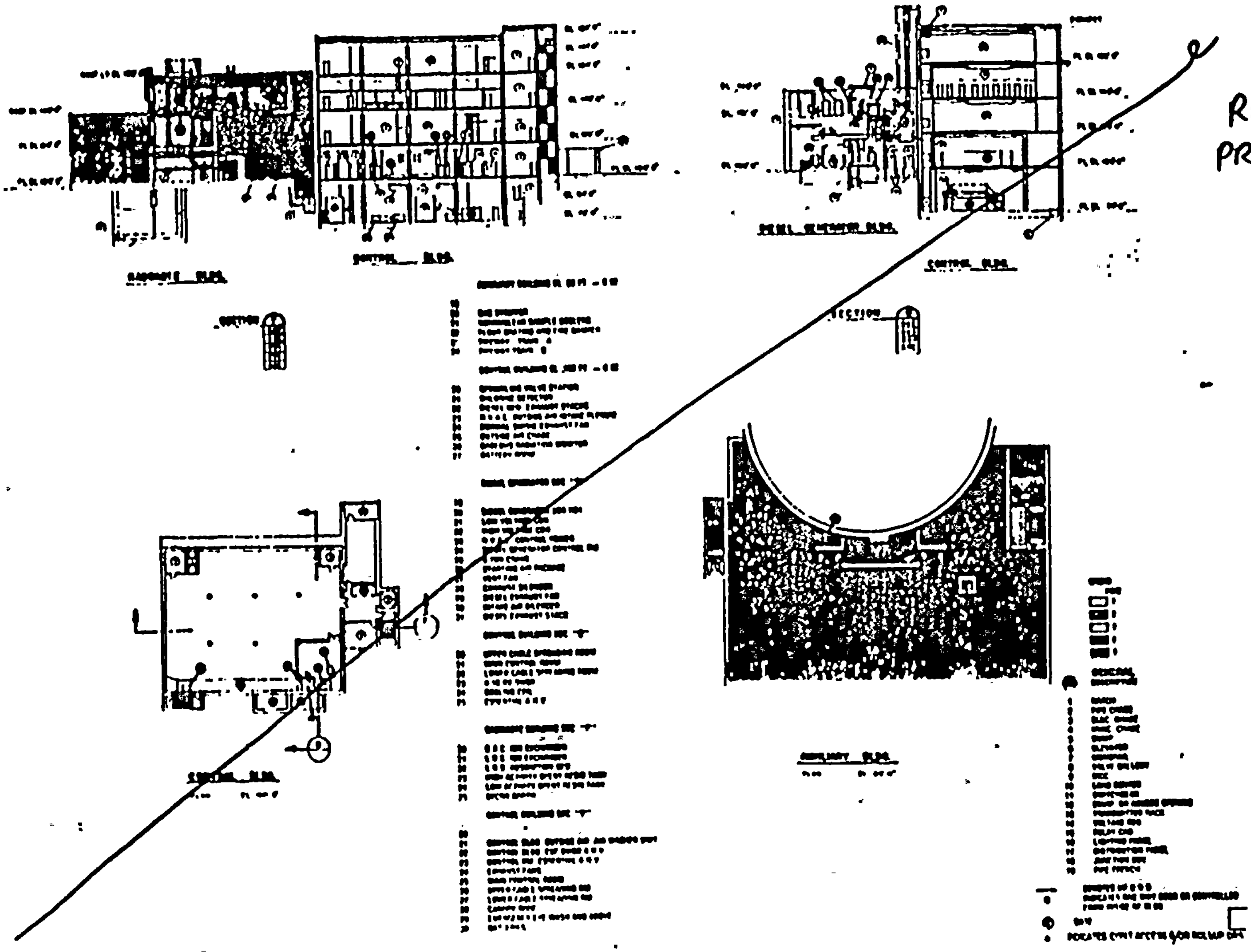


PROPOSED

Palo Verde Nuclear Generating Stat[ion]
FSAR

RADIATION ZONES (OPERATION)
SECTIONS B, C, AND K

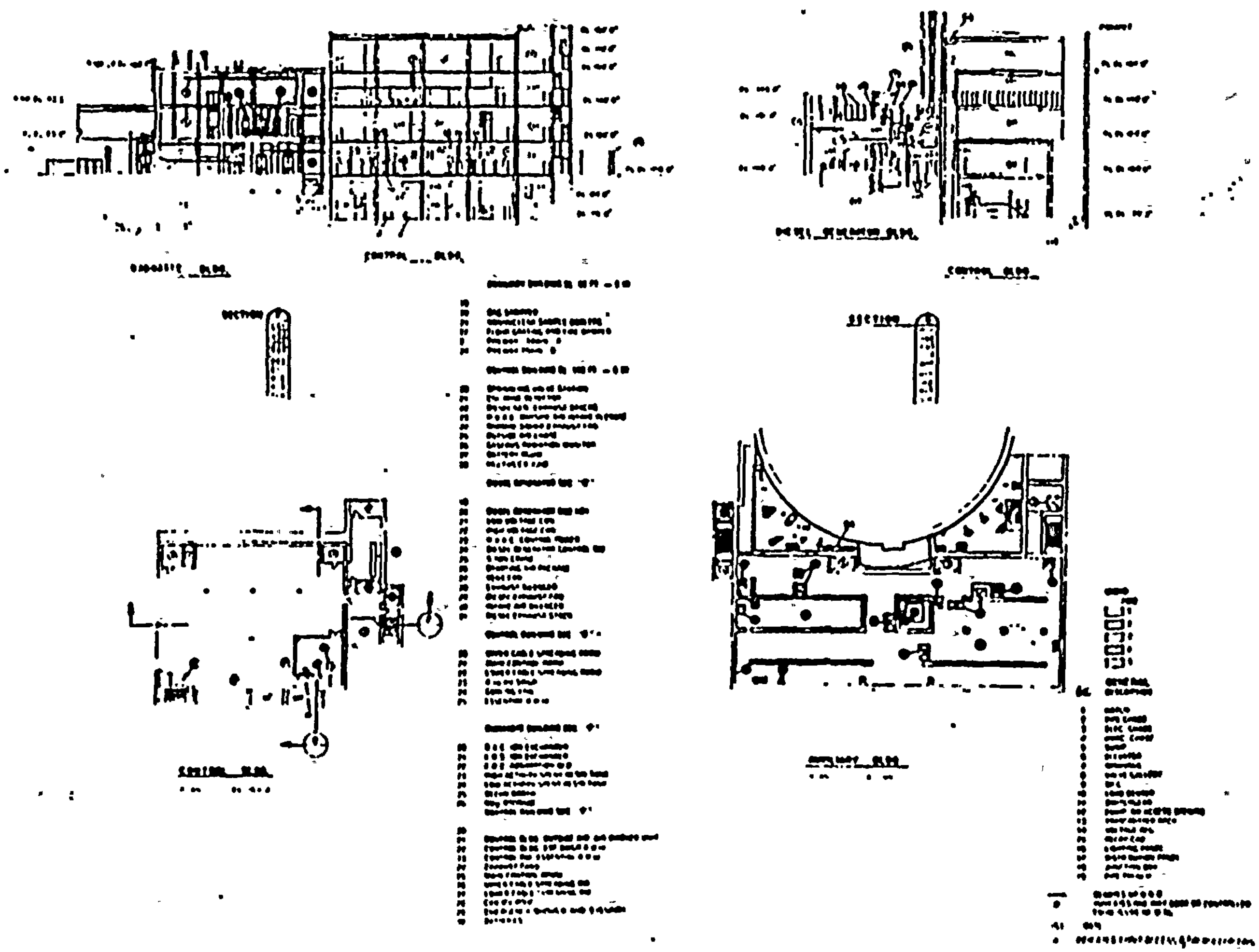
Figure 12.3-9



REPLACE WITH PROPOSED FIGURE 12.3-10

Palo Verde Nuclear Generating Stat
FSAR

RADIATION ZONES (OPERATION)
PLAN AT AUX BLDG EL 88'-0", &
PLAN AT CONTROL BLDG EL 160'-0",
& SECTIONS F AND G
Figure 12.3-10



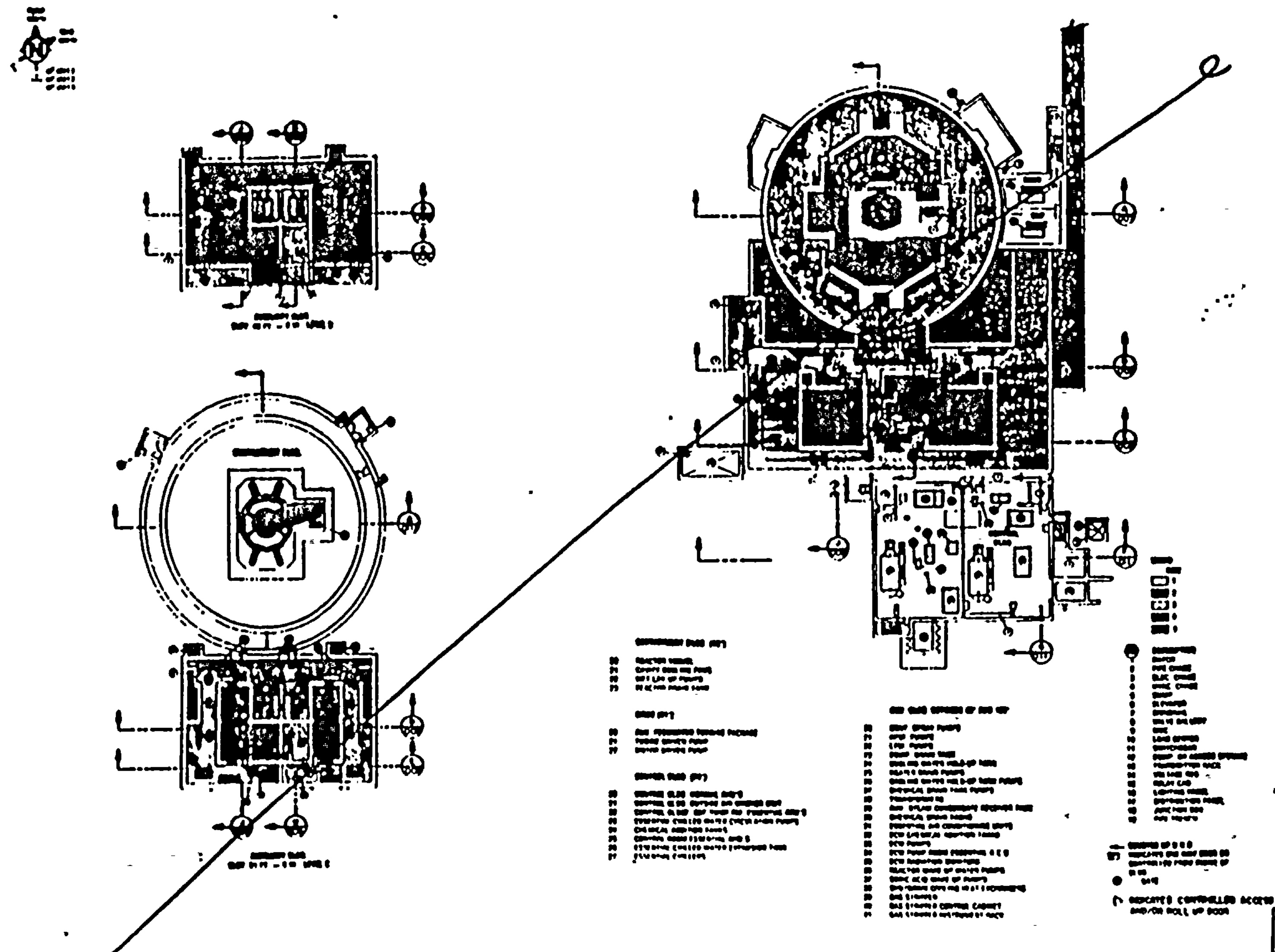
PROPOSED

13-N-RAR-010, REV.

Palo Verde Nuclear Generating Station
FSAR

RADIATION ZONES (OPERATION)
AT AUXILIARY BLDG.
EL. 88'-0", CONTROL BLDG. EL. 100'-0",
AND SECTIONS F AND G
Figure 12.3-10

SARCN# 2088

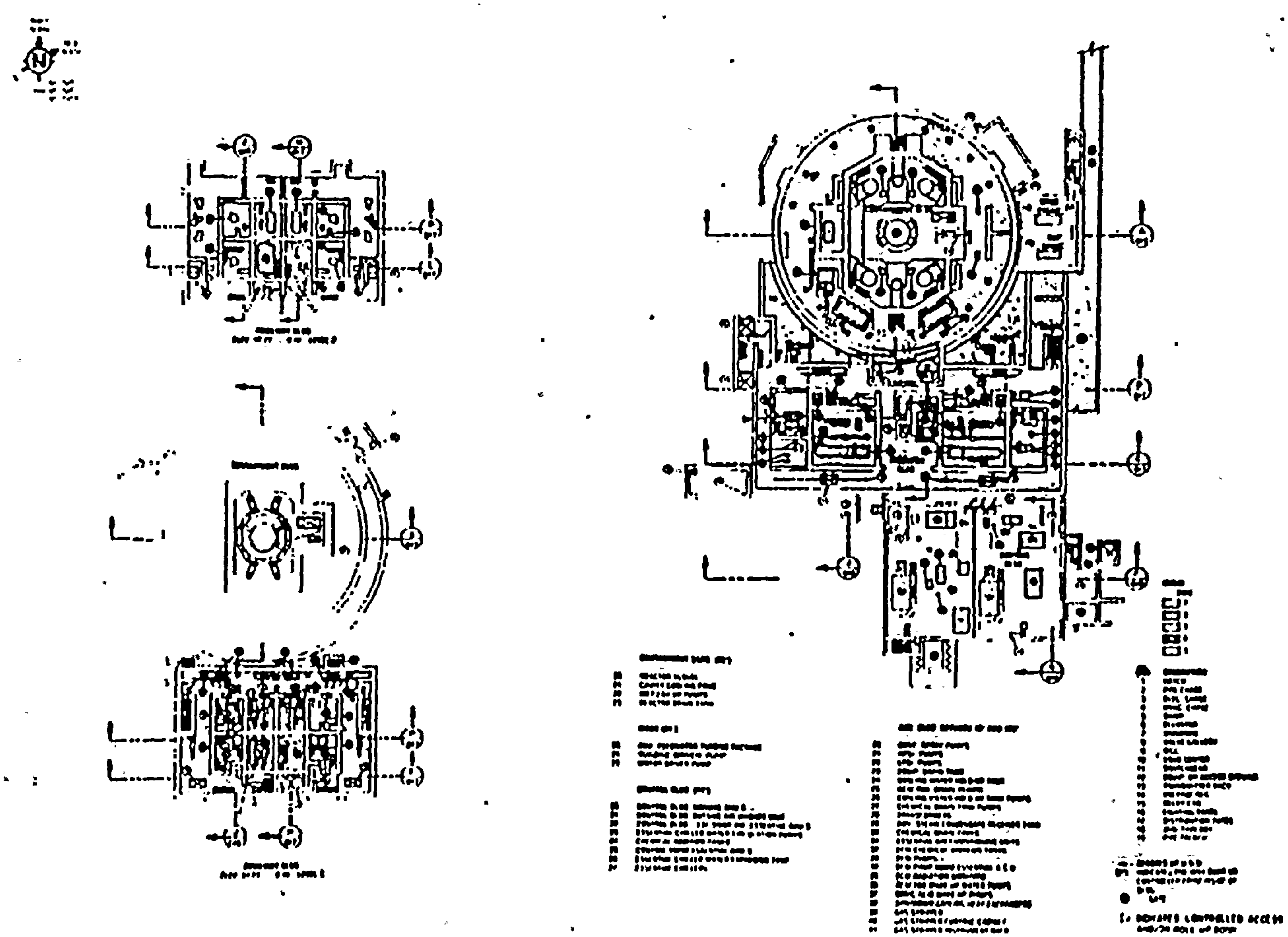


REPLACE WITH PROPOSED FIGURE 12.3-11

Palo Verde Nuclear Generating Station
FSAR

RADIATION ZONES (REFUELING)
BETWEEN EL 40'-0" & 100'-0"

Figure 12.3-11

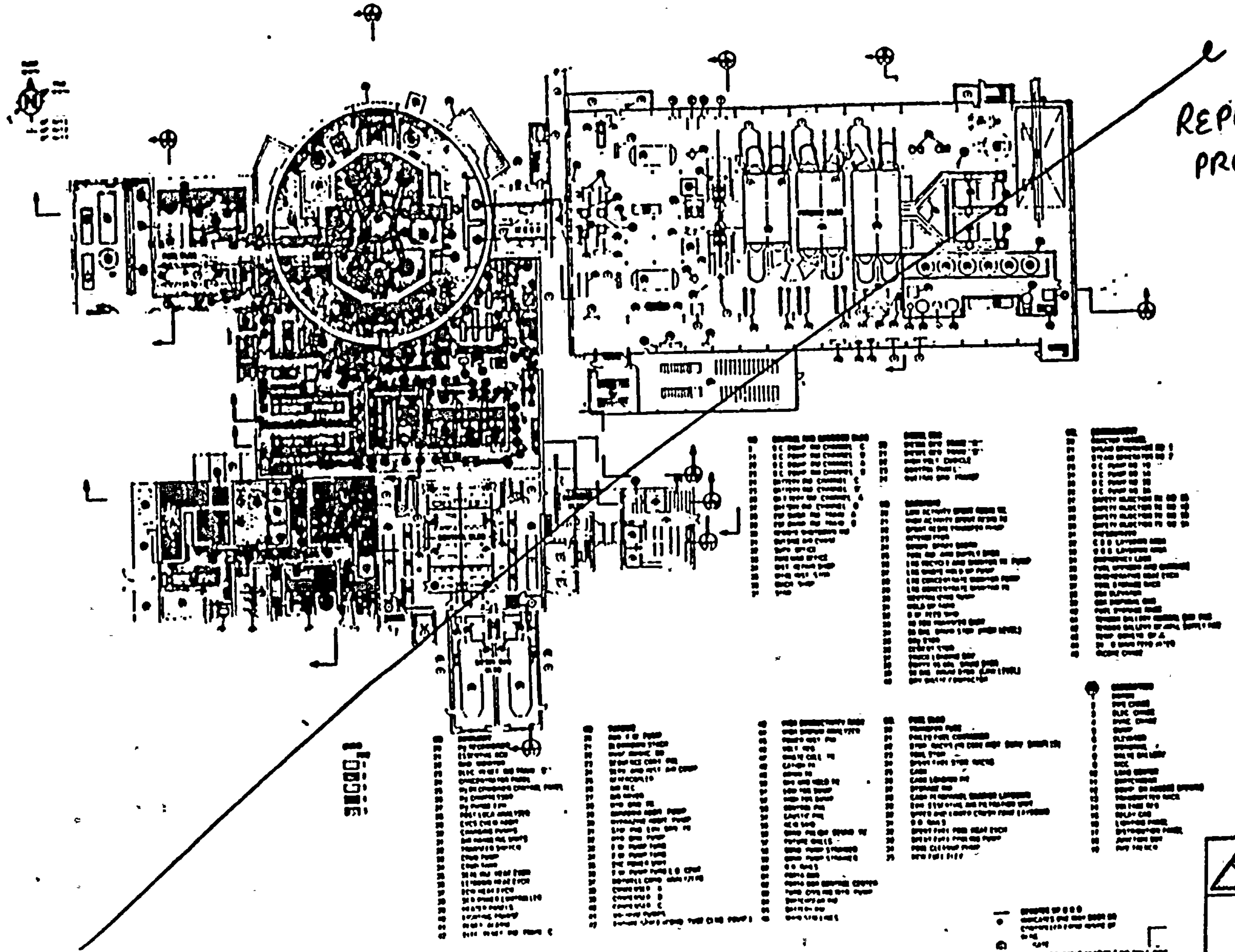


PROPOSED

Palo Verde Nuclear Gen
FSAR

RADIATION ZONES (RE
BETWEEN EL. 40'-0" AND

Figure 12.3-1



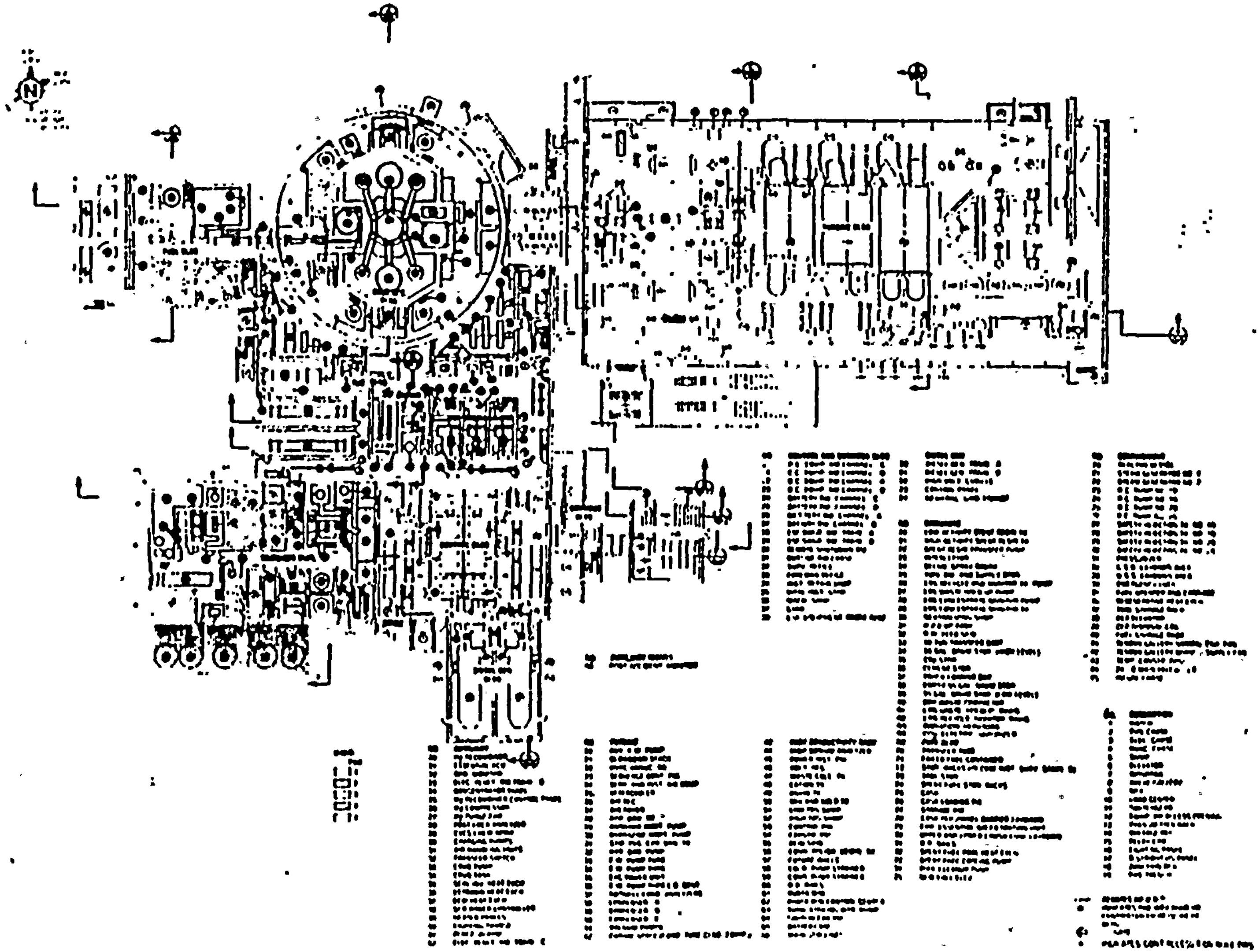
REPLACE WITH PROPOSED FIGURE 12.3-12

SAKCN # 2088

Palo Verde Nuclear Generating Station
FSAR

RADIATION ZONES (REFUELING)
AT EL 100'-0"

Figure 12.3-12



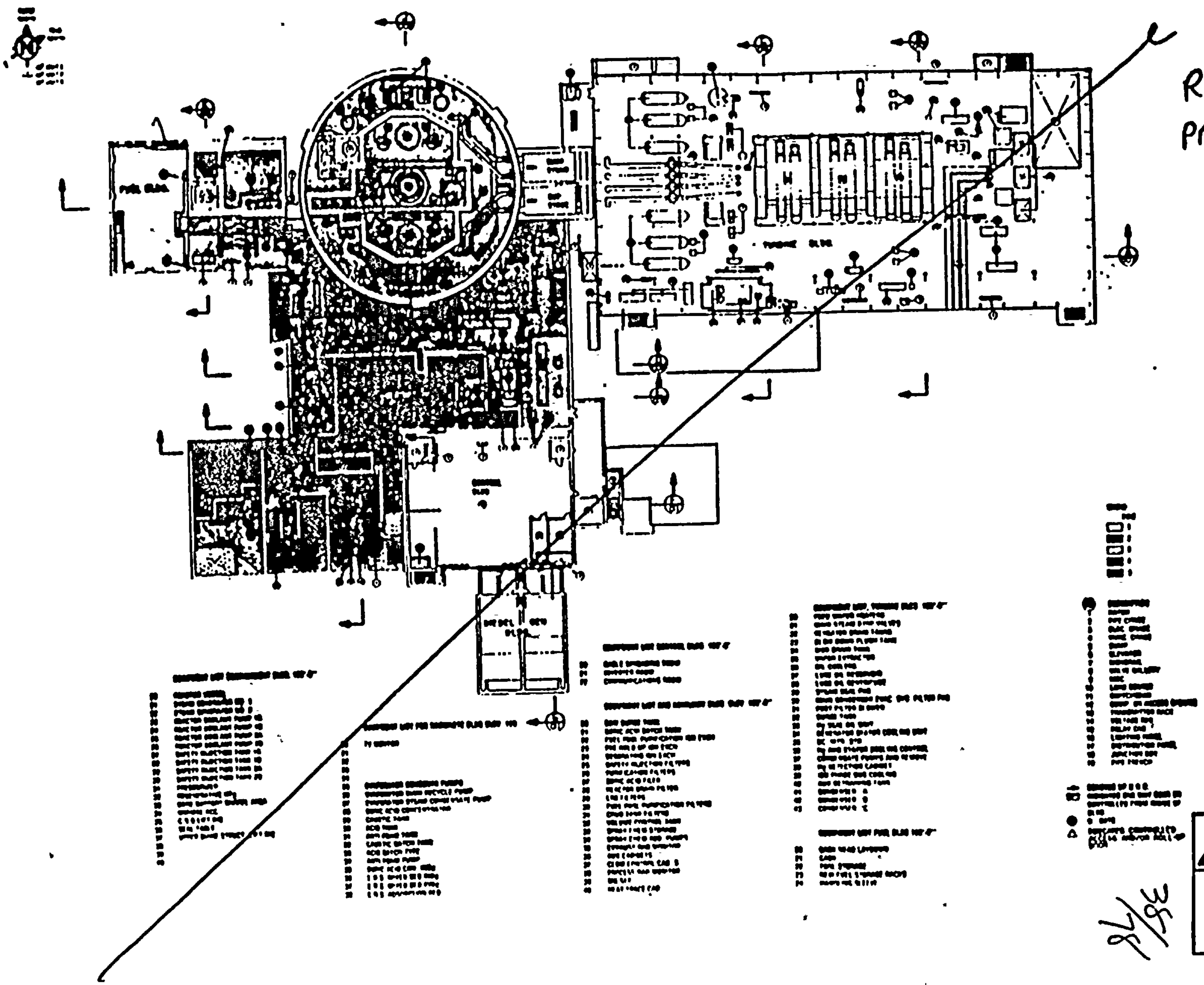
PROPOSED

13-N-RAR-012, REV. [illegible]

Palo Verde Nuclear Generating Station
FSAR

RADIATION ZONES (REFUELING)
AT EL. 100'-0"

Figure 12.3-17

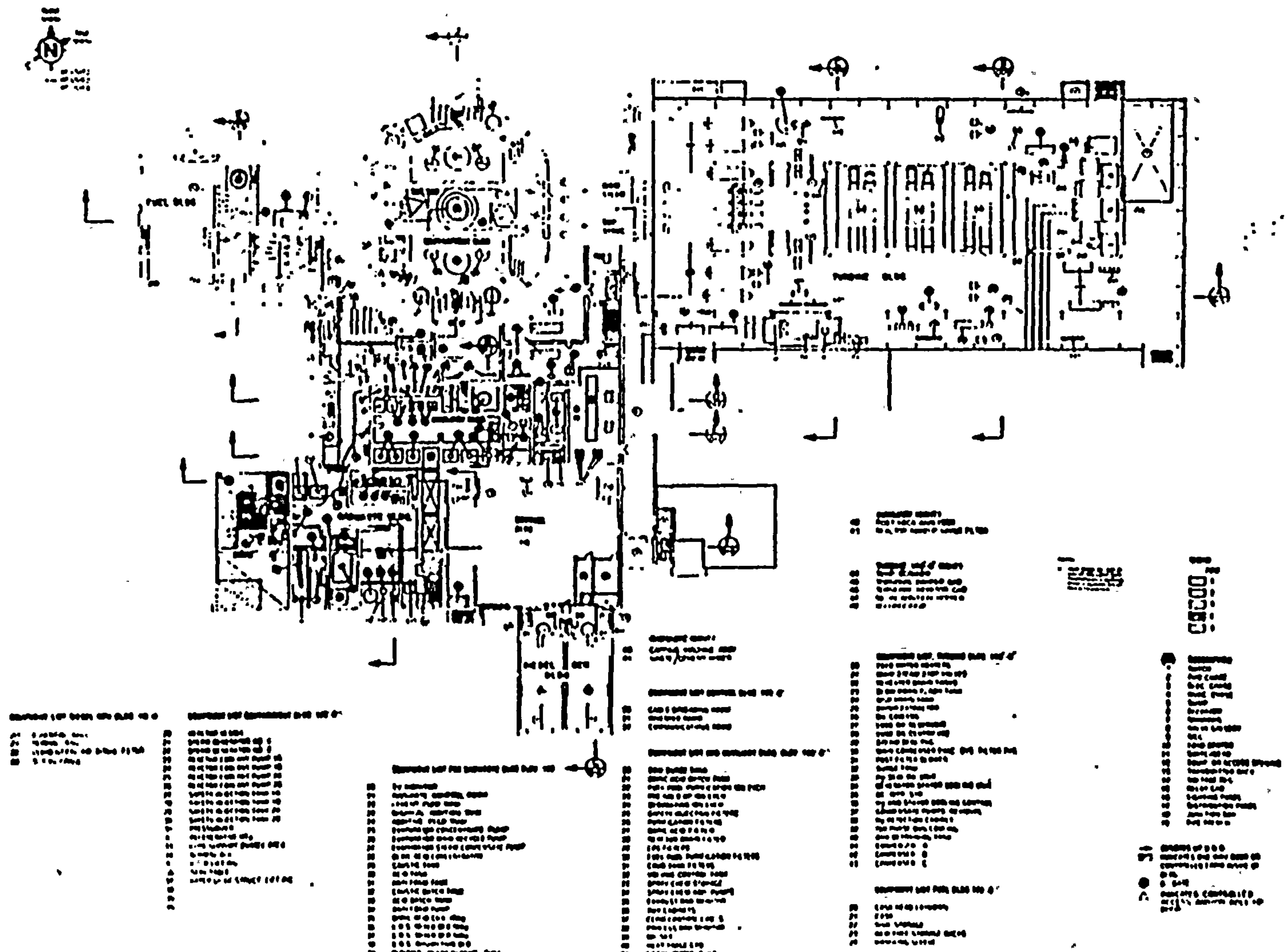


REPLACE WITH PROPOSED FIGURE 12.3-13

Palo Verde Nuclear Generating Station
FSAR

RADIATION ZONES (REFUELING)
BETWEEN EL 120'-0" & 140'-0"

Figure 12.3-13



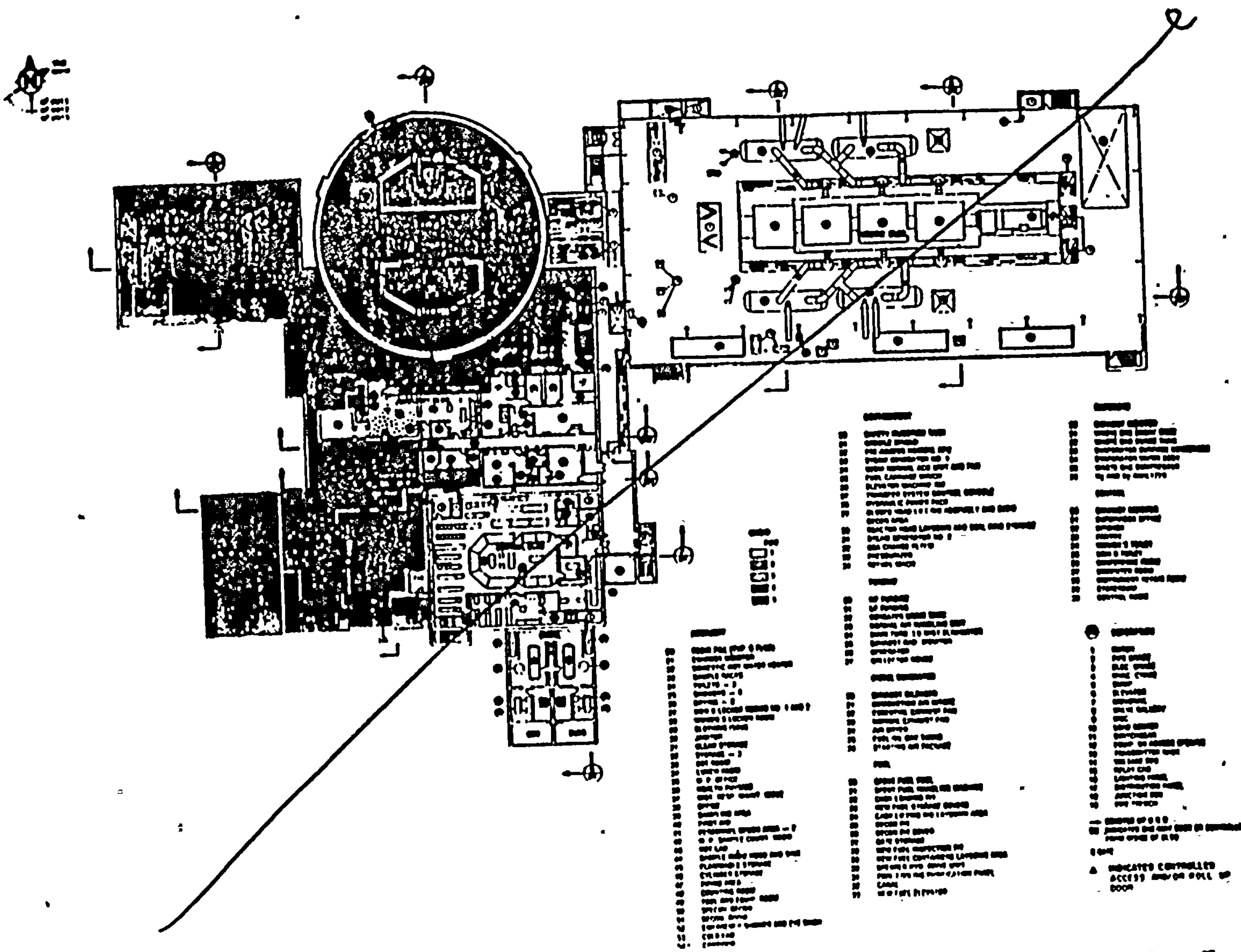
PROPOSED

13-N-RAR-012, REV. 0

Palo Verde Nuclear Generating Station
FSAR

RADIATION ZONES (REFUELING)
BETWEEN EL. 120'-0" AND EL. 140'-0"

Figure 12.3-11

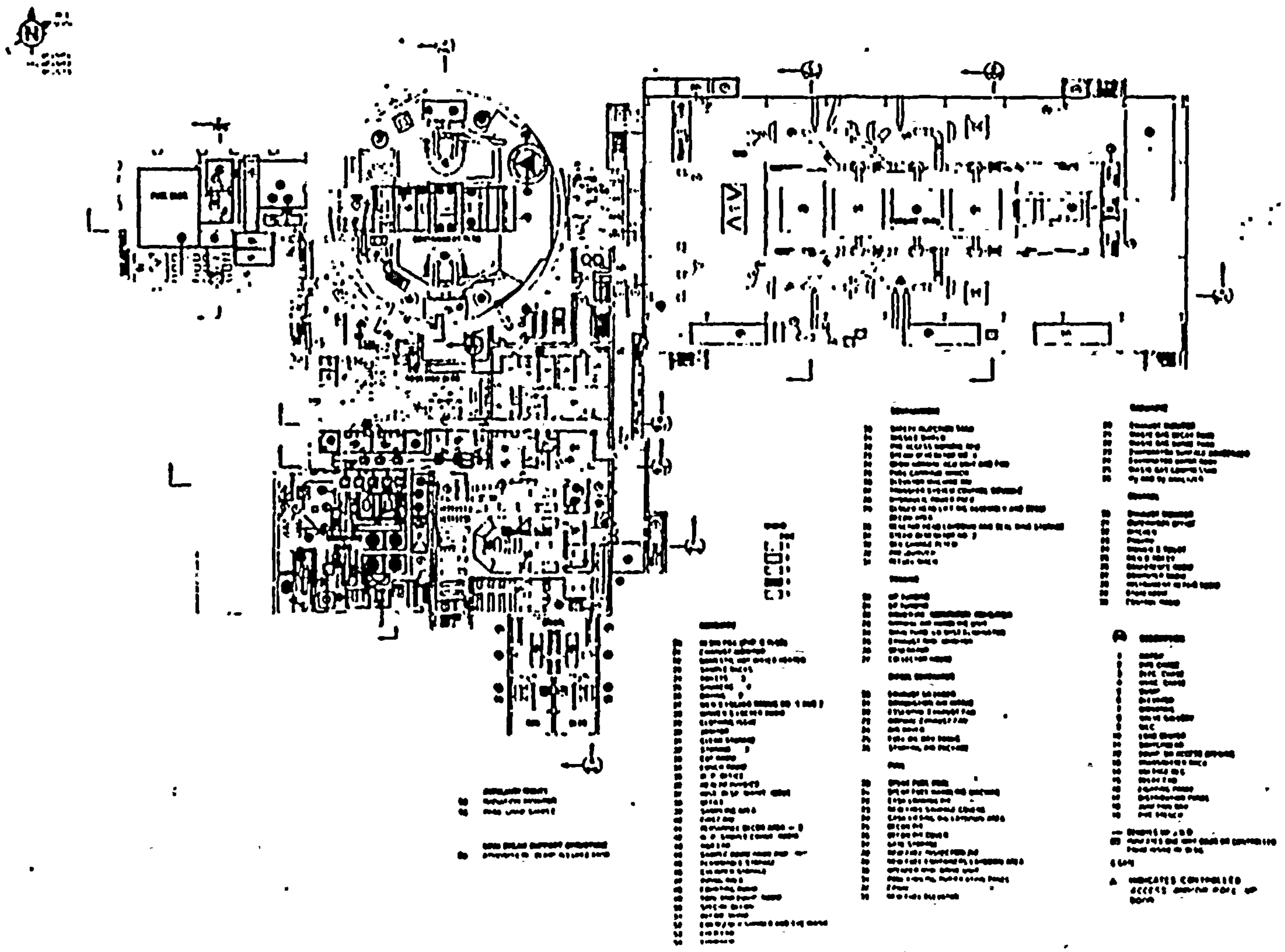


REPLACE WITH PROPOSED FIGURE 12.3-14

Palo Verde Nuclear Generating Station FSAR

RADIATION ZONES (REFUELING) BETWEEN EL 140'-0" & 200'-0"

Figure 12.3-14



PROPOSED

13-N-RAR-016, REV. 0

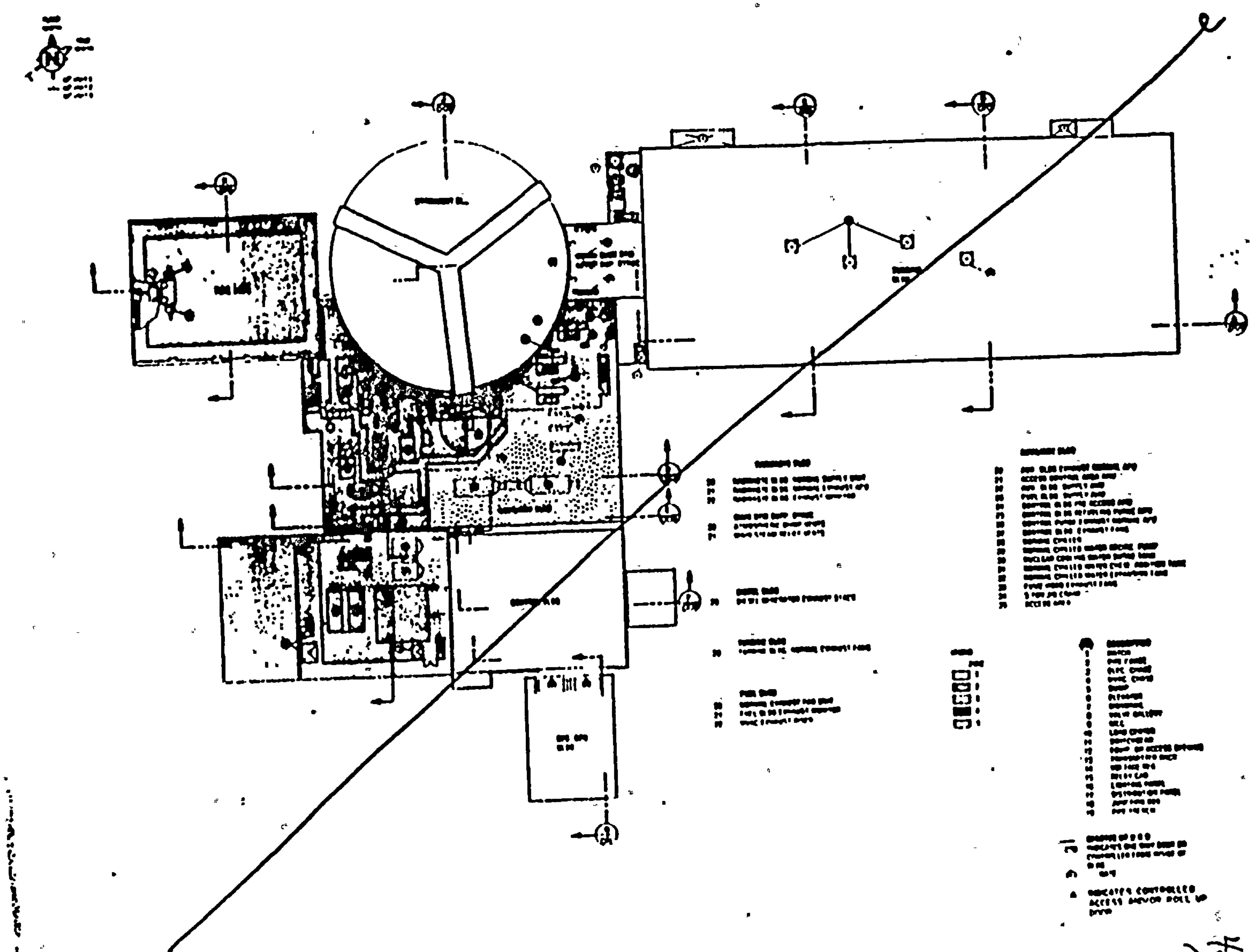
Palo Verde Nuclear Generating Station
FSAR

RADIATION ZONES (REFUELING)
BETWEEN EL. 140'-0" AND EL. 200'-0"

Figure 12.3-14

SARCN #2088

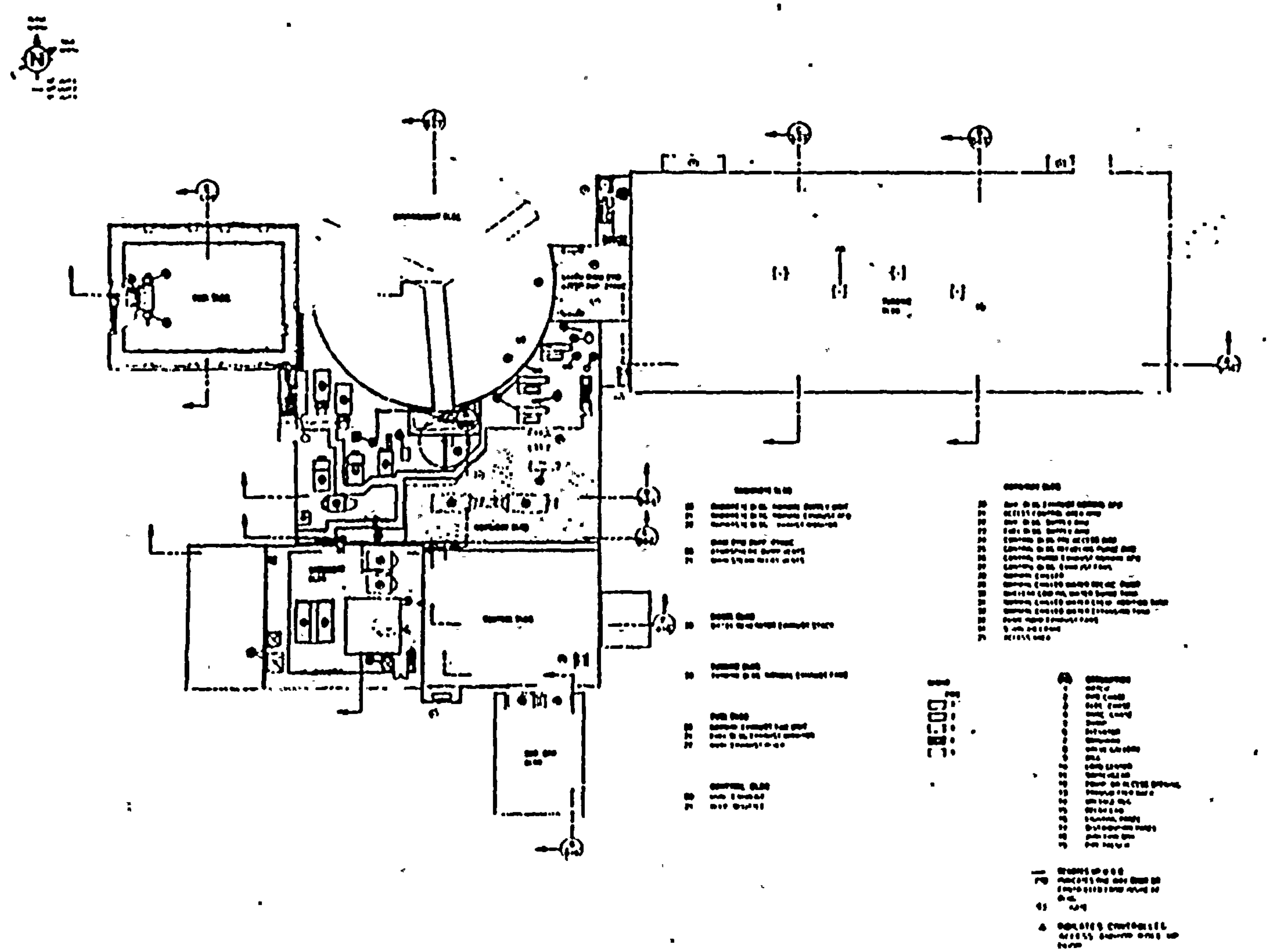
REPLACE WITH PROPOSED FIGURE 12.3-15



Palo Verde Nuclear Generating Station FSAR

RADIATION ZONES (REFUELING) AT ROOF ELEVATION

Figure 12.3-15



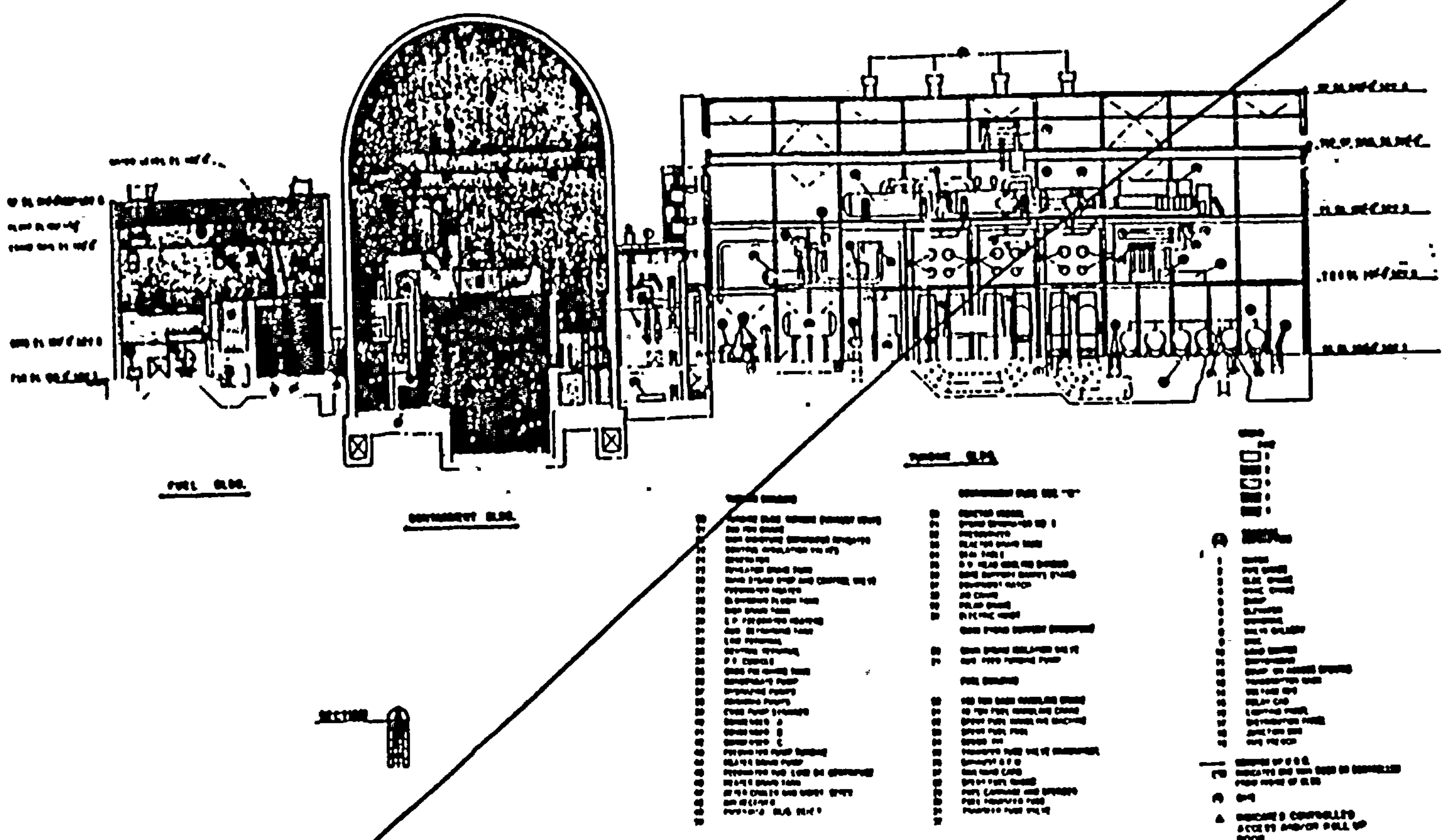
SARCN #2088

PROPOSED

Palo Verde Nuclear Generating Station
FSAR

RADIATION ZONES (REFUELING)
AT ROOF ELEVATION

Figure [illegible]



REPLACE WITH PROPOSED FIGURE 12.3-16

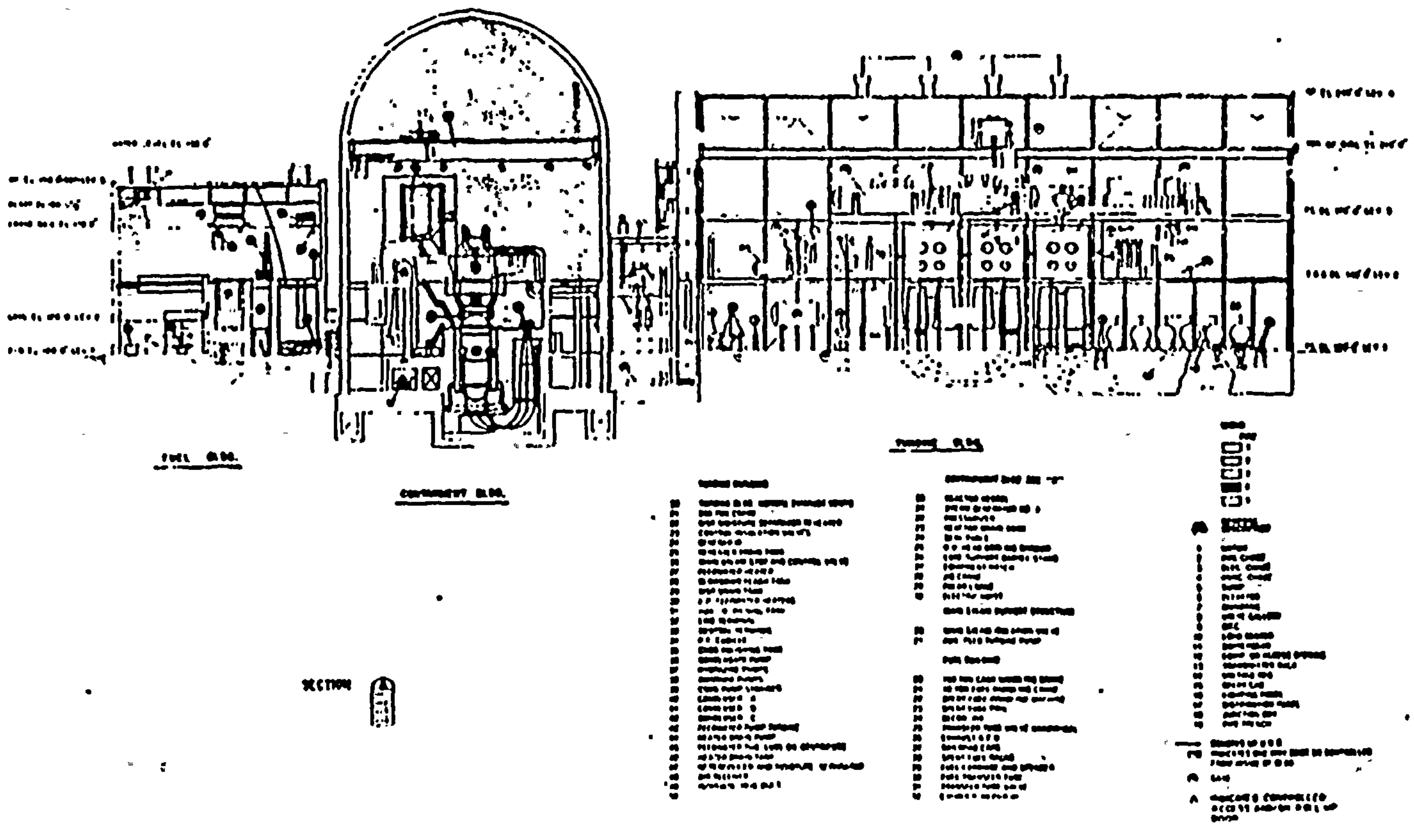
SARCN # 2088

Palo Verde Nuclear Generating Station
FSAR

RADIATION ZONES (REFUELING)
SECTION A

Figure 12.3-16.

18 Jul 78



PROPOSED

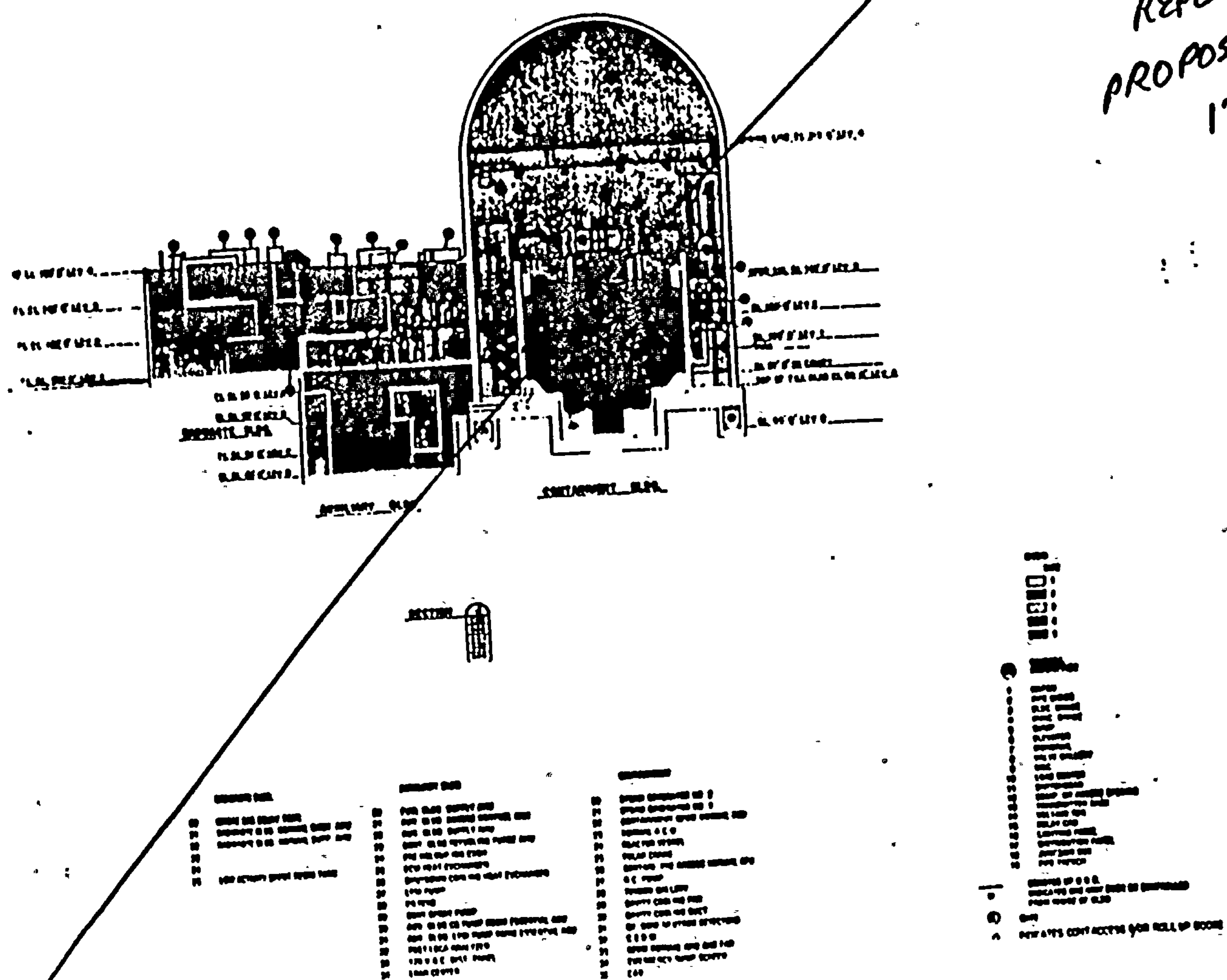
13-N-RAR-019, REV. [illegible]

Palo Verde Nuclear Generating Station
FSAR

RADIATION ZONES (REFUELING) -
SECTION A-A

Figure 12.3-16

REPLACE WITH PROPOSED FIGURE 12.3-17

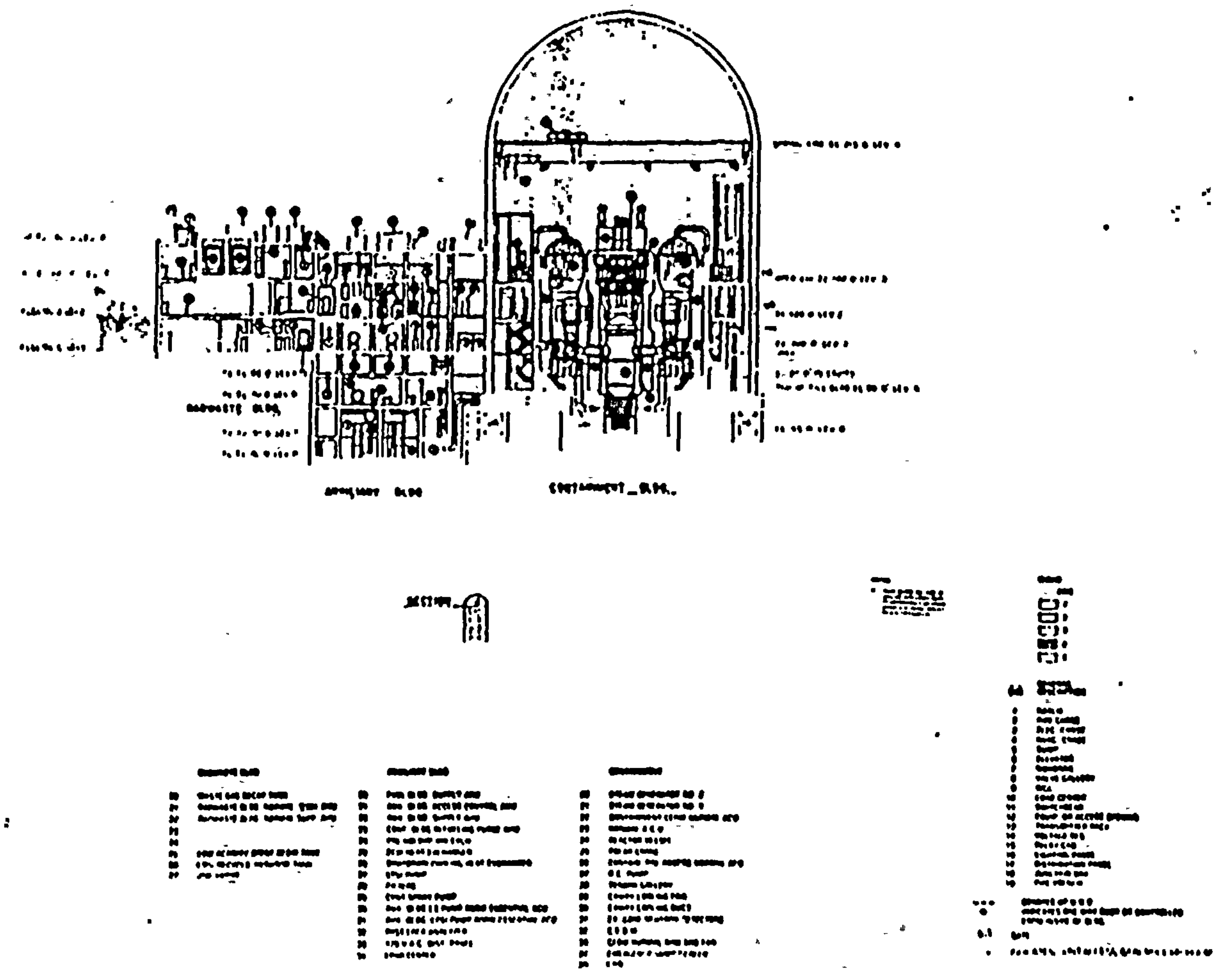


SARCN #2088

Palo Verde Nuclear Generating Station
FSAR

RADIATION ZONES (REFUELING)
SECTION J

Figure 12.3-17



PROPOSED

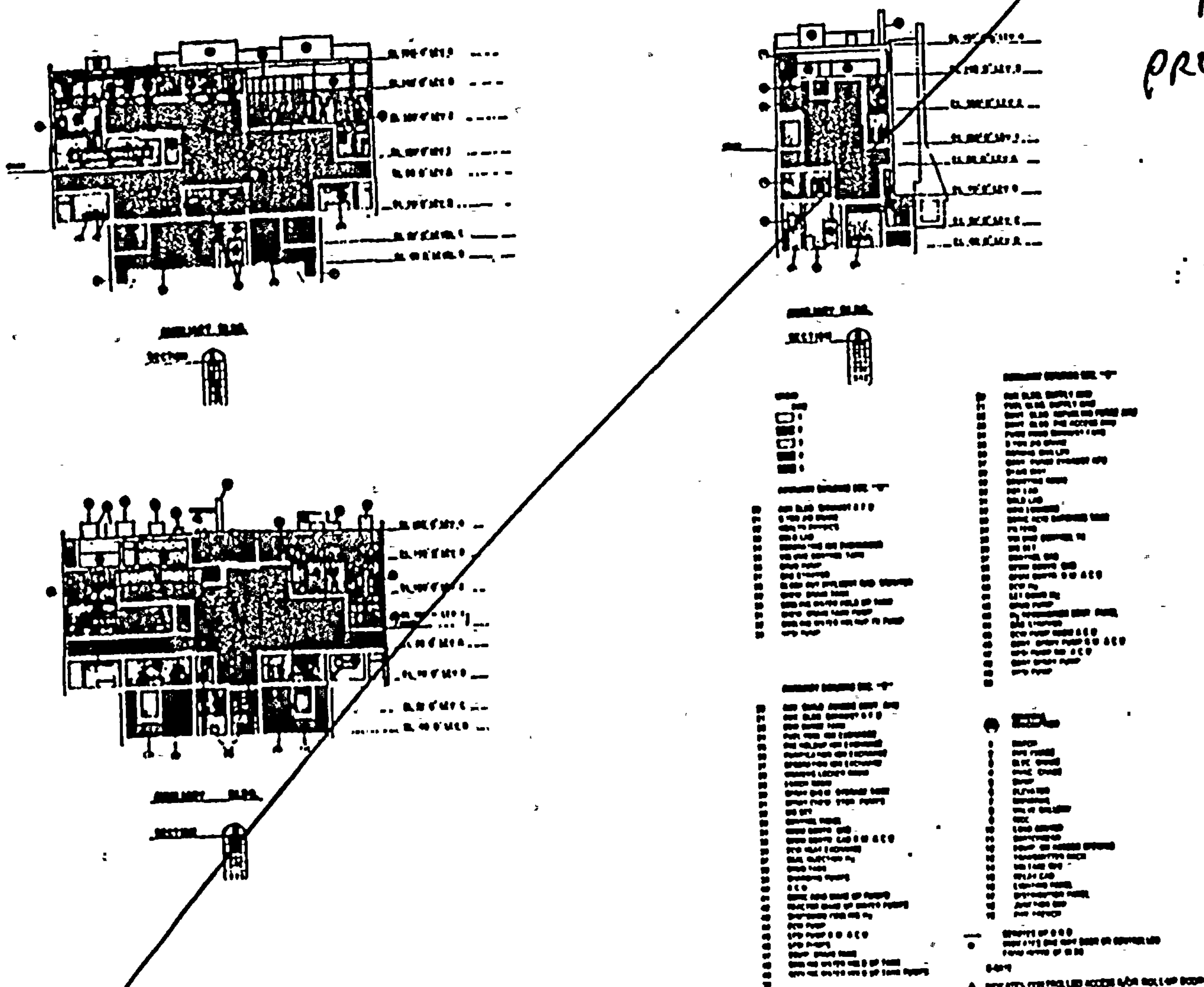
13-N-RAR-016, REV. 0

Palo Verde Nuclear Generating Station
FSAR

RADIATION ZONES (REFUELING)
SECTION 3-3

Figure 12.3-17

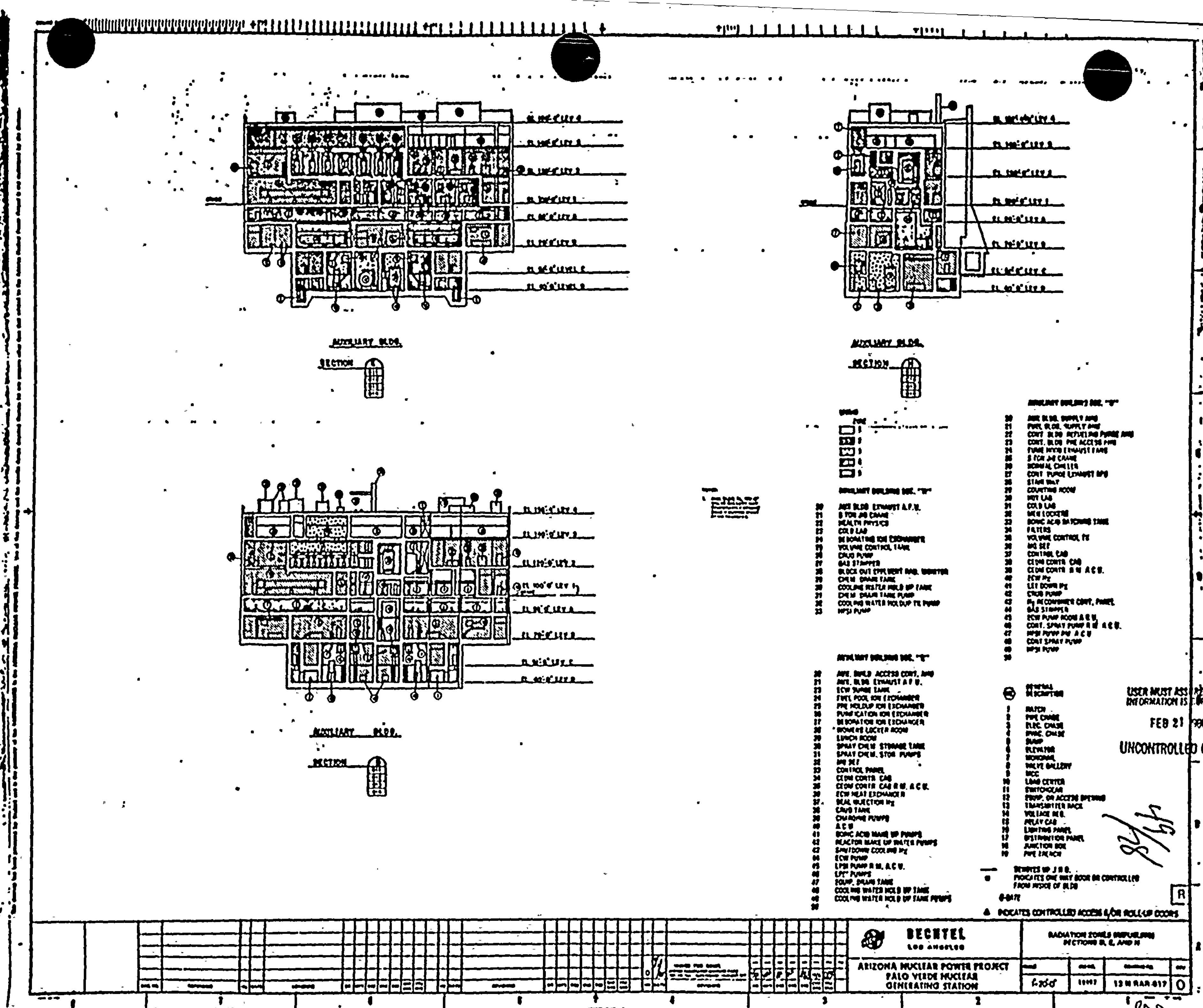
REPLACE WITH PROPOSED FIGURE 12.3-18



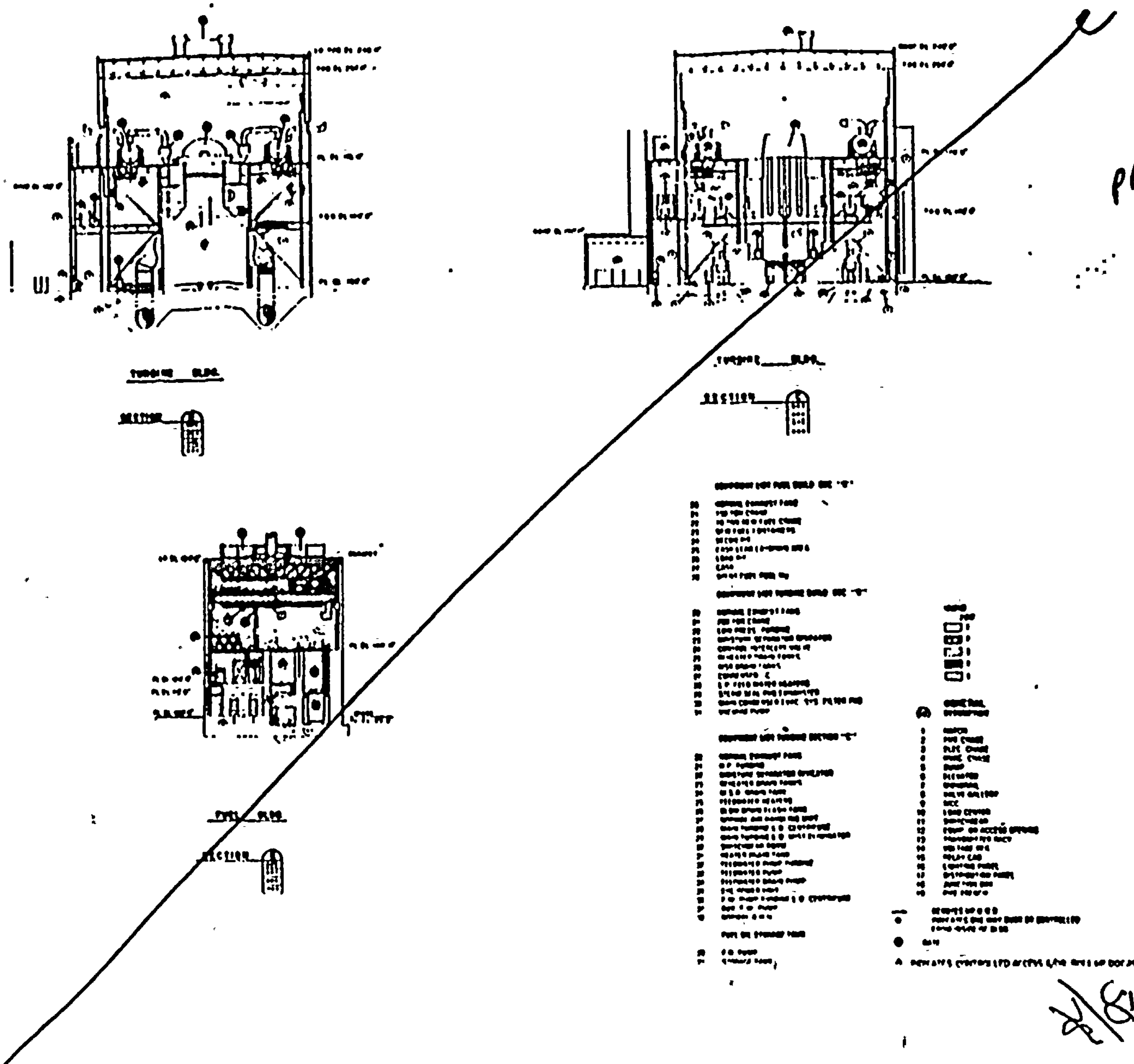
Palo Verde Nuclear Generating Station
FSAR

RADIATION ZONES (REFUELING)
SECTIONS D, E, M

Figure 12.3-18



FSAR Figure 12.3-18

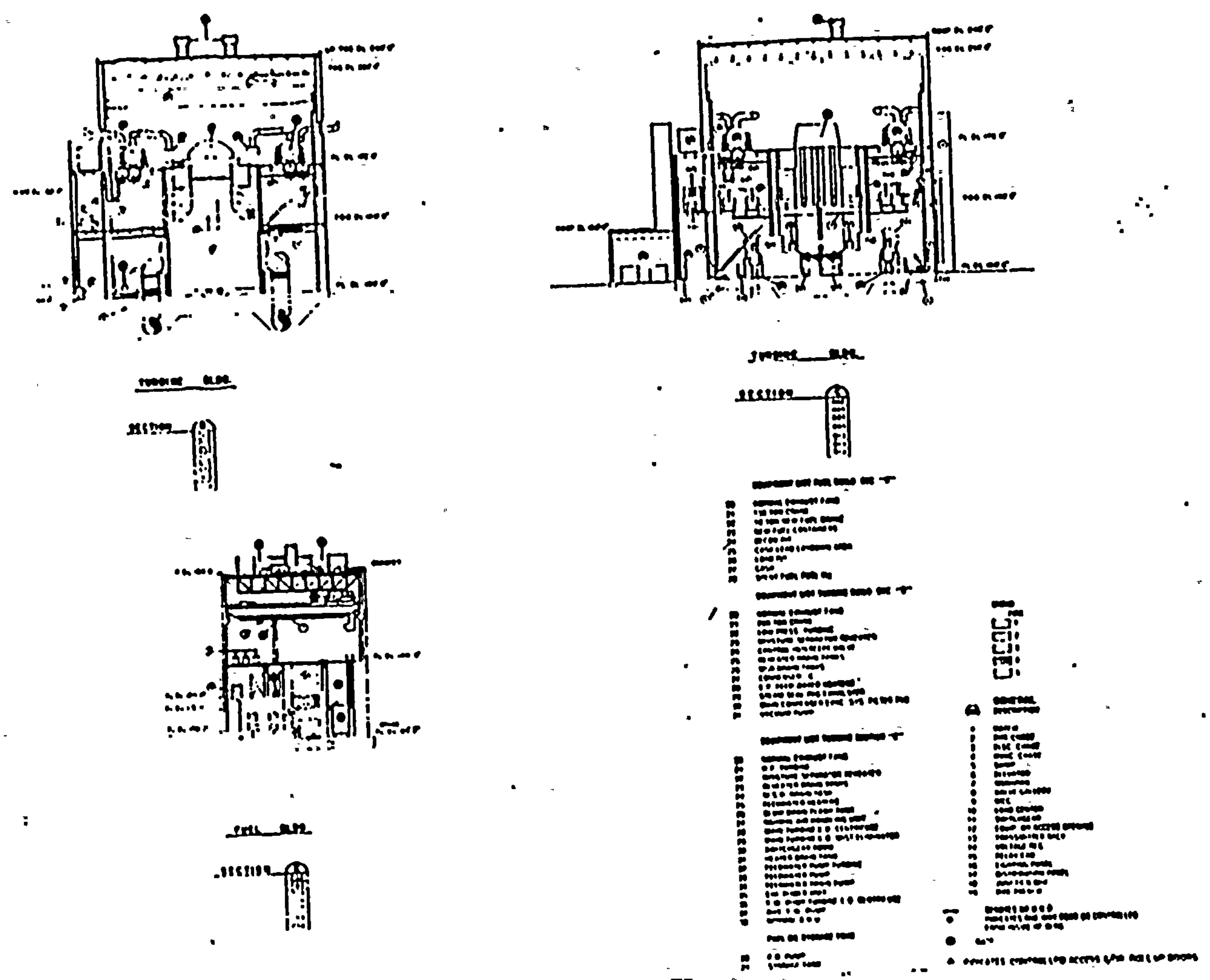


REPLACE WITH PROPOSED FIGURE 12.3-19

Palo Verde Nuclear Generating Station
FSAR

RADIATION ZONES (REFUELING)
SECTIONS B, C, K

Figure 12.3-19



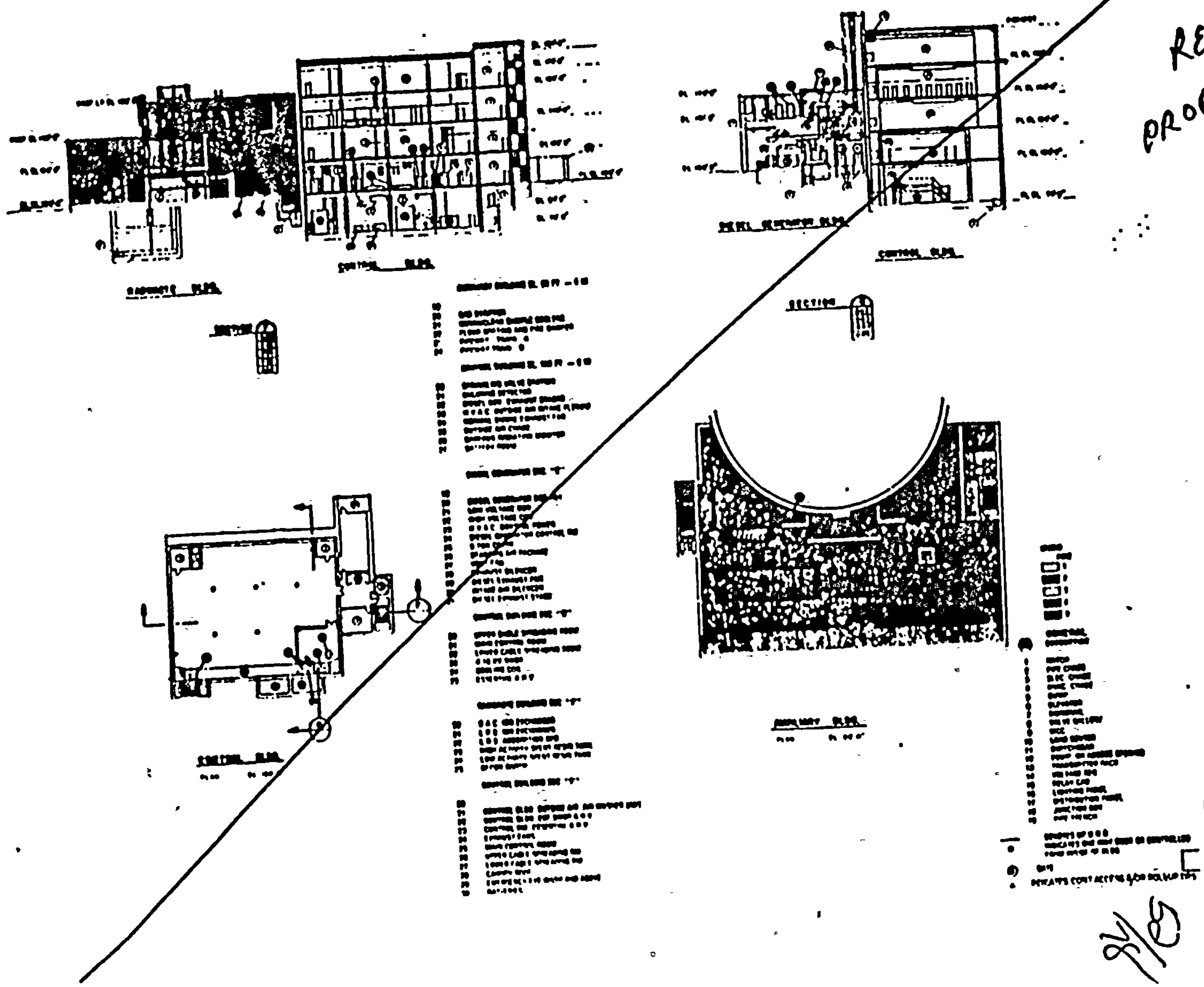
PROPOSED

Palo Verde Nuclear Generating Station
FSAR

RADIATION ZONES (REFUELING)
SECTIONS B, C, AND K

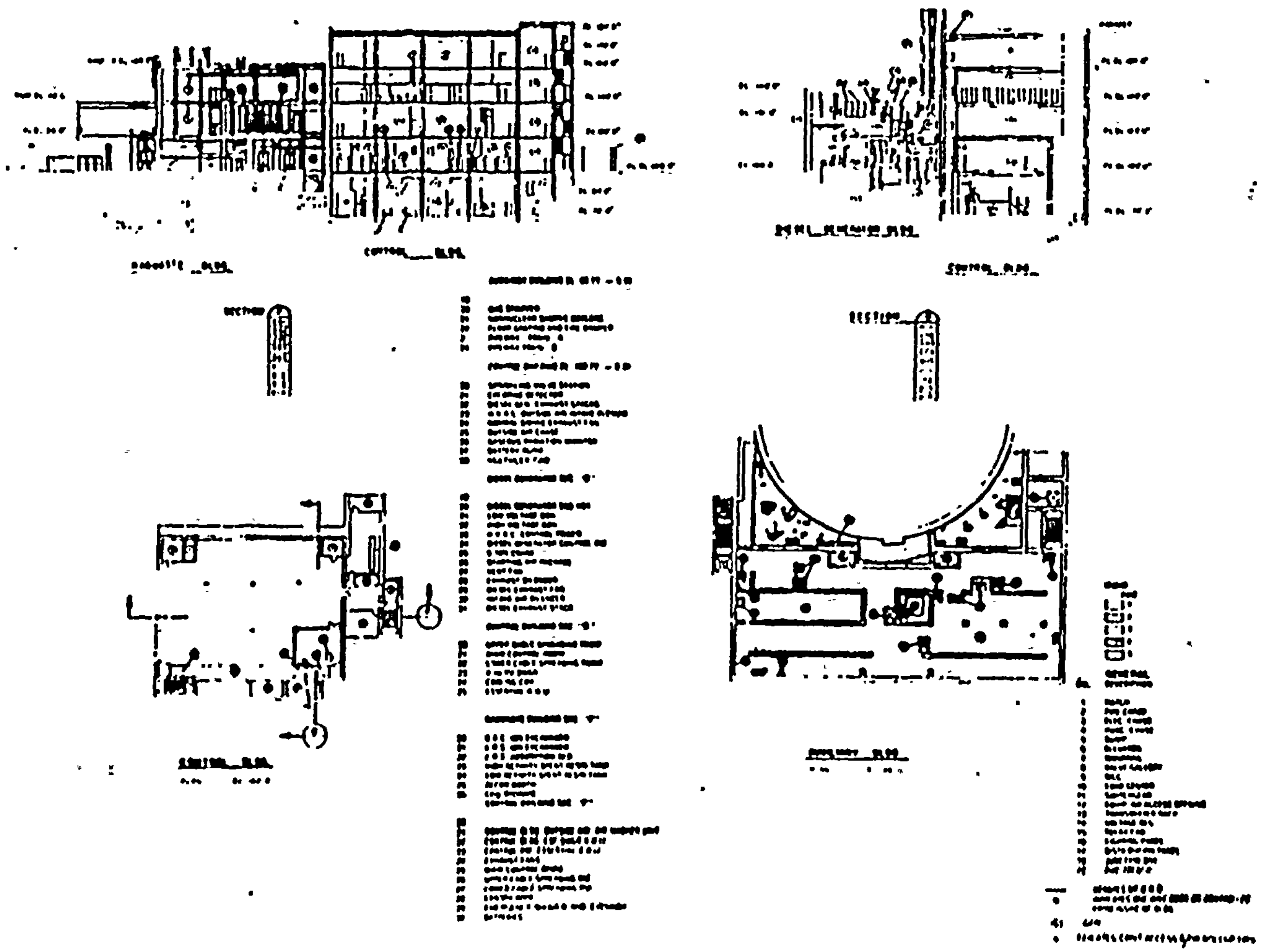
Figure 12.3-19

REPLACE WITH PROPOSED FIGURE 12.3-20



Palo Verde Nuclear Generating Station
FSAR

RADIATION ZONES (REFUELING)
PLAN AT AUX BLDG EL 88'-0", &
PLAN AT CONTROL BLDG EL 160'-0",
& SECTIONS F AND G
Figure 12.3-20

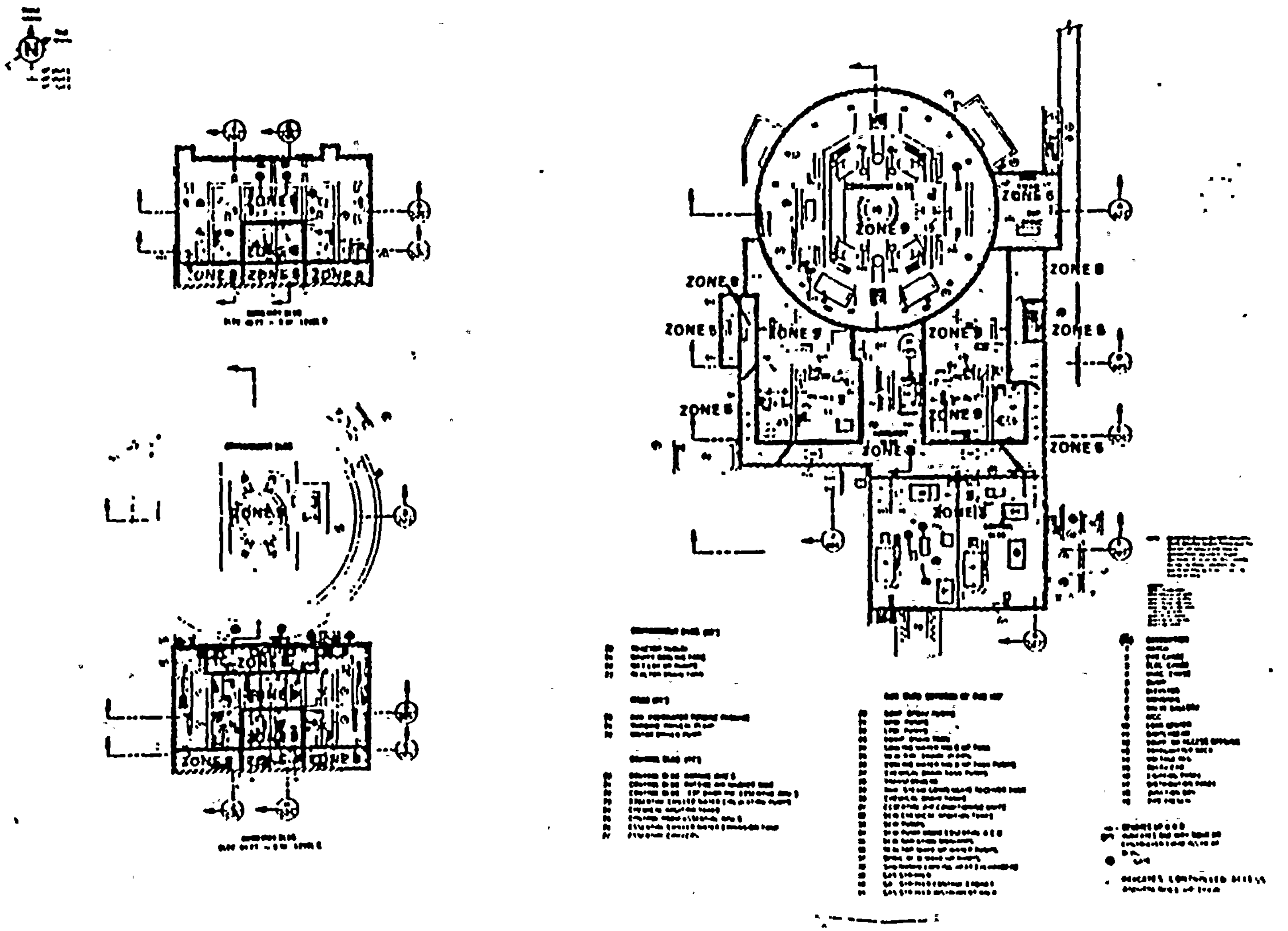


PROPOSED

13-N-RAR-010, REV.0

Palo Verde Nuclear Generating Station
FSAR

RADIATION ZONES (REFUELING)
AT AUXILIARY BLDG.
EL. 80'-0", CONTROL BLDG. EL. 160'-0",
AND SECTIONS F AND G
Figure 12.1-20

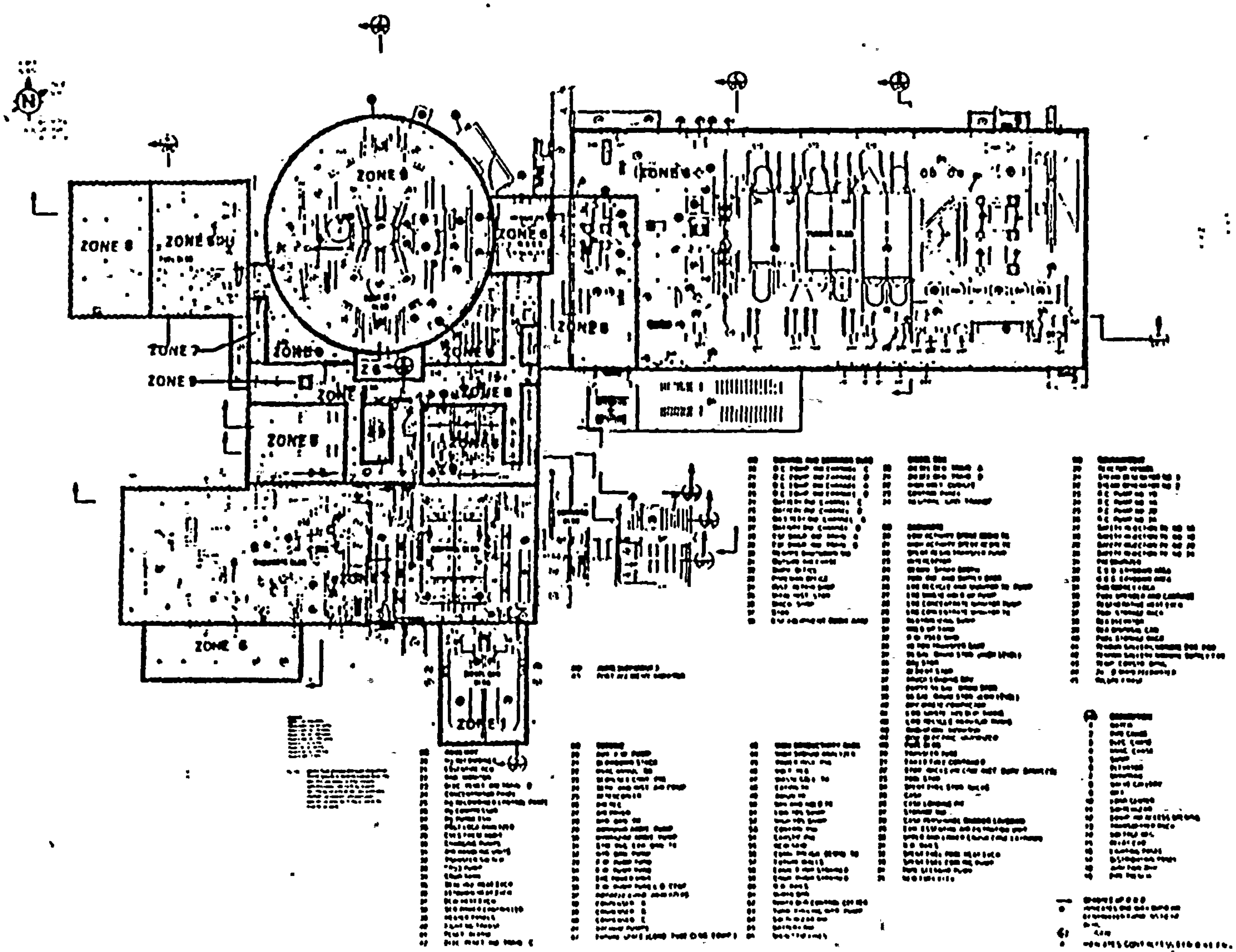


13-N-RAR-016, REV. 0

Palo Verde Nuclear Generating Station
FSAR

RADIATION ZONES
LOCA WITH SUMP RECIRCULATION
[illegible]

[illegible]



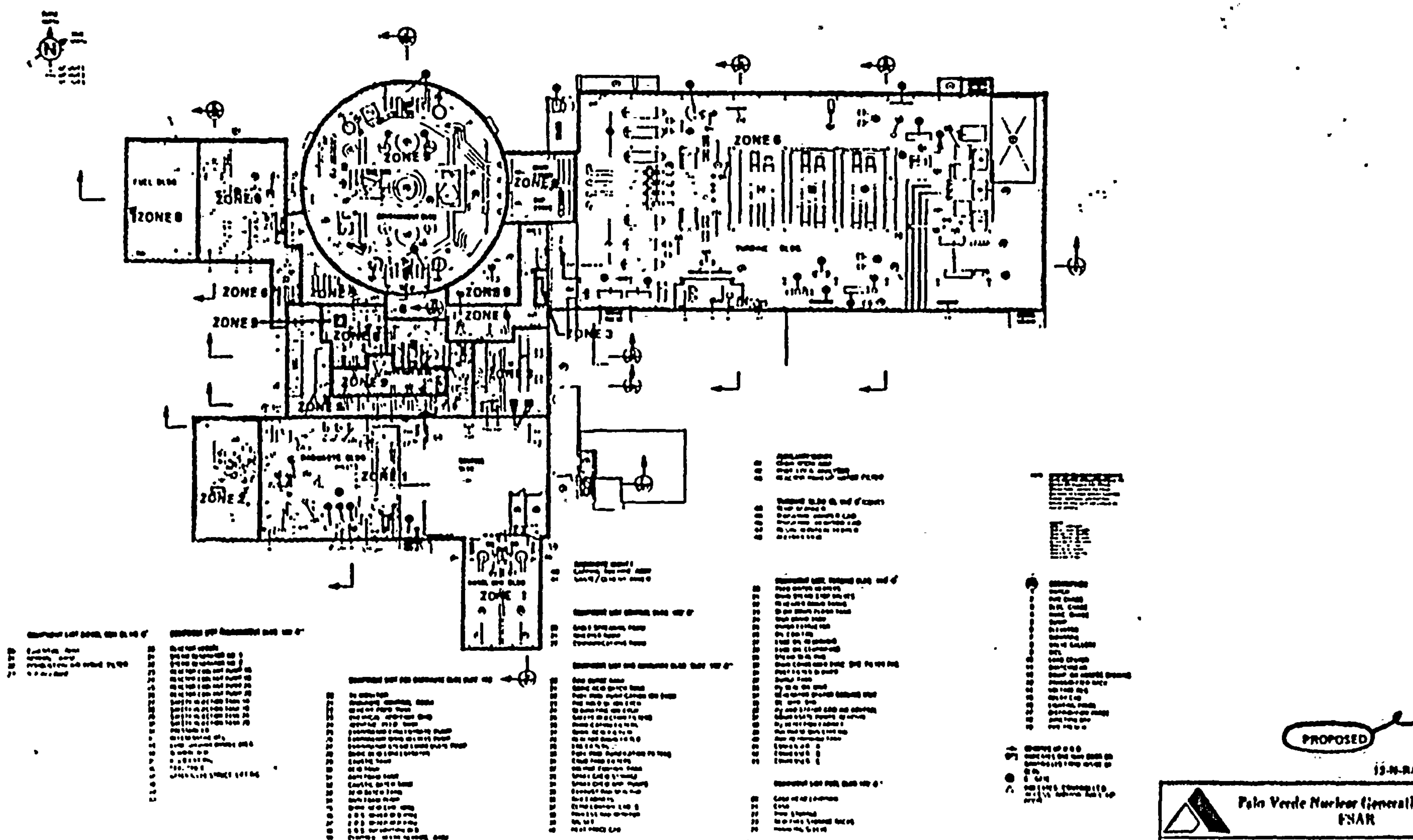
PROPOSED

13-N-RAR-019, REV. 0

Palo Verde Nuclear Generating Station
FSAR

RADIATION ZONES
LOCA WITH SIFP RECIRCULATION
AT EL. 100'-0"

Figure 12.3-26

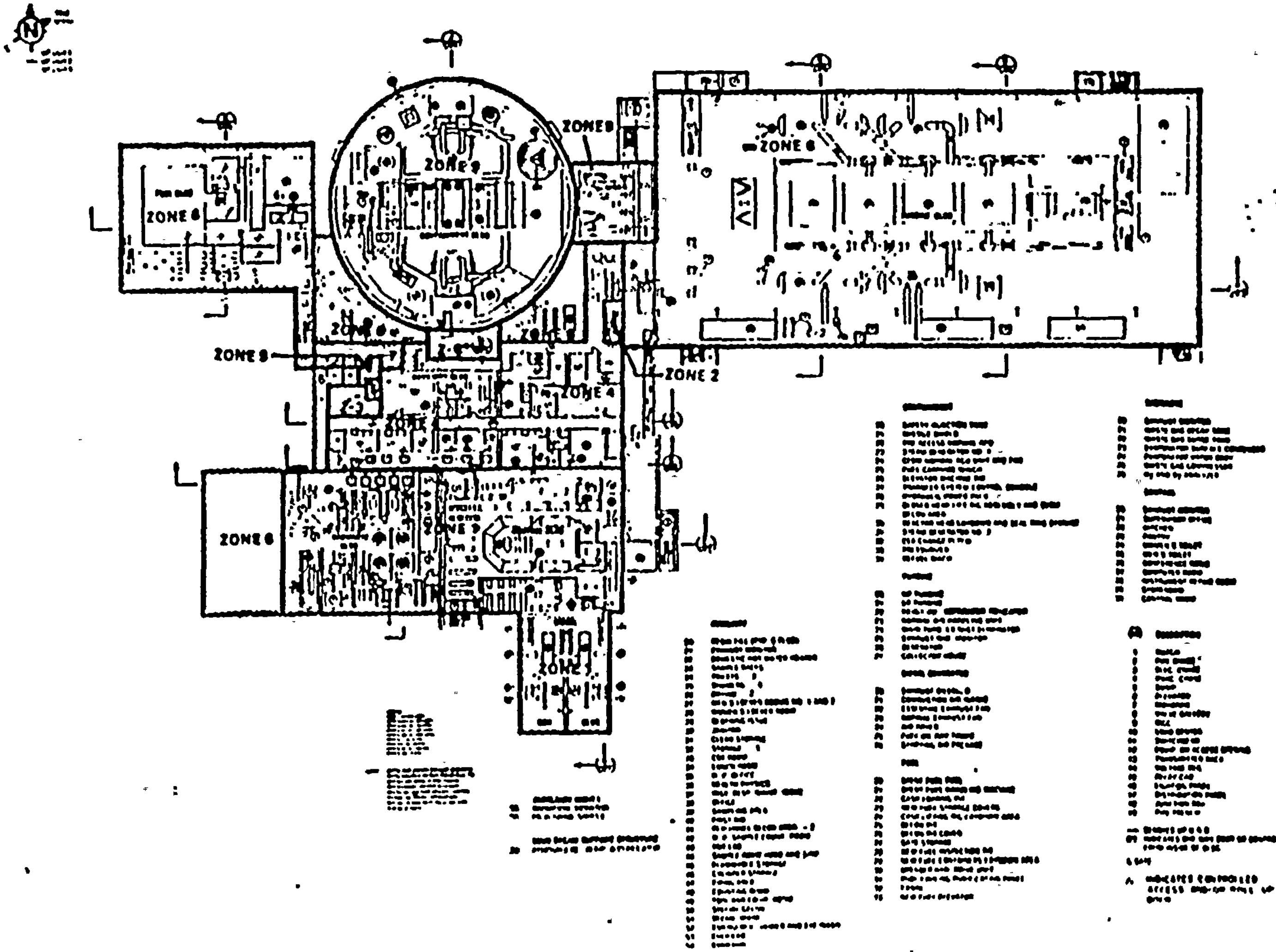


13-N-RAR-030, REV. 0

Palo Verde Nuclear Generating Station
FSAR

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

Figure 12.3-27



PROPOSED

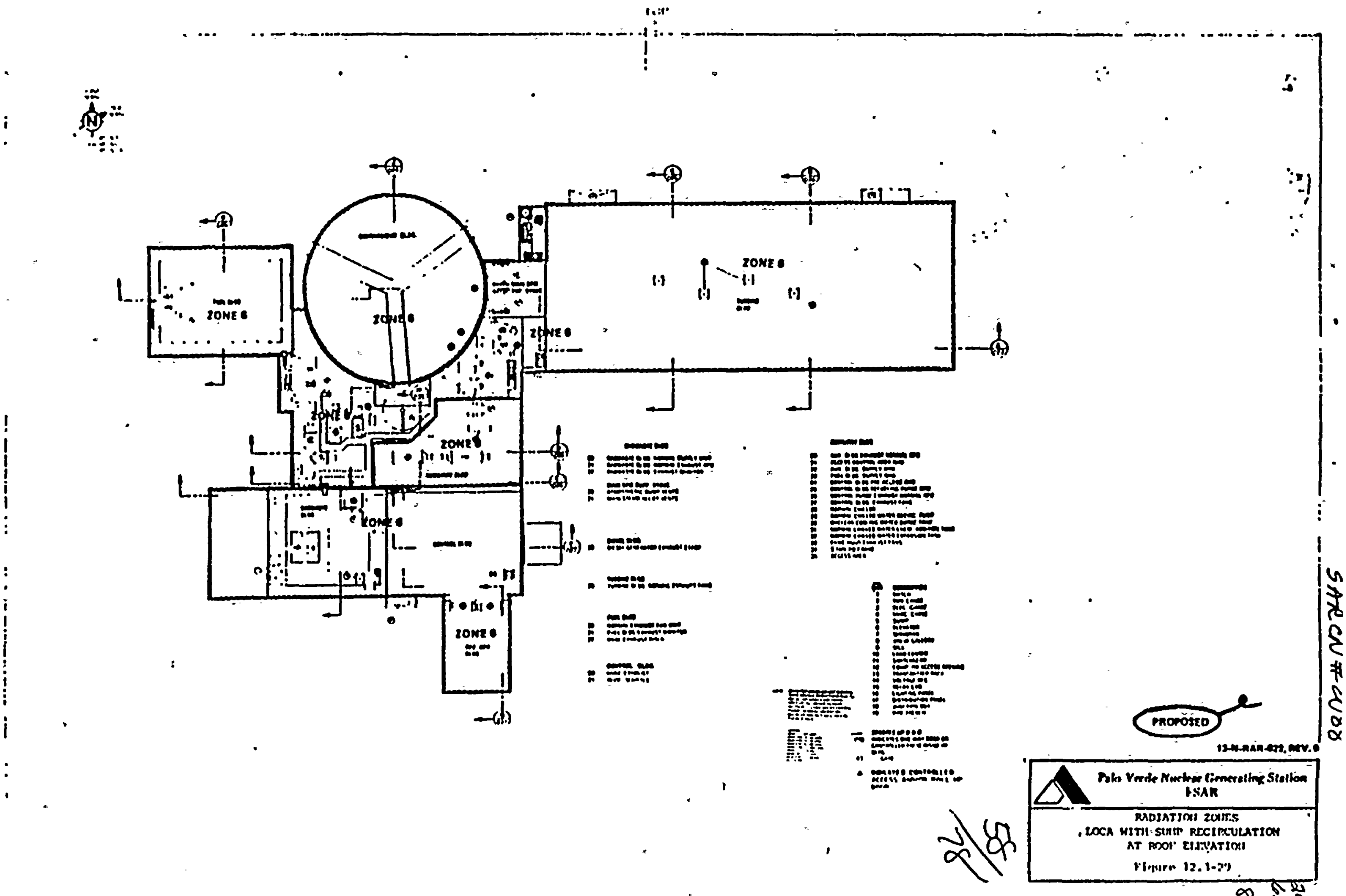
13-N-RAR-021, REV. 0

Palo Verde Nuclear Generating Station
FSAR

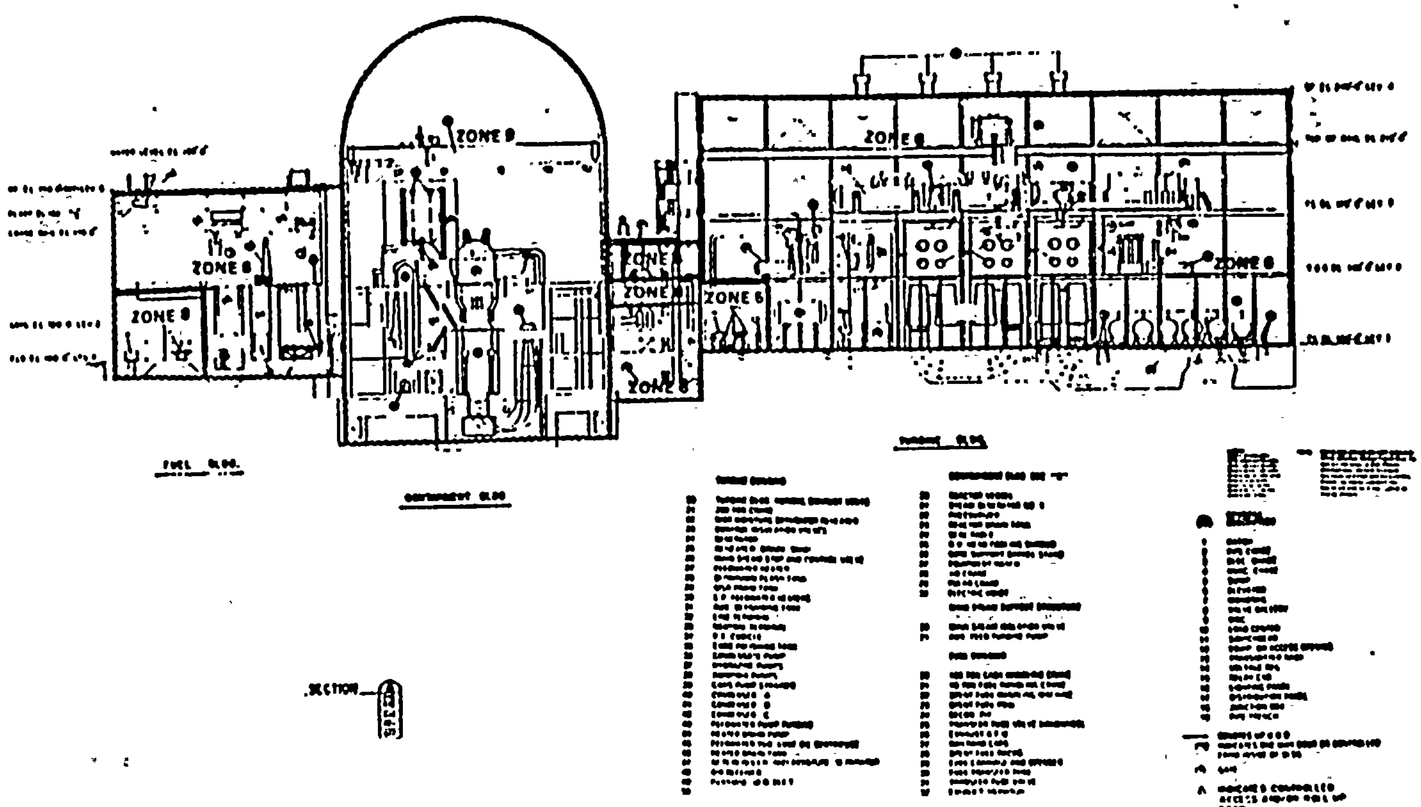
RADIATION ZONES
LOCA WITH SUMP RECIRCULATION
BETWEEN EL. 110'-0" & 200'-0"

Figure 12.3-20

SAHCN # 2025



ZONE 6
PROPOSED
13-N-RAR-022, REV. 0
Palo Verde Nuclear Generating Station
FSAR
RADIATION ZONES
LOCA WITH SUMP RECIRCULATION
AT ROOF ELEVATION
Figure 12.1-29



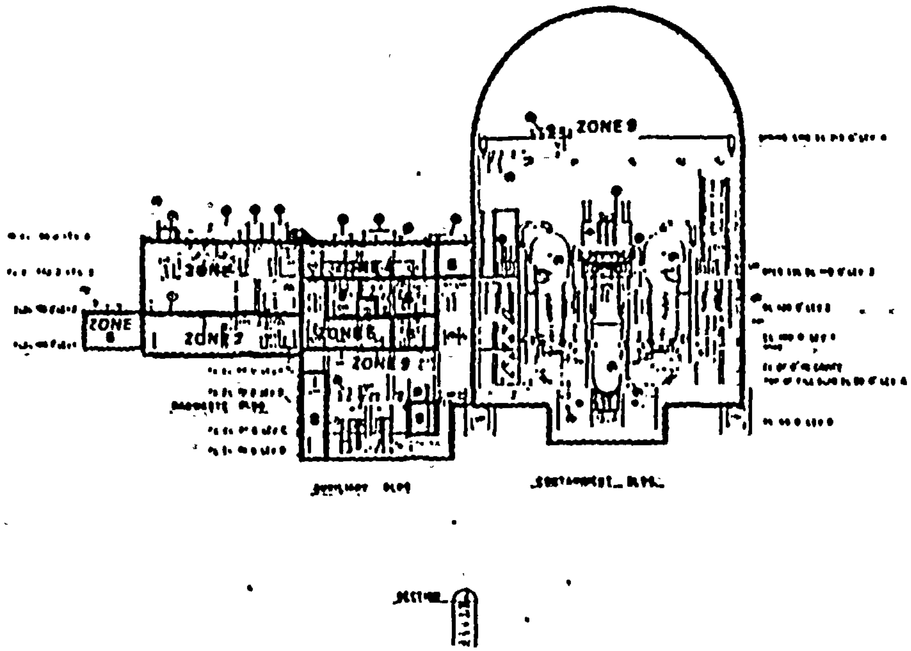
PROPOSED

12-M-RAR-022, REV. 0

Palo Verde Nuclear Generating Station
FSAR

RADIATION ZONES
LOCA WITH SUMP RECIRCULATION
SECTION A

Figure 12.3-[illegible]



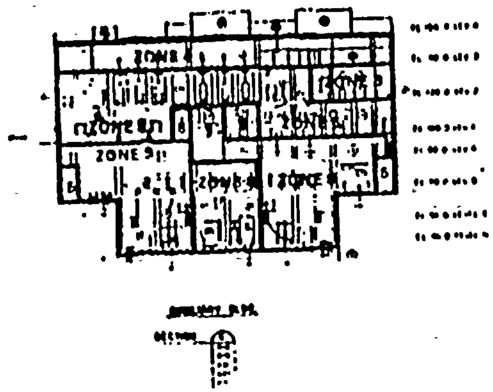
PROPOSED

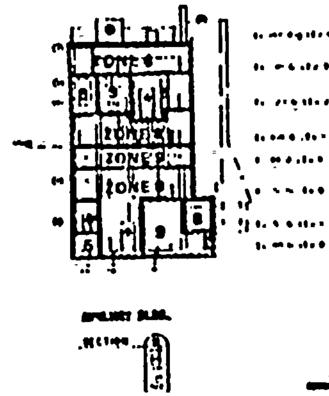
13-N-RAR-024, REV. 0

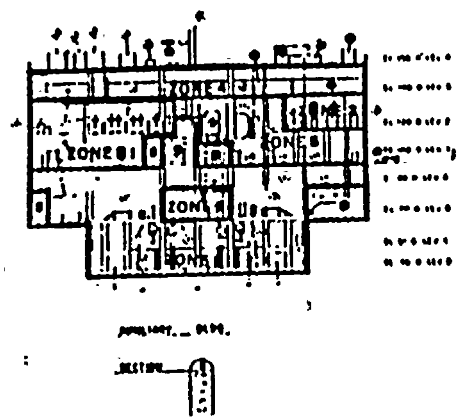
Palo Verde Nuclear Generating Station
FSAR

RADIATION ZONES
LOCA WITH SUMP RECIRCULATION
SECTION J-J

Figure 12.3-11







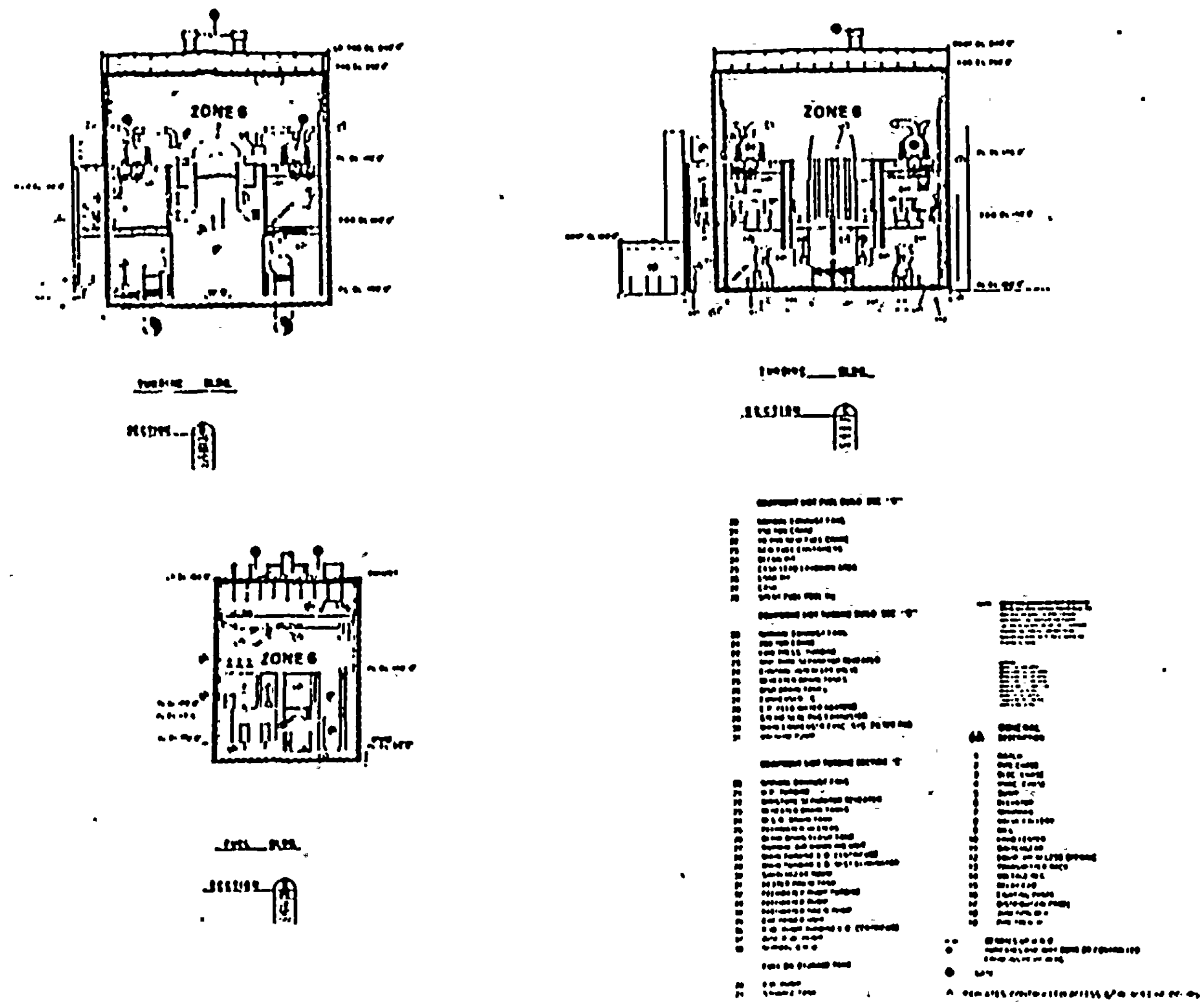
PROPOSED

13-N-RAR-020, REV. 0

Palo Verde Nuclear Generating Station
FSAR

RADIATION ZONES
LOCA WITH SUMP RECIRCULATION
SECTIONS D, E, H

Figure 12.3-17

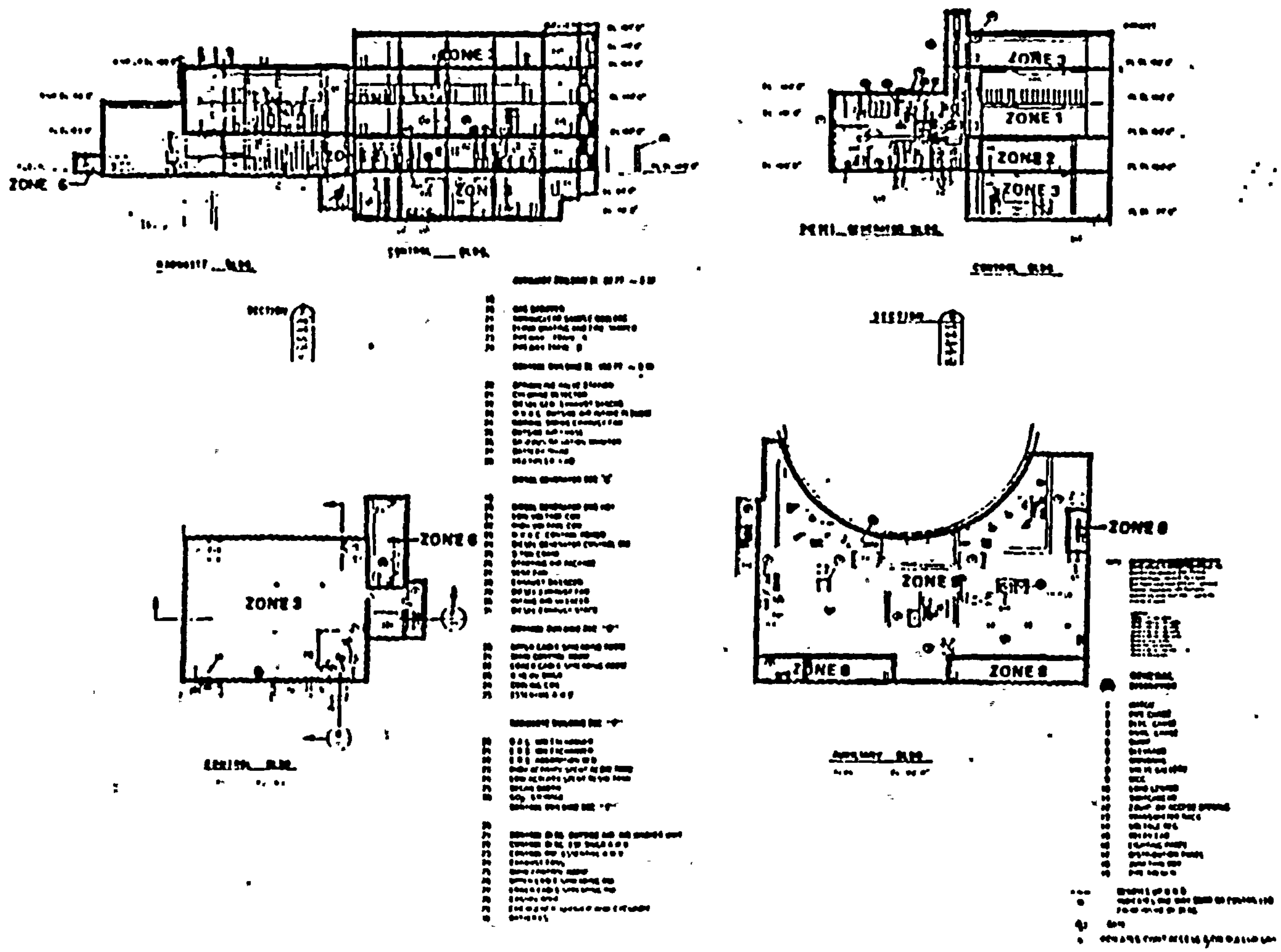


PROPOSED

Palo Verde Nuclear Generating Station
FSAR

RADIATION ZONES
LOCA WITH SUMP RECIRCULATION
SECTIONS B, C, E

Figure 12.3-11



PROPOSED

13-N-RAR-027, REV. 0

Palo Verde Nuclear Generating Station
FSAR

RADIATION ZONES
LOCA WITH SUMP RECIRCULATION
PLAN AT AUX. 100', CONTROL 1[illegible]',
SECTS. F, G
Figure 12.[illegible]

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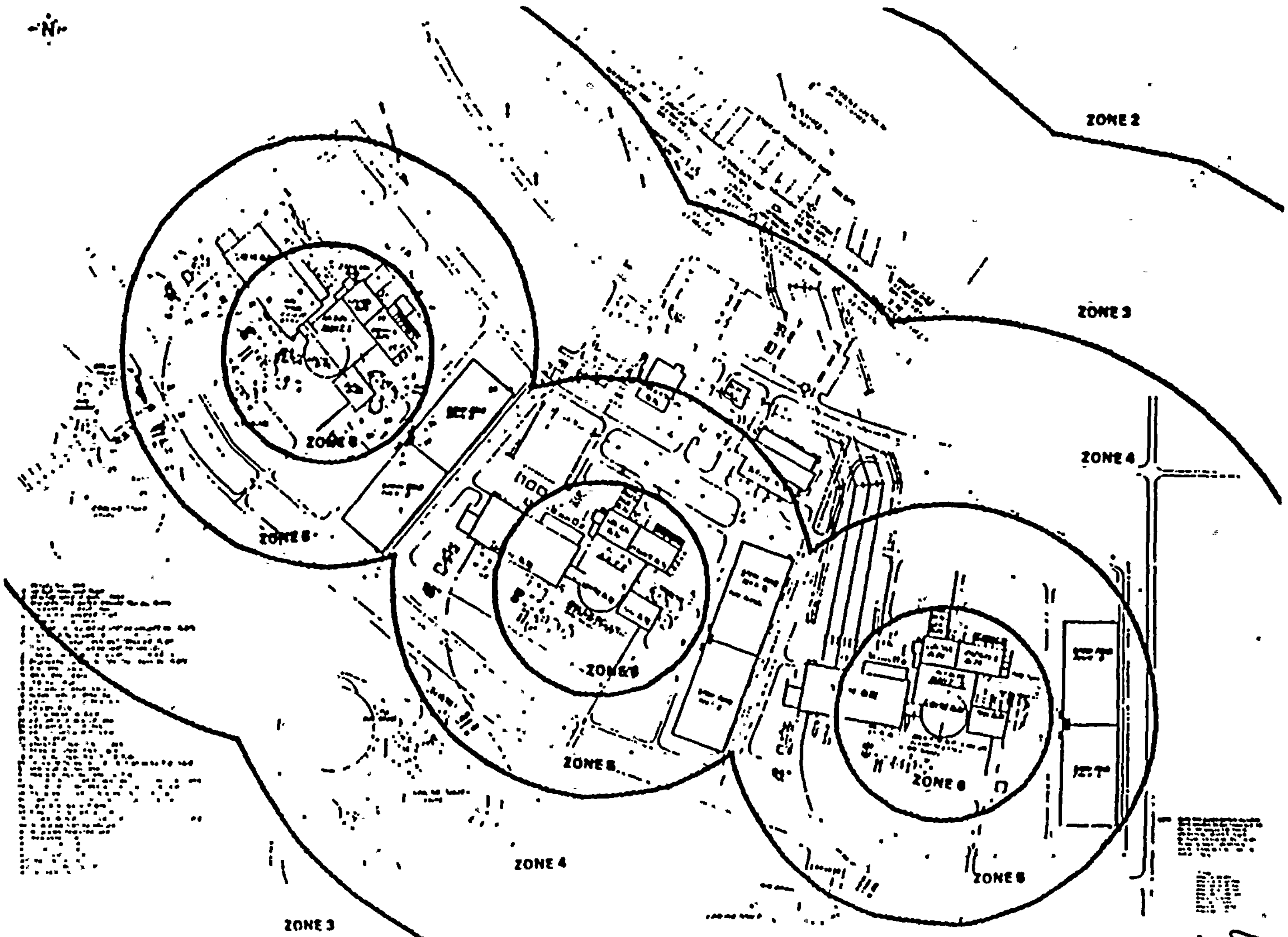
2088



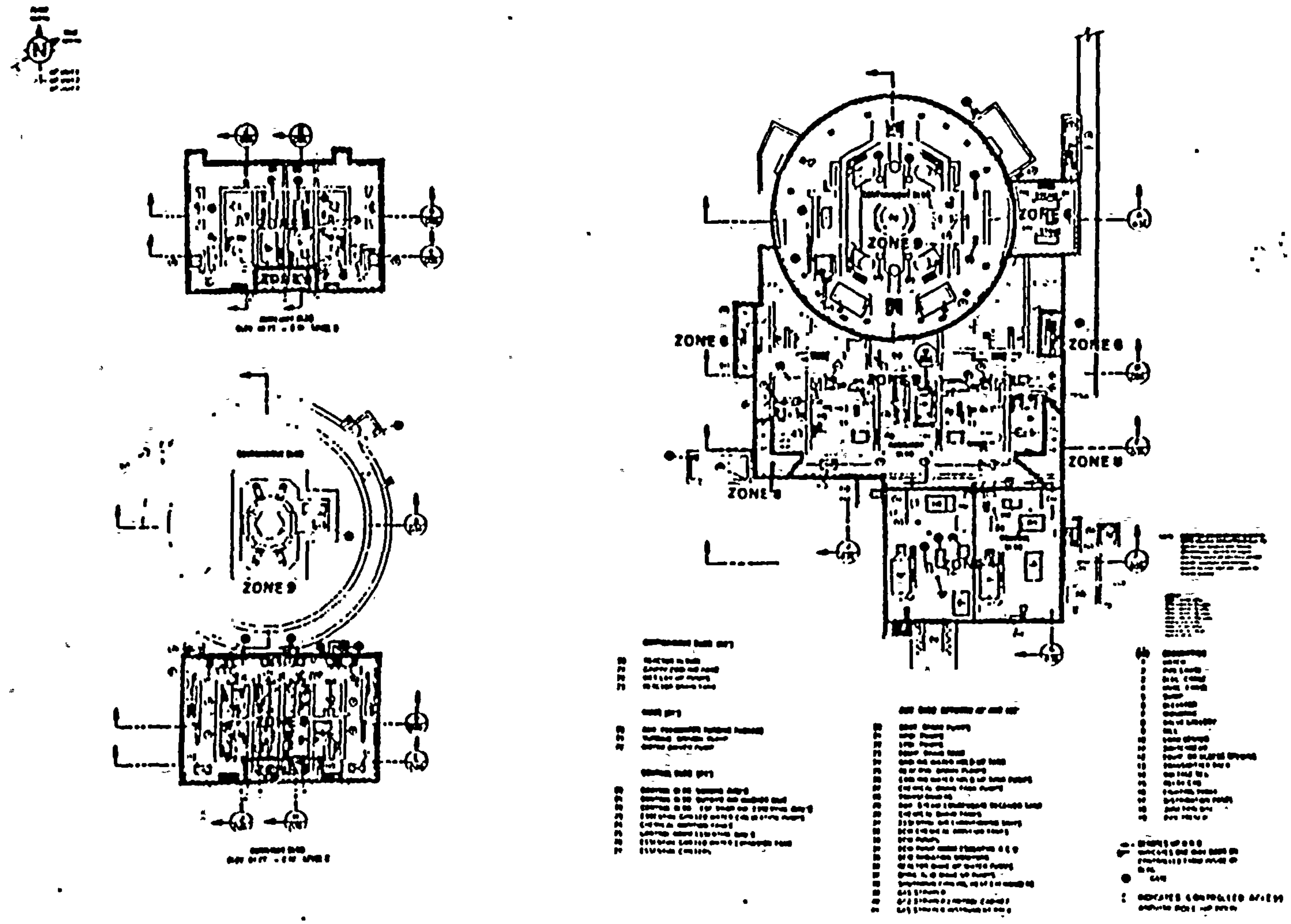
PROPOSED

Palo Verde Nuclear Generating Station
FSAR

RADIATION ZONES-LOCA

OUTSIDE AREAS
(DIRECT DOSE FROM BUILDINGS ONLY)

Figure [illegible]



PROPOSED

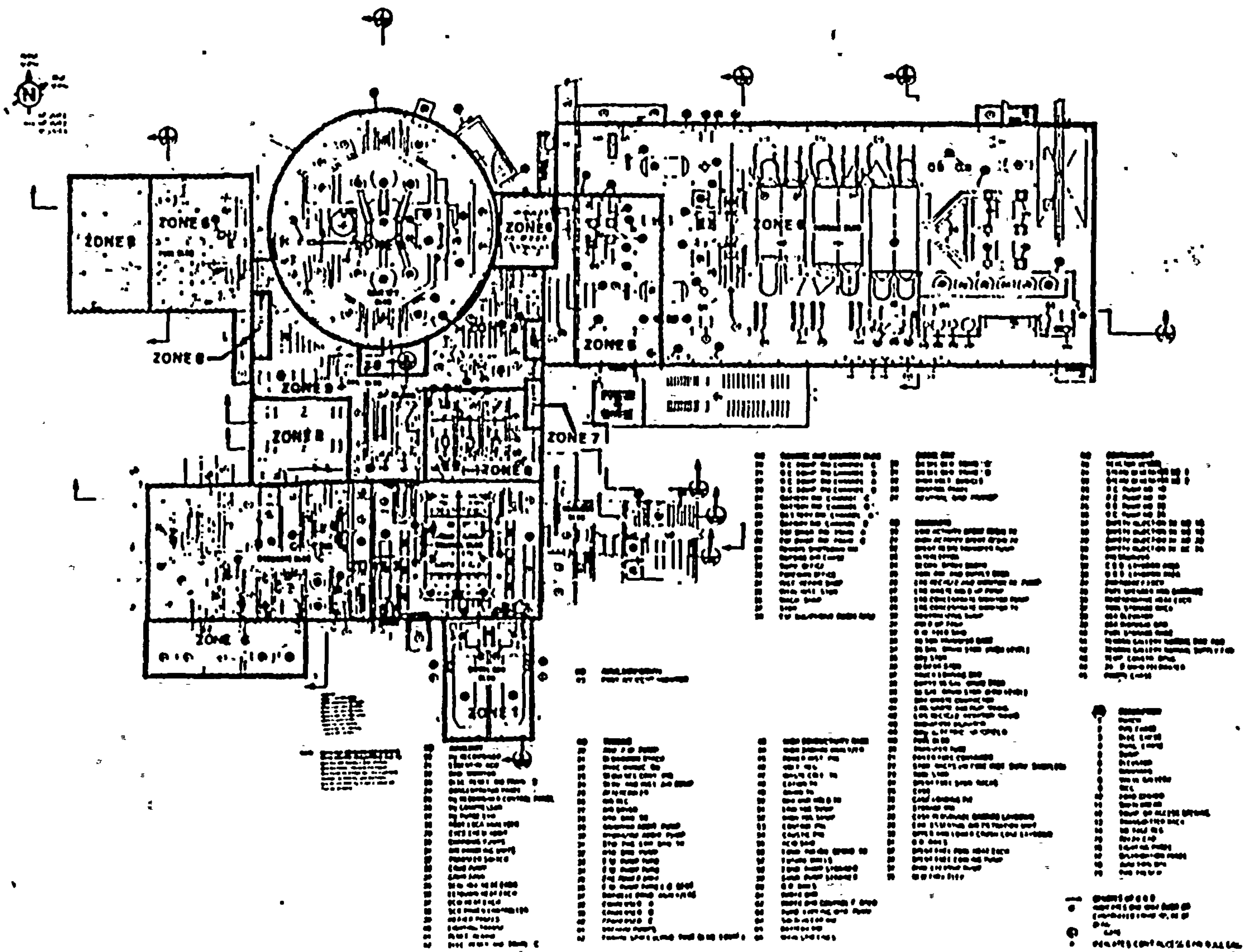
12-N-RAR-020, REV. 0

Palo Verde Nuclear Generating Station
FSAR

RADIATION ZONES-LOCA
DEGRADED CORE- INTACT PRIMARY
BETWEEN EL. 40'-0" & 100'-0"

Figure 12.3-36

SARCN #2088



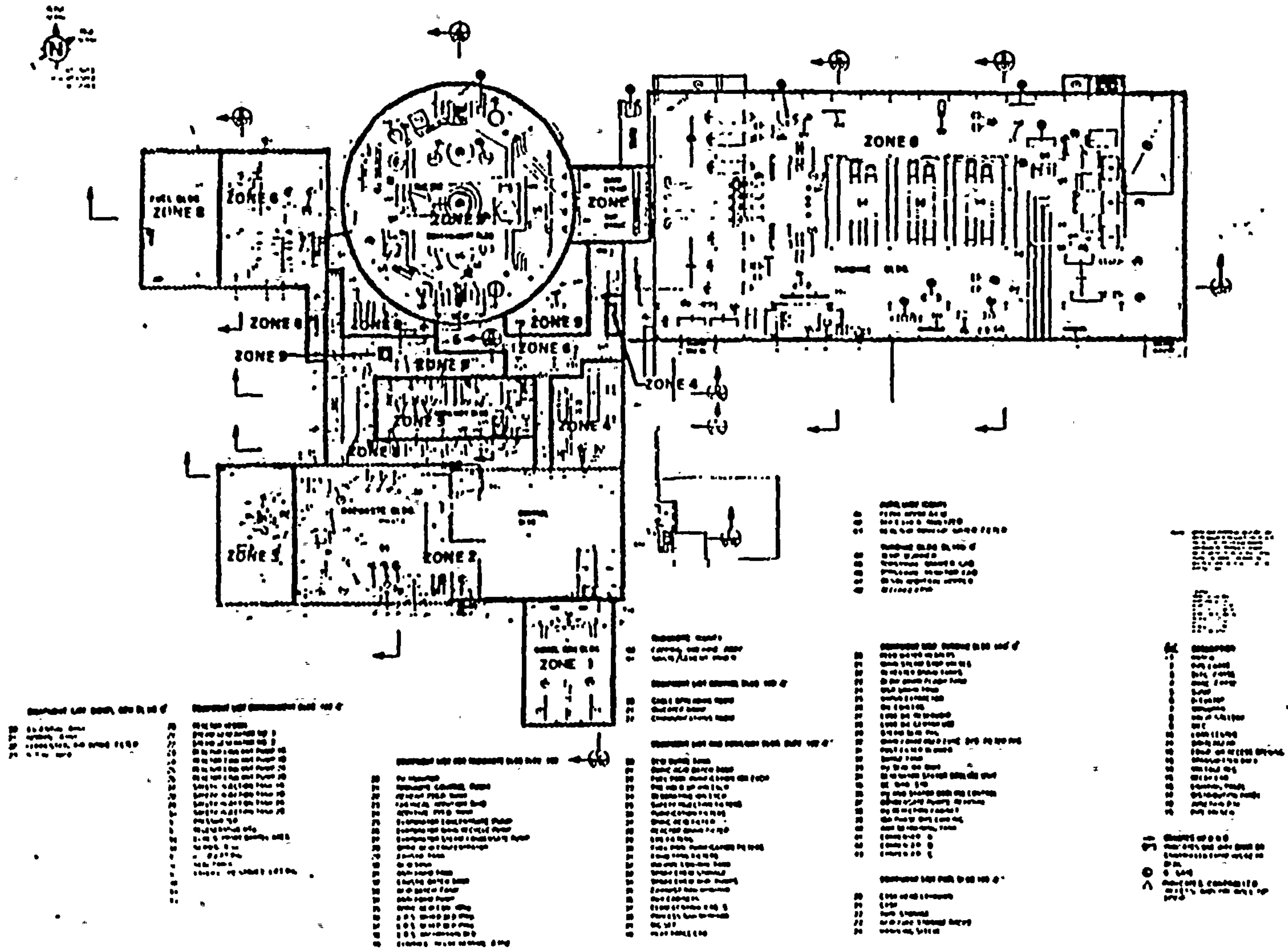
PROPOSED

13-N-RAR-030, REV. 0

Palo Verde Nuclear Generating Station
FSAR

RADIATION ZONES-LOCA
DEGRADED CORE-INTACT PRIMARY
AT EL. 100'-0"

Figure 12.3-17



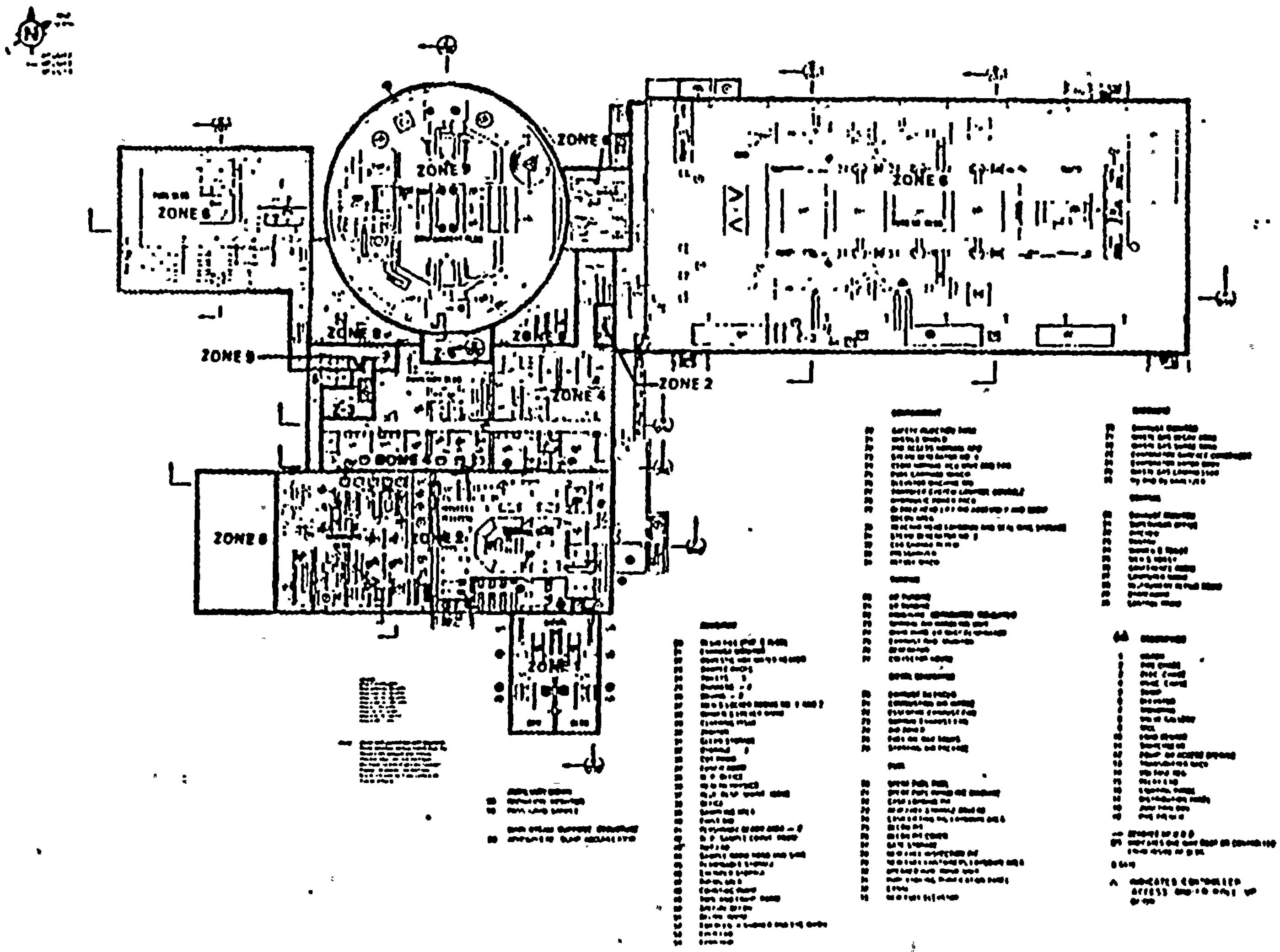
PROPOSED

13-N-RAR-031, REV. 0

Palo Verde Nuclear Generating Station
FSAR

RADIATION ZONES-LOCA
DEGRADED CORE-INTACT PRIMARY
DETAILED EL. 120'-0" & 140'-0"

Figure 12.3-14



PROPOSED

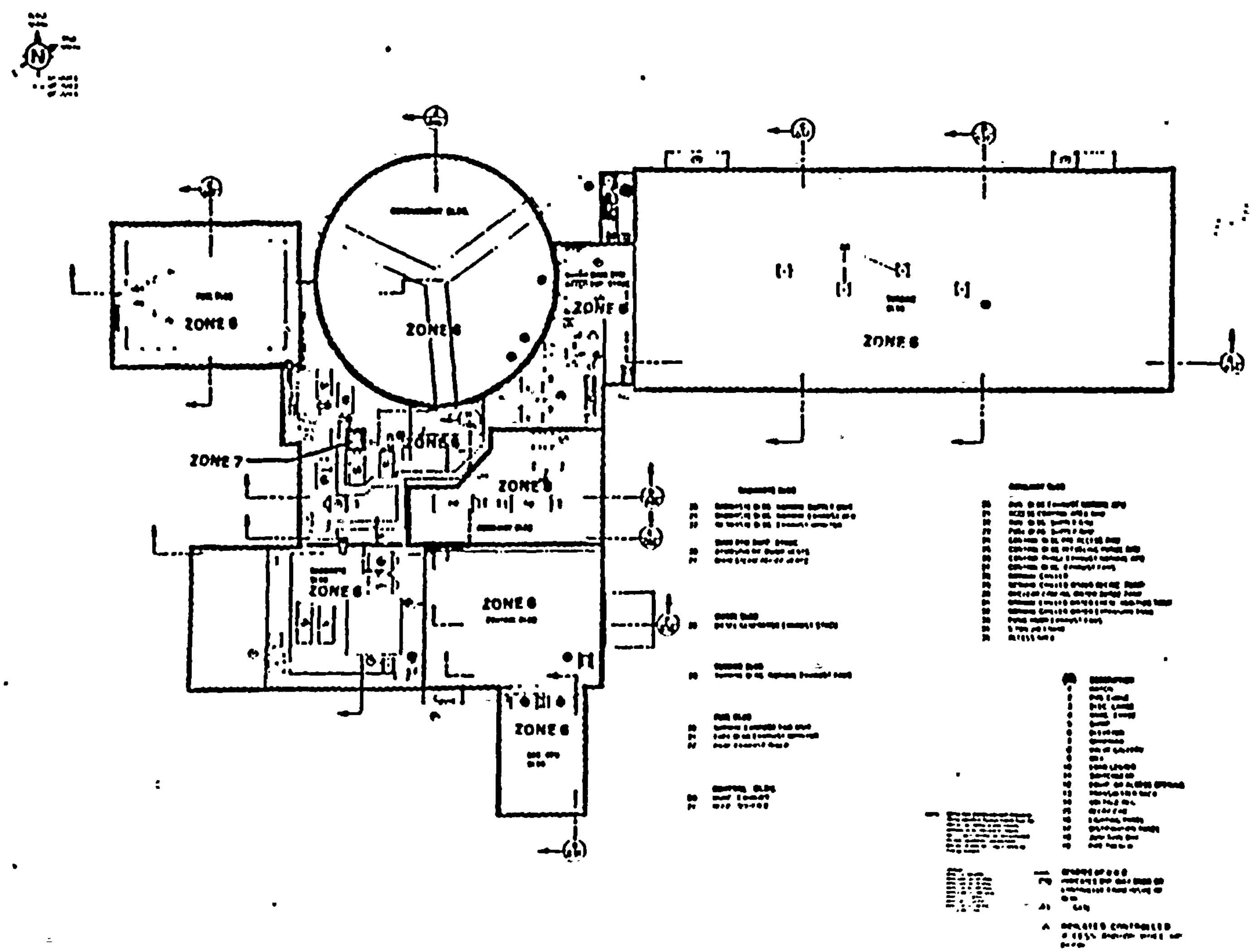
13-N-RAR-022, REV. 0

Palo Verde Nuclear Generating Station
FSAR

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

Figure 12.1-19

SARCN #2088



PROPOSED

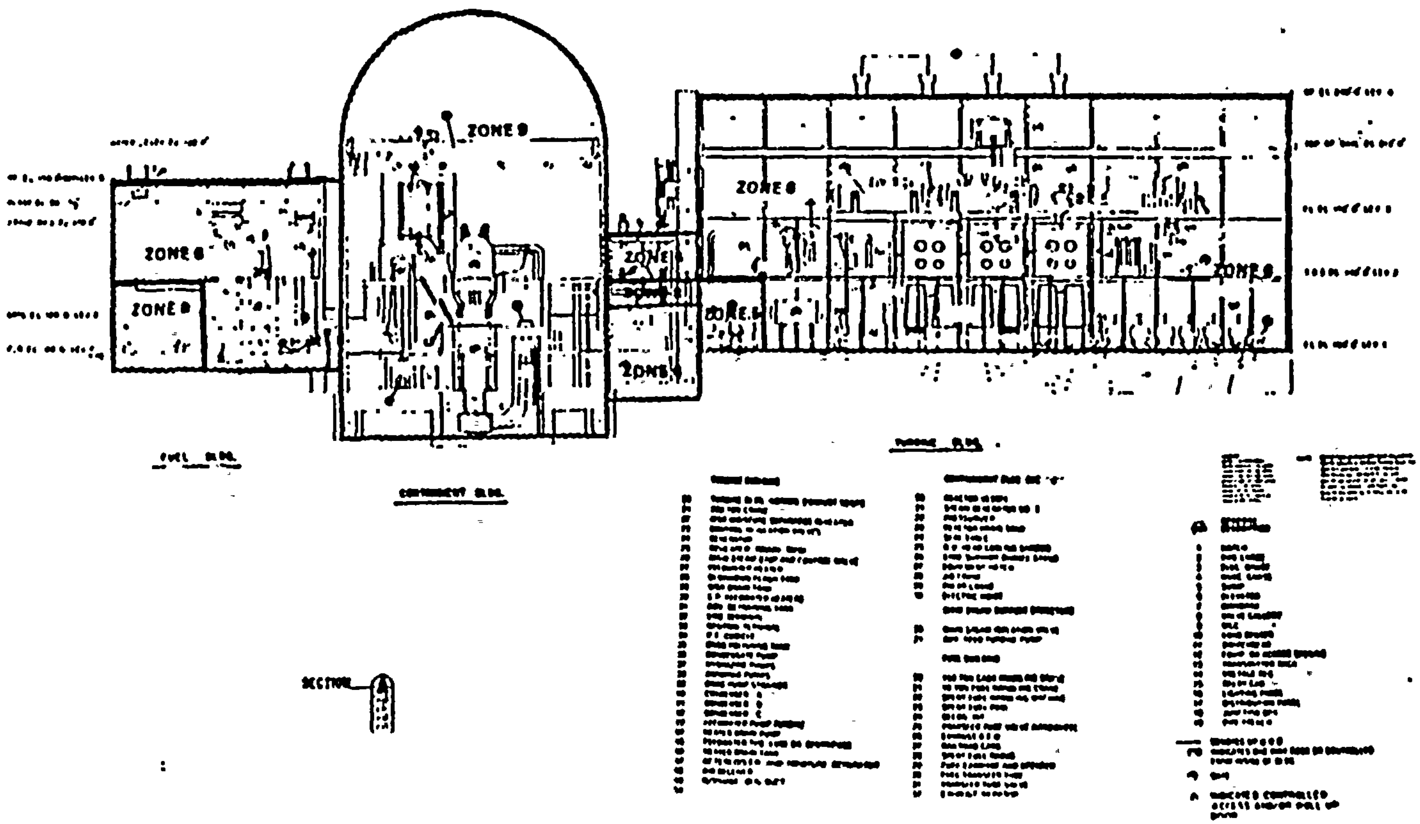
13-N-RAR-033, REV. 0

Palo Verde Nuclear Generating Station
FSAR

RADIATION ZONES-LOCA
DEGRADED CORE-INTACT PRIMARY
AT ROOF ELEVATION

Figure 12.3-49

SARCN#2088



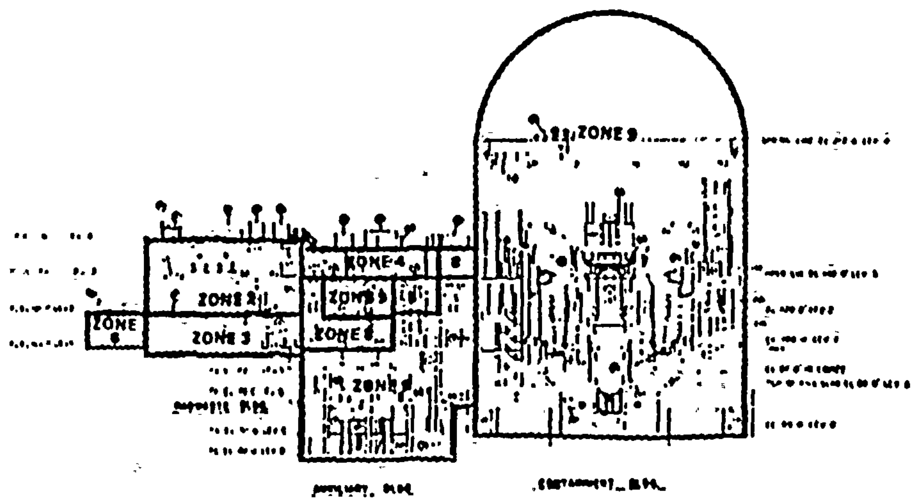
PROPOSED

13-N-HAR-030, REV. 0

Palo Verde Nuclear Generating Station
FSAR

RADIATION ZONES-LOCA
DEGRADED CORE-INTACT PRIMARY
SECTION A

Figure 12.3-41



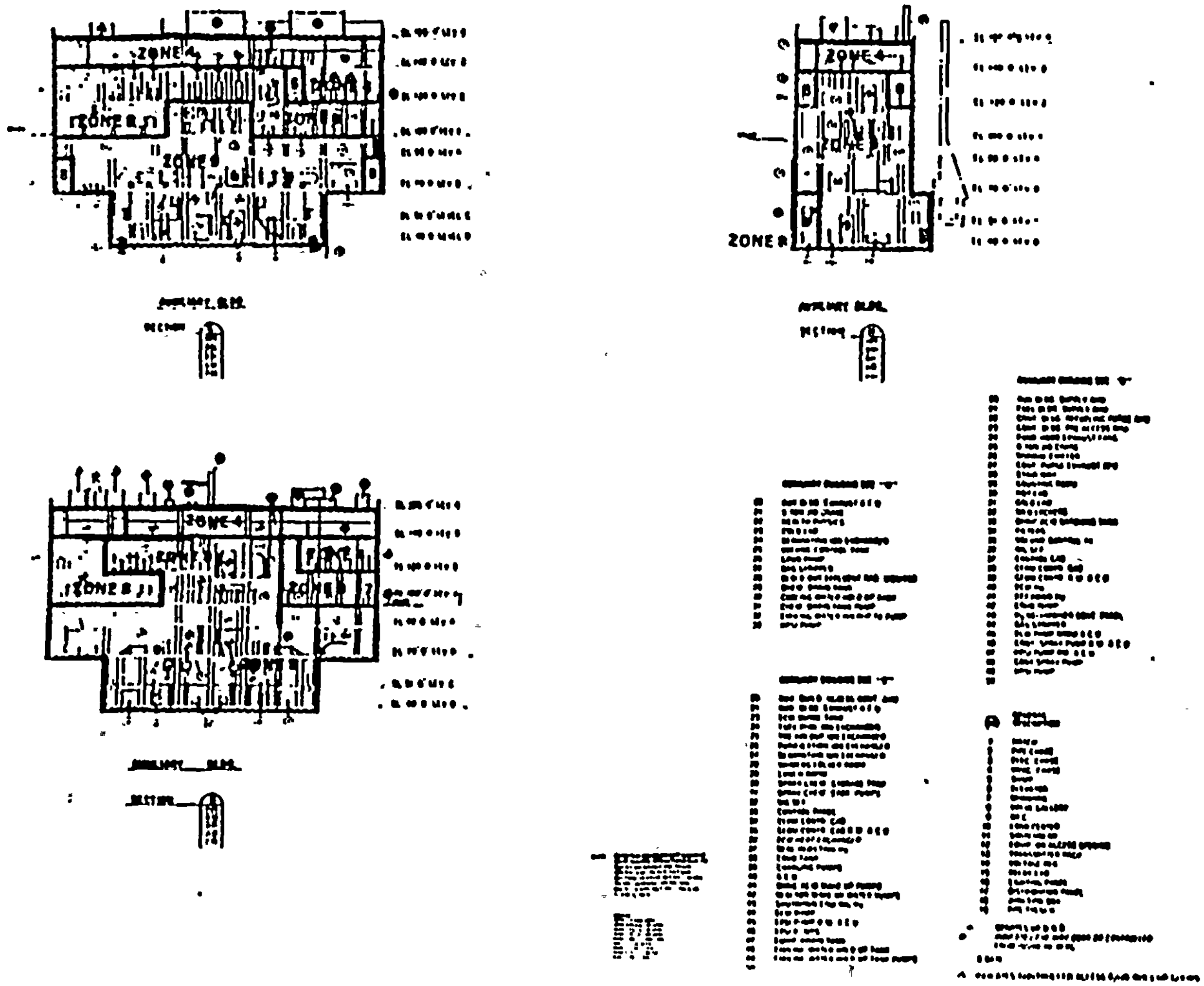
PROPOSED

13-N-RAR-038, REV. 0

Palo Verde Nuclear Generating Station
FSAR

RADIATION ZONES-LOCA
DEGRADED CORE-INTACT PRIMARY
SECTION J-J

Figure 12.3-42



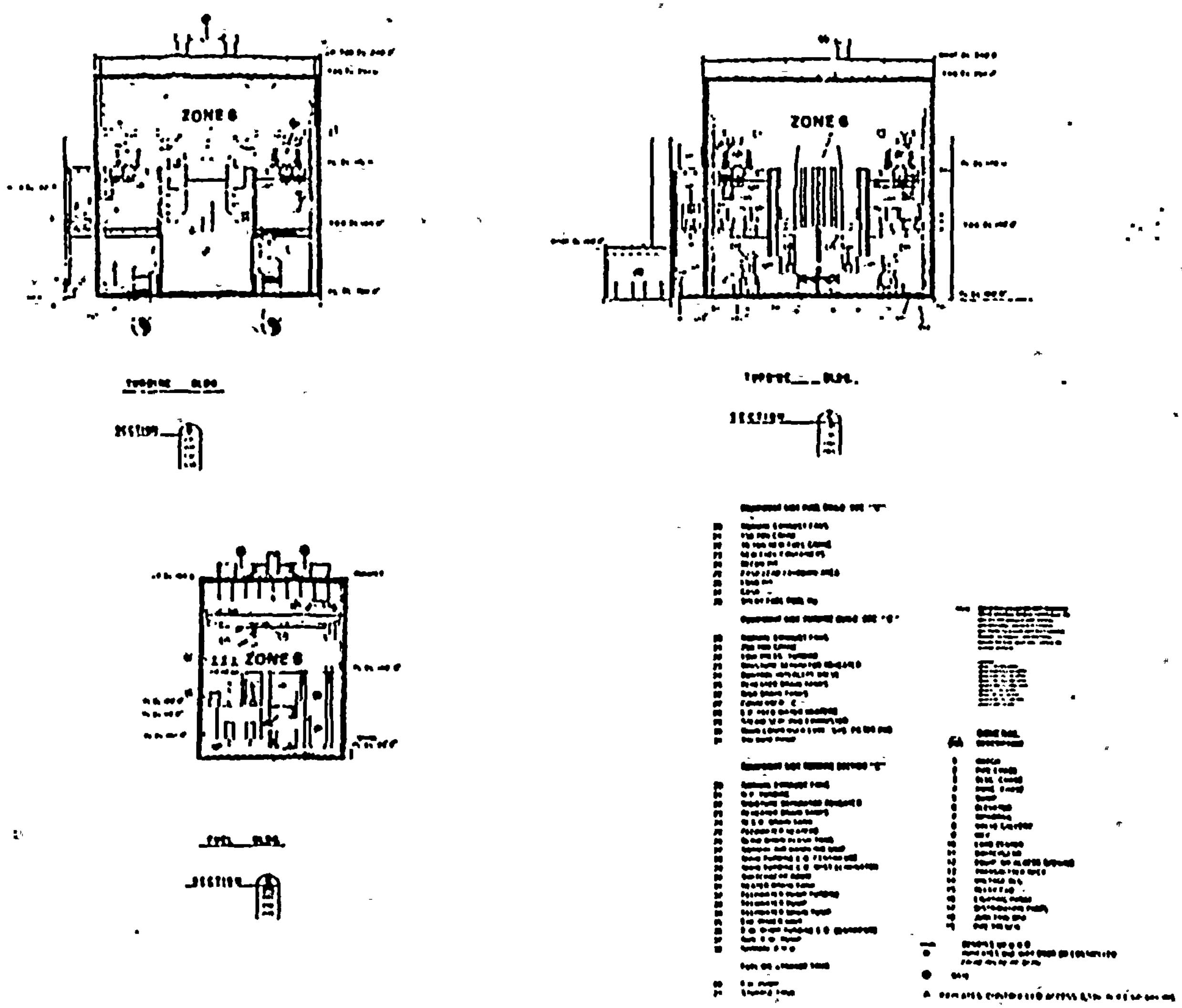
PROPOSED

13-N-RAR-038, REV. 0

Palo Verde Nuclear Generating Station
FSAR

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

Figure 12.3-43



PROPOSED

13-N-RAR-037, REV. 0

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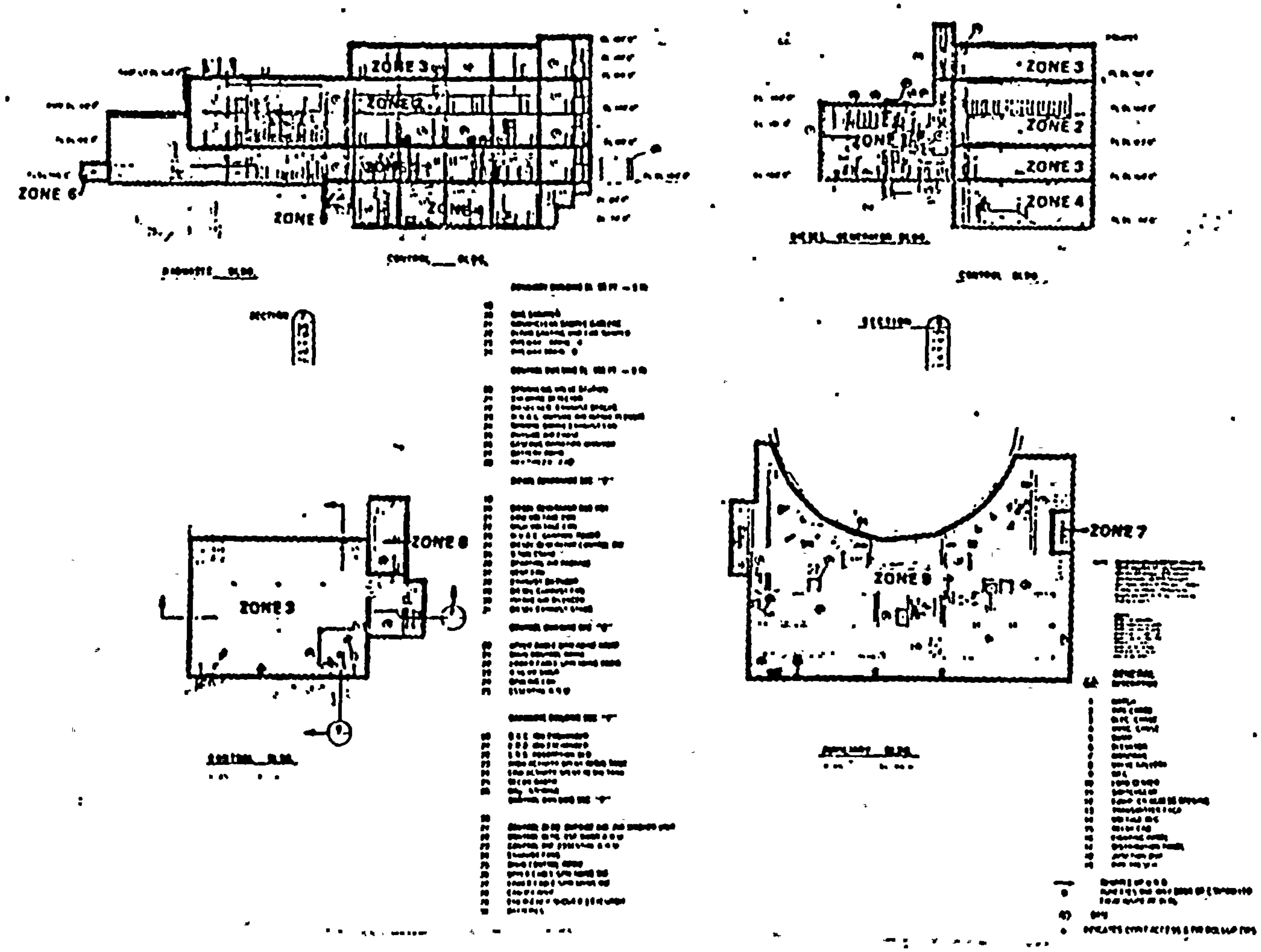
RADIATION ZONES-LOCA
DEGRADED CORE-INTACT PRIMARY
SECTIONS B, C, K

Figure 12.3-41



PROPOSED

Palo Verde Nuclear Generating Station
FSAR

RADIATION ZONES-LOCA
DEGRADED CORE-INTACT PRIMARY
PLAN AT AUX. 88', CONTROL 160',
SECTS. F, G
Figure 12.3-45

RESPONSE: The response is given in amended sections 12.1.1.2, 12.5.1.1, figure 13.1-6, figure 13.1-7, 13.1.2.2.2.2, and 13.1.3.1:

The Radiation Protection Section is a separate organization from the Chemistry Section, and the Radiation Protection Supervisor reports to the Engineering and Technical Services Manager who is independent of the Station Operations and Maintenance Departments. The Radiation Protection Supervisor has direct access to the Manager of Nuclear Operations in matters relating to radiological protection and ALARA programs as authorized in the Station Manual. He is a permanent member of the Plant Review Board.

QUESTION 12A.3 (NRC Question 471.2) (12.5)

You should describe your plan to provide backup coverage in
5 the event of absence of the RPM, and you should outline the qualifications of the individual who will act as the backup. The December 1979 revision of ANSI 3.1 specified that the temporary replacement for an RPM should have a BS degree in science or engineering, 2 years experience in radiation protection, 1 year of which should be nuclear power plant experience, 6 months of which should be onsite. It is our position that this experience be professional experience.

RESPONSE: The response is given in amended section 13.1.2.2.2.2. The minimum requirements for the position providing backup coverage in event of absence of the RPM is discussed in section 13.1.3.1.

QUESTION 12A.4 (NRC Question 471.3) (12.3)

You should provide information in response to TMI Lessons Learned review for the following NUREG-0737 areas: II.B.2 Post Accident Shielding and Vital Area Access; II.B.3 ALARA for

Post-Accident Sampling; II.F.1 High Range In-Containment Radiation Monitors; ~~and~~ III.D.3.3 Post-Accident Iodine Sampling and Analysis. Sections 18.II.B.2, 18.II.B.3, 18.II.F.1, and 18.III.D.3.3.

RESPONSE: The response is provided in ~~the PVNGS LLIR. 9~~ Additional information is provided in sections 12.1.2.4 and 12.3.1.3 for Items II.B.2 and II.B.3, in section 11.5 and figure 12.3-4 for II.F.1, and in sections 9.3.2.2.2 and 11.5 for III.D.3.3.

QUESTION 12A.5 (NRC Question 471.4) (12.1)

Paragraph B, of Section 12.1.2.1.2, stating that "as minimum, shielding is designed to reduce gamma dose rates from sources external to a radioactive compartment to levels comparable to dose rates resulting from equipment within that compartment," is not clear.

It appears that this could refer to shielding between two adjacent compartments in which radioactive equipment is located. If this is the case, then shielding should be designed to reduce radiation from the operating equipment in one compartment to levels below that which is expected in the adjacent compartment from the shutdown equipment to be maintained or repaired. Please clarify. 5

RESPONSE: Paragraph B of section 12.1.2.1.2 has been revised to provide the requested clarification.

QUESTION 12A.6 (NRC Question 471.5) (12.1)

In accordance with Section C.2.e, "Crud Control," of Regulatory Guide 8.8, it is our position that consideration should be given to the selection of corrosion resistant, low cobalt content alloys to reduce the concentrations of radioactive corrosion product buildup in systems.

Section 12.1.2.3, "Equipment General Design Considerations for ALARA," of your FSAR, should be revised to reflect your design considerations for selection of low cobalt alloys.

18.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~~ detector (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 1
instrumentation.

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

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1 instrument bus with Class IE backup power. The system is 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~~ are completed prior to fuel loading ~~of PVNGS Unit 1.~~ for each of the PVNGS units

18.II.F INSTRUMENTATION AND CONTROLS

18.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 monitoring do not increase the potential for operator error (see section 18.II.D.1). Installation will be completed prior to fuel load.

18.II.F.1.1 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

Section 11.5 provides detailed descriptions of the effluent monitors installed at Palo Verde Units 1, 2 and 3. This includes the additional monitors that have been added specifically to address NUREG-0737 and Regulatory Guide 1.97, Rev. 2 requirements for radiation monitoring. A description of the calibration sources, frequency of calibration, and

technique is provided in table ~~11.5.1~~ 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 table 11.5-1.

Sampling of effluents meets the criteria of ANSI N13.1-1969 as discussed in sections 11.5.2.1.1.7.2 and 11.5.2.2.1. Section 11.5.2.1.1.7.2 also describes the sampling assembly.

A description of effluent radiation monitoring is presented in section 11.5.2.1.4. Included in this section are discussions of monitors located on the plant vent, main condenser/gland seal exhaust and fuel building vent ~~and the main steam line~~. These monitors operate in conjunction with other monitors, as described in section 11.5, and fulfill the requirements as outlined in NUREG-0737 and Regulatory Guide 1.97, Rev. 2.

(~~tag no's XJ-SQN-RU-141 and XJ-SQN-RU-142~~) (~~tag no's XJ-SQN-RU-143 and XJ-SQN-RU-144~~) (~~tag no's XJ-SGB-RU-145 and XJ-SGB-RU-146~~)

The main steam line monitors (~~tag no's XJ-SQN-RU-139 and XJ-SQN-RU-140~~) are area monitors required for post-accident monitoring and are discussed in section 11.5.2.1.5.

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18.II.F.1.2 SAMPLING AND ANALYSIS OF PLANT EFFLUENTS

Position

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.

PVNGS Evaluation

The ~~E~~ffluent sampling assembly ~~monitoring of radioiodines~~ is discussed in section 11.5.2.1.1.7.2. For particulate and iodine channels, the sampler is a lead-shielded filter assembly. Four ~~1/2~~ shielding is furnished for all process and effluent detectors.

Airborne particulate and iodine monitors and samplers, (XJ-SQN-RU-08, -RU-14, -RU-141, -RU-142, -RU-143, -RU-144, and XJ-SQB-RU-145 and -RU-146) sample isokinetically in accordance with the principles and methods of ANSI N13.1-1969, Guide to Sampling Airborne Radioactive Materials in Nuclear Facilities. The particulate and iodine sample flow is maintained constant over the normal expected range of filter paper and/or charcoal cartridge differential pressure by an automatic control system. Local flow indication and high- and low-sample flow alarm signals are provided. These signals actuate local alarms and the channel failure alarm. Two particulate monitors (XJ-SQN-RU-08 and -RU-14) are moving paper filter type and incorporate microcomputer-controlled step advance, and feed failure channel failure alarm. Sampling assembly fittings are provided which allow grab sampling of the monitored airstreams. The automatic control system maintains isokinetic flow based on a comparison of sample flow to HVAC duct flow. The HVAC duct flow input is performed manually for XJ-SQN-RU-08 and -RU-14 and automatically for XJ-SQN-RU-141 through -RU-144, and XJ-SQB-RU-145 and -RU-146.

A flow-integrating elapsed sample volume indicator is provided downstream of each particulate and/or iodine channel. It has a local digital readout and is resettable to zero.

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~~INSERT (C) TO SECTION 11.5.2.1.1.7.22. (Page 11.5-26)~~:

To be inserted on pg. 18.II.F-3 (A)

¶ Monitors are designed to meet a 90% efficiency 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 ~~PSAR~~ section 12.1.2.4. ~~and item II.B.2.~~ Monitors are designed to allow personnel to remove, replace, and transport sampling media without exceeding the criteria of General Design Criterion 19 ~~GDC19~~ of 5 rem whole-body and 75 rem to the extremities.

18.II.F.1.3 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 photon-only measurement with an upper range of 10^7 R/hr.

PVNGS Evaluation

As noted in table 11.5-1, in-containment area monitors XJ-SQA-RU-148 and XJ-SQB-RU-149 are provided to measure γ-photon activity with an upper range of 10^7 R/hr.

18.II.F.1.4 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 is provided as described in section 7.5.1.1.5, and the Appendix 6A response to Question 6A.14.

¶ Wide range containment pressure measurement ~~will be~~ is provided consisting of redundant pressure transmitters whose signals of containment pressure ~~will be~~ is continuously displayed within the control room. Continuous recording ~~will be~~ is provided for one channel over the entire range of pressure measurement. The transmitters ~~will be~~ are 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~~ is from -5 to 180 psig, three times the containment pressure.

The transmitters ~~will be~~ are 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.

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18.II.F.1.5 CONTAINMENT WATER LEVEL MONITOR

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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

Narrow-range water level instrumentation monitoring the containment ~~normal~~ radwaste sumps, and wide-range containment water level instrumentation are discussed in section 7.5.1.1.5.

by narrow range level instrumentation.

¶ Continuous indication of the containment radwaste sumps (containment normal sumps) water level is provided in the control room by narrow range level instrumentation. Each sump ~~will be~~ is 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~~ is 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.

¶ insert (B)

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(To be inserted on pg. 18.II.E-6)

(B)

~~(2) Wide Range Level Measurement~~

Continuous control room indication is provided for containment water level by wide range level instrumentation 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~~ are 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.

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18.II.F.1.6 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

A description of the containment hydrogen monitoring system is provided in section 6.2.5.2.2.2. The range and accuracy of the hydrogen analyzer is given in table 7.5-1.

The analyzers, one per each train, ~~will be~~ are 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~~ can 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.

A remote control panel mounted on the main control board provides control of each analyzer. Recording for one channel is also provided in the control room.

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Tests have verified that the hydrogen-oxygen recombination is not a catalytic surface effect associated with the heaters, but occurs due to the increased temperature of the process gases. As the phenomenon is not a catalytic effect, saturation of the unit is not predicted to occur. Results of testing a prototype electric hydrogen recombiner and production unit test results are given in reference 1. There is no difference between the hydrogen recombiner units to be installed in PVNGS and the unit for which the tests were conducted.

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6.2.5.2.2.2 Hydrogen Monitoring Subsystem. The hydrogen monitoring subsystem for each unit consists of two completely redundant trains. Each train consists of a hydrogen sensor,

9 an electronic subassembly and local and remote readout/alarms. The electronic subassemblies for trains A and B are housed separately in cabinets located in the auxiliary building.

A bottled nitrogen and hydrogen supply is used to calibrate the sensors at those intervals specified in section 16.3/4.

Hydrogen measurement is accomplished by using a thermal conductivity cell and a catalytic reactor. The sample gas first flows through the sample section of the cell, then passes through the catalytic converter where hydrogen in the sample is catalytically combined with free oxygen to form water vapor,

9 then passes through the reference section of the cell. The hydrogen content is indicated by the difference in thermal conductivity between the sample and reference sides of the cell. Oxygen, in an amount sufficient to combine hydrogen at the highest range of the analyzer, is added to the sample gas, prior to passing through the sample section of the cell. ¶ The range and accuracy of the hydrogen analyzer is given in table 7.5-1. INSERT (A)

¶ A single failure analysis is given in table 6.2.5-2.

¶ Refer to section 18.II.F.1.6 for TMI related information pertaining to "Containment Hydrogen Monitor"

A remote control panel mounted on the main control board provides control of each analyzer. Recording for one channel is also provided in the control room

for manual actuation of the containment combustible gas control system. Redundant analog instrument channels provide the required information.

Control room indications are provided to allow the operator to monitor and evaluate the operation of active system components during system operation, including periodic tests and the post-accident period. Table 7.5-1 lists parameters monitored in this system.

Control of the containment combustible gas control system is local and indication of system air flow and temperature is provided at the local panel.

C. Monitoring of Auxiliary Feedwater System

Refer to section 7.3.1.1.10.7. Information is provided in the control room to allow the operator to monitor and evaluate the operation of the active system components during system operation including periodic tests and the post-accident period. Table 7.5-1 lists parameters monitored in this system.

7.5.1.1.4 CEA Position Indication

Refer to CESSAR Section 7.5.1.1.4.

7.5.1.1.5 Post-Accident Monitoring

Refer to CESSAR Section 7.5.2.5 and Table 7.5-3. INSERT (C) 16

7.5.1.1.6 Automatic Bypass Indication on a System Level

A status monitoring panel in the control room displays the availability of the CESSAR ESFAS, the one-out-of-two ESFAS, all the ESF systems (including the NSSS ESF systems and the containment combustible gas control system), and the automatic

INSERT © TO SECTION 7.5.1.1.5 (Page 7.5-7) 14/30
(sheet 1 of 13)

In addition to the instrumentation described in CESSAR, also required as post accident monitoring instrumentation are extended range noble gas effluent radiation monitoring, containment high-range radiation monitoring, containment pressure monitoring, containment water level monitoring, and containment hydrogen monitoring.

¶ A detailed discussion on radiation monitoring is provided in section 11.5.

¶ A discussion on hydrogen monitoring is provided in section 6.2.5.2.2.2.

¶ Refer to section 18.II.F.1 for TMI related information pertaining to post-accident monitoring instrumentation.

Materials for fasteners of stainless steel parts (bolts or nuts) and pump wearing parts (wear rings or seals) are austenitic stainless steel or other ASTM specified material suitable for the water chemistry and/or radiation environment of the sampler.

11.5.2.1.1.7.2 Sampling Assembly

11.5.2.1.1.7.2.1 Each process or effluent channel includes a sampling assembly which consists of a sampler and the associated piping, fittings, and other components as required to transport the sample through the system. The sampling assembly is a closed sealed system and includes a sampling pump, valves, interconnecting piping, filters, fittings, flow and pressure transducers, and other local control and instrumentation elements as required. Samplers, with the exception of XJ-SQN-RU-141 and XJ-SQB-RU-145 particulate-iodine samplers, house radiation detection equipment and check source(s). Sampler piping and connections are welded except where maintenance considerations make flanged or Swagelok joints necessary. Sampler outlet piping connections are located to minimize cleaning requirements and background buildup due to the adherence of radioactive particles to the sampler walls. For liquid samplers, welding of pressure-containing components is performed in accordance with ANSI B31.1. For ESF monitors, welding of pressure-containing components is performed in accordance with AWS D1.1-1972 (with 1973 revisions). Welding of other equipment is performed in accordance with industry standards.

11.5.2.1.1.7.2.2 For liquid and process channels, the sampler is a lead-shielded steel chamber. For particulate and iodine channels, the sampler is a lead-shielded filter assembly. Four ~~1/2~~ shielding is furnished for all process and effluent detectors.

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14 Airborne particulate and iodine monitors and samplers, (XJ-SQN-RU-08, -RU-14, -RU-141, -RU-142, -RU-143, -RU-144, and XJ-SQB-RU-145 and -RU-146) sample isokinetically in accordance with the principles and methods of ANSI N13.1-1969, Guide to Sampling Airborne Radioactive Materials in Nuclear Facilities. The particulate and iodine sample flow is maintained constant over the normal expected range of filter paper and/or charcoal cartridge differential pressure by an automatic control system.
14 Local flow indication and high- and low-sample flow alarm signals are provided. These signals actuate local alarms and the channel failure alarm. Two particulate monitors
14 (XJ-SQN-RU-08 and -RU-14) are moving paper filter type and incorporate microcomputer-controlled step advance, and feed failure channel failure alarm. Sampling assembly fittings are provided which allow grab sampling of the monitored airstreams. The automatic control system maintains isokinetic flow based on a comparison of sample flow to HVAC duct flow.
4 The HVAC duct flow input is performed manually for XJ-SQN-RU-08 and -RU-14 and automatically for XJ-SQN-RU-141 through -RU-144, and XJ-SQB-RU-145 and -RU-146.

A flow-integrating elapsed sample volume indicator is provided downstream of each particulate and/or iodine channel. It has a local digital readout and is resettable to zero.

7 ~~Particulate collection efficiency is greater than 90% for 0.3m particulates. Volatile iodine adsorption efficiency is greater than 90%.~~

11.5.2.1.1.7.3 Detector Assembly. The detector assembly is a completely weatherproofed assembly, housing a detector, preamplifier, and radiation check source. The assembly is capable of withstanding the design pressure and temperature of the piping system of which it is a part, without leakage, collapse of the tube walls, or damage to the detector.

The detector assembly is incorporated in the sampler assembly

¶ Refer to section 18.II.F.1.2 for TMI related information pertaining to the particulate collection and iodine adsorption efficiency.

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shut at 4 to 6 psig to protect the monitor from significant
5 pressure.and temperature transients inside the containment.
Additional containment isolation valves (refer to section 6.2.4)
shut on CIAS when containment pressure reaches the containment
isolation pressure setpoint. Therefore the CB-E monitor is
designed to function properly subsequent to an event where
pressure is applied to the sampler piping.

INSERT D

11.5.2.1.4 Effluent Radiation Monitoring

12 11.5.2.1.4.1 <u>Condenser Vacuum Pump/Gland Seal Exhaust (CVSE, CVHR) (XJ-SQN-RU-141 and XJ-SQN-RU-142) Monitor</u>. The vacuum
pumps and gland seal condenser remove gases from the secondary
system. The exhaust is continuously monitored for gaseous
activity resulting from primary-to-secondary system leakage.
The exhaust is continuously and isokinetically sampled for
airborne radioactive particulates and iodines. Since the
exhaust is piped to its own separate exhaust from the build-
ing, no other airborne monitors are provided for the turbine
12 building. The monitor/sampler provides automatic initiation of
12 filtration of condenser vacuum pump/gland seal exhaust whenever
the monitor channel is in a HIGH-HIGH alarm condition.

Sampled air is pulled from the condenser vacuum pump/gland
steam exhaust piping at conditions of 125F and 100 percent
relative humidity. ~~Heaters are~~ Heat tracing is provided ~~to raise the tempera-~~ R1
~~ture of the air to 137F and an RH of 70 percent at the inlet to~~
to the sampler piping in order to prevent degradation of particu-
late and I-131 sampler filters due to excessive moisture. The
downstream gas channel detector is designed to withstand these
sample conditions in continuous service.

5 A low and a high range monitor is used to cover a range of
14 eleven decades with one decade of overlap. Particulate/iodine
cartridge samples exist in the low and high range monitor and
12 are removed for analysis. High range cartridge samplers are
shielded.

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INSERT (D) TO SECTION 11.5.2.1.4 (Page 11.5-40)

¶ The effluent radiation 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. The installed particulate and iodine sample collection media ~~Filter materials~~ used minimize absorption of noble gases. R1
Samples are preconditioned as necessary to assure accurate 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 for selected data and alarms.

¶ Refer to section II.F.1.1. for TMI related information pertaining to noble gas effluent monitors.

a HIGH-HIGH dose rate alarm. Redundancy and diversity are provided by the fuel building ventilation exhaust (XJ-SQB-RU-145) gas monitor. Refer to section 7.3 for a discussion of the safety function of the monitor. 12 2

11.5.2.1.5.5 Refueling Area Monitor, Channel "A" (RMAA) (XJ-SQA-RU-33). The monitor is located on a wall overlooking the refueling cavity where it monitors for a release of activity due to a fuel handling accident in the containment. 12 12

11.5.2.1.5.6 Power-Access Purge Area Monitors, Channels "A" (PAPA) and "B" (PAPB) (XJ-SQA-RU-37 and XJ-SQB-RU-38). The monitors are located between the containment power-access purge exhaust duct, and the refueling purge exhaust duct just outside the containment wall. During power operations, these channels monitor the duct for airborne radioactivity concentrations which could potentially result in an offsite dose exceeding 10CFR100 limits. These monitors perform the safety function of isolating the containment building purge supply and exhaust ducts (initiate CPIAS signals) on a HIGH-HIGH dose rate alarm. Refer to section 7.3 for a discussion of the safety functions of the monitors. 12 2 2 12

11.5.2.1.5.7

11.5.2.1.6 Inspection, Calibration and Maintenance

11.5.2.1.6.1 Maintenance. Outdoor sampling systems are housed in outdoor-type weatherproof enclosures. The enclosures are designed to permit performance of all control and routine maintenance and cleaning operations from the front or top of the enclosure. Lifting eyes or other devices are provided for hoisting the unit, to facilitate replacement if it is ever required. Interior wiring is run in conduit to terminal boards mounted in junction boxes.

Insert (E)

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INSERT (E) (SECTION ~~11.5.2.1.4.4~~ 11.5.2.1.5.7) (Page ~~11.5-41~~ 43)

~~11.5.2.1.4.4~~ 11.5.2.1.5.7 Main Steam Line (XJ-SQN-RU-139 and XJ-SQN-RU-140) Monitor.

One area monitor with a collimating lead shield is mounted adjacent to each main steam line in the Main Steam Support Structure approximately one foot upstream of the atmospheric dump valves. Refer To ~~FSAR~~ figure ~~12.3-2~~ 11.5-4. 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. 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 1 mr/hr to 10^7 mr/hr.

The detector is designed to operate in a post-accident environmental condition with a background of 10 R/hr

= blank space

Refer to section 18.II.F.1.1 for TMI related information pertaining to noble gas effluent monitors.

filters and disposable crud filters are related to their corresponding input activities provided in table 11.4-2.

RESPONSE:

The following assumptions were utilized in establishing SRS output activities in table 11.4-6:

1. Evaporator concentrates are solidified and stored in the high activity storage area for one month (i.e., 1 month decay) prior to shipment.

2. Spent resin beads are stored for 6 months prior to solidification. Solidified resin is stored in the high activity storage area for 1 month (i.e., 1 month decay) prior to shipment.

3. Cartridge filters are solidified and stored in the high activity storage area for one month (i.e., 1 month decay) prior to shipment.

4. Disposable crud filters are stored for one month (i.e., 1 month decay in the high activity storage area prior to shipment.

QUESTION 11A.12 (NRC No. 460.18) (11.5)

We have reviewed your submittal dated April 6, 1981, relating to TMI Action Plans II.F.1, Attachments 1 and 2, and III.D.1.1 of NUREG 0737. We find your information scant and very inadequate. Please provide the information on these action items as required by NUREG-0737. For guidance, you may refer to submittals on these action plans for other PWRs such as San Onofre, Units 2 and 3, and Summer Nuclear Station, which have been found acceptable by the staff.

RESPONSE: ~~Amendment 1 to the PVNGS Lessons Learned Implementation Report (LLIR) was submitted August 3, 1981. It contains~~ an expanded discussion of noble gas monitoring and effluent sampling per Attachments 1 and 2 to NUREG-0737, Item II.F.1, is provided in ~~section 11.5 as noted in~~ sections 18.II.F.1.1 and 18.II.F.1.2.

Items II.F.1, III.D.1.1

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Section 12.1.3.6

~~The PVNGS LLIR~~ also addresses leak reduction design measures per Item III.D.1.1 of NUREG-0737. Measurement and testing of covered systems will not take place until the startup of these systems. Accordingly, expansion of the LLIR for operational leak reduction testing can not be provided until after startup.

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Post-Accident Sampling; II.F.1 High Range In-Containment Radiation Monitors; and III.D.3.3 Post-Accident Iodine Sampling and Analysis.

RESPONSE: The response is provided in ~~the PVNGS LLIR.~~ Sections 18.II.B.2, 18.II.B.3, 18.II.F.1 and 18.III.D.3.3. ~~Additional information is provided in sections 12.1.2.4 and 12.3.1.3 for Items II.B.2 and II.B.3, in section 11.5 and figure 12.3-4 for II.F.1, and in sections 9.3.2.2.2 and 11.5 for III.D.3.3.~~

QUESTION 12A.5 (NRC Question 471.4) (12.1)

Paragraph B, of Section 12.1.2.1.2, stating that "as minimum, shielding is designed to reduce gamma dose rates from sources external to a radioactive compartment to levels comparable to dose rates resulting from equipment within that compartment," is not clear.

It appears that this could refer to shielding between two adjacent compartments in which radioactive equipment is located. If this is the case, then shielding should be designed to reduce radiation from the operating equipment in one compartment to levels below that which is expected in the adjacent compartment from the shutdown equipment to be maintained or repaired. Please clarify. 5

RESPONSE: Paragraph B of section 12.1.2.1.2 has been revised to provide the requested clarification.

QUESTION 12A.6 (NRC Question 471.5) (12.1)

In accordance with Section C.2.e, "Crud Control," of Regulatory Guide 8.8, it is our position that consideration should be given to the selection of corrosion resistant, low cobalt content alloys to reduce the concentrations of radioactive corrosion product buildup in systems.

Section 12.1.2.3, "Equipment General Design Considerations for ALARA," of your FSAR, should be revised to reflect your design considerations for selection of low cobalt alloys.

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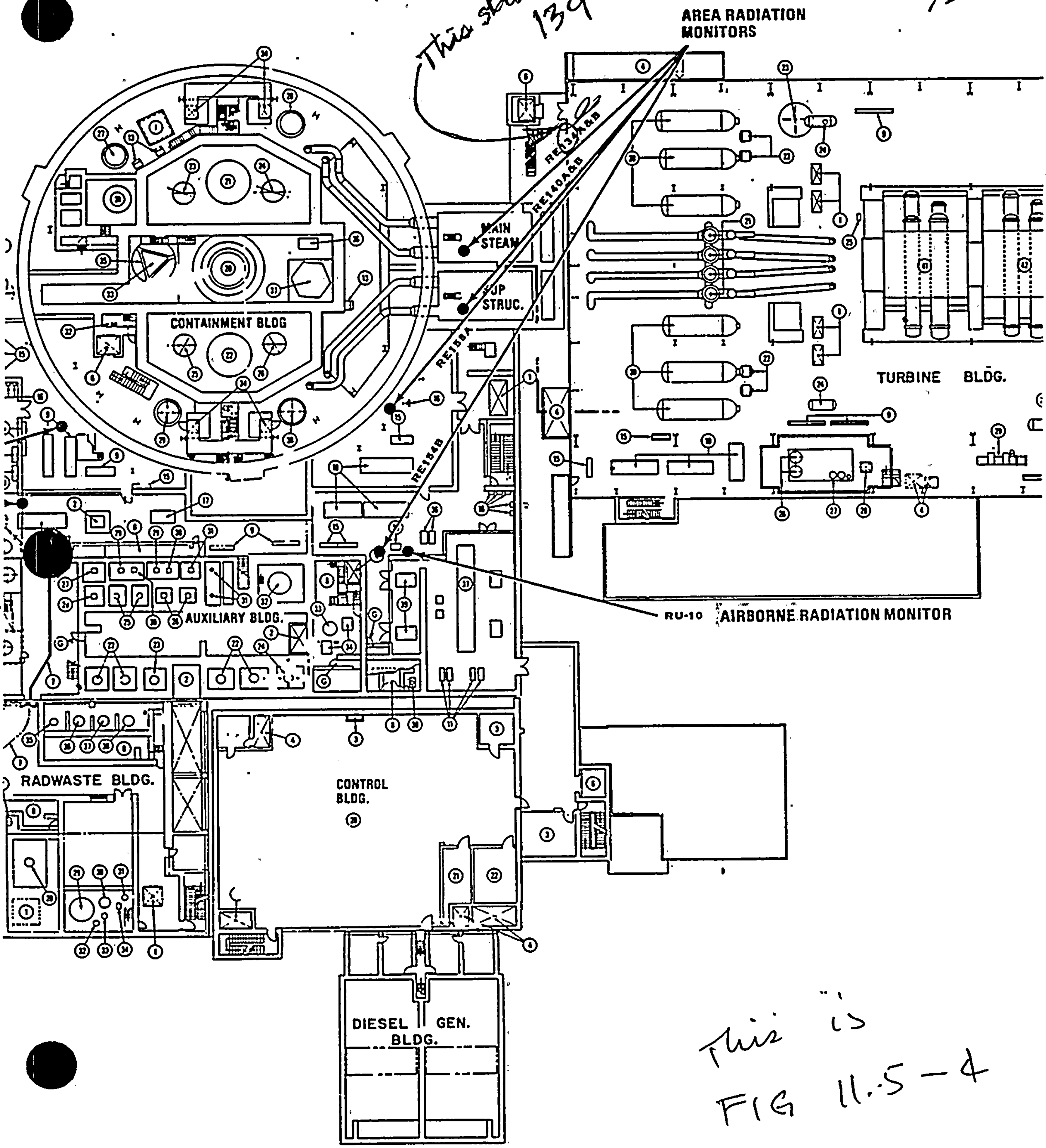
This should be 139



This is FIG 11.5-4

Amend 1

18.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:

1.67×10^5 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 = 29,900 ~~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~~ See FSAR 6.4.4.3(E)

(i) automatic isolation capability-damper closing time, damper leakage and area:

closing time = ~~15~~ 35 sec.

leakage = zero leakage (by bubble test)

area = 12.~~8~~2 sq ft (largest damper)

(j) chlorine detectors or toxic gas (local or remote):

insert A ~~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

Insert A

There is no onsite storage of liquid or gaseous chlorine at PVNGS. Chlorine is maintained as sodium hypochlorite in a liquid form. Due to the absense of a chlorine gas source, no chlorine detectors are present in the control room HVAC system.

Smoke detectors are provided in the outside air intake plenum to alert the control room operator to manually initiate the smoke removal mode.

(n) control-room personnel capacity (normal and emergency):

Refer to section 6.4.4.3.C for the personnel capacity of the control room.

~~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:

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 (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~~. Refer to the discussion provided in item 2j.

(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~~. Refer to Technical Specification 3/4.7.7, "Control Room Essential Filtration System".

J. Design Basis Ten

The control room essential HVAC system shall be designed to remain functional during and after a safe shutdown earthquake (SSE).

Air ducts and their supports shall be Seismic Category I.

The control room normal HVAC system is described in section 9.4.1.

Protection of the habitability systems in the control room from wind and tornado effects is discussed in section 3.3. Flood design is discussed in section 3.4. Missile protection is discussed in section 3.5. Protection against dynamic effects associated with the postulated rupture of piping is discussed in section 3.6. Environmental design is discussed in section 3.11. The fire protection system is discussed in section 9.5.1.

Codes and standards applicable to the control room emergency ventilation system are listed in table 3.2-1. The system is consistent with the recommendations of Air Moving and Conditioning Association (AMCA) standards and NRC Regulatory Guide 1.52, except as noted in section 1.8.

6.4.2 SYSTEM DESIGN

6.4.2.1 Definition of the Control Room Envelope

The areas, equipment, and materials to which the control room operator could require access during an emergency are shown in figure 7.5-1. Those spaces requiring continuous or frequent operator occupancy are also shown in figure 7.5-1. A layout drawing and a description of shielding required to maintain habitability of the control room during the course of postulated accidents is provided in section 12.3.

¶ Refer to section 18.III.D.3.4 for TMI related information pertaining to "Control Room Habitability Requirements".
INSERT (A)

INSERT (A) TO SECTION 6.4.2.1 (Page 6.4-3):

Functional areas included on the Control Building's elevation 140'0" are: the Satellite Technical Support Center (TSC), the control room, the kitchen, and the sanitary facilities.

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The closest distance between the Containment and the air intakes is approximately 150 feet.

6.4.2.2 Ventilation System Design

6.4.2.2.1 General Description

Section 9.4.1 contains an overall description of the control room HVAC system. The system is shown schematically in figures 9.4-1 and 9.4-2. Figure 6.4-1 shows the plant layout, including the location of potential radiological release points with respect to the control room air intakes. Elevation and plan drawings with descriptions providing building dimensions and locations are located in section 1.2. Potential sources of toxic gas releases are discussed in section 2.2.3.

The volume of the habitability zone served by the HVAC system in the emergency mode or the isolation mode is approximately 1.6×10^5 cubic feet.

Enviromental design criteria for the air purification system are based on the most limiting conditions resulting from any of the postulated design basis accidents (DBA) and on their duration in accordance with Regulatory Guide 1.52, as discussed in section 1.8.

Two identical, physically separted high efficiency filtration trains with charcoal adsorbers are provided to process intake air flow and recirculated air flow in the control room. Components are listed in table 6.4-1. Section 1.8 presents the system design conformance to each position in Regulatory Guide 1.52. The seismic classifications of components, instrumentation, and ducting are given in table 3.2-1.

¶ Refer to section 18.III.D.3.4 for TMI related information pertaining to "Control Room Habitability Requirements".

6.4.2.2.2 Component Description

The essential air handling unit contains a fan, a prefilter, a HEPA prefilter, an activated charcoal filter, a HEPA afterfilter, and a cooling coil. Pneumatic-operated dampers are provided for system isolation purposes.

under the environmental conditions associated with the postulated DBA.

Refer to section 18.III.D.3.4 for TMI related information pertaining to "Control Room Habitability Requirements."

G. Ductwork

The system ductwork and dampers are Seismic Category I. Ductwork is redundant where required to provide functional support to active components in meeting the single active failure criteria. Leaktight ductwork and isolation dampers are provided where required to isolate the system from unfiltered outside air.

In general conformance with Position C.4 of Regulatory Guide 1.52, accessibility and adequate working space for maintenance and testing operations are provided in the design and layout of the air purification system equipment.

I. Control Access Doors

To minimize inleakage, the control access doors are equipped with self-closing devices that shut the doors automatically following the passage of personnel. Alarms are also provided to annunciate if any of the doors are open after a changeover to emergency operation. Two sets of doors with a corridor between, acting as an air lock, are provided at each of the two entrances to the control room and associated spaces.

J. Isolation Dampers

System isolation dampers are capable of automatically closing ~~in 15~~ within 35 seconds after receipt of an actuation signal, as verified by testing. The isolation dampers, ~~each with an~~ maximum area of approximately 12.2 sq. ft. per Damper, are tested as bubble-tight dampers for zero leakage, as part of the manufacturer's test program.

Refer to section 18.III.D.3.4 6.4-7 for TMI related information pertaining to "Control Room Habitability Requirements".

atmospheric pressure during emergency operation. The control room essential ventilation system maintains the same temperature and humidity conditions when operating in the isolation mode.

C. Safety Evaluation Three

The control room ventilation system is capable of removing sensible and latent heat loads of 981,552 Btu/h and 13,330 Btu/h, respectively, which includes consideration of equipment heat loads and minimum personnel occupancy requirements.

The transfer to essential or isolation operation mode does not create a hazard for CO_2 buildup. In case of emergency operation, there is a supply of outside air [a maximum] of 1000 ft^3/min and the long term equilibrium for CO_2 will remain below 1 part per thousand for up to 50-person occupancy. In case of isolation mode operation, where the control room is sealed, the critical level of 3% would be reached at the following times for the various occupancies:

6 persons	26.2 days
12 persons	13.1 days
30 persons	5.2 days

D. Safety Evaluation Four

Food, water, medical supplies [(including a potassium iodide drug supply),] and sanitary facilities are provided for a minimum occupancy of 6 persons for 7 days. Storage locations provided ensure that the above supplies will not be contaminated as a result of postulated accidents.

The supply of food and water is sufficient for a prolonged occupancy since outside supplies can be provided within the 7-day interval.

(including a potassium iodide drug supply),

¶ Refer to section 18.6.4-19 III.D.3.4 for TMI related information pertaining to "Control Room Habitability Requirements".

I. Position C.4.b

See Regulatory Guide 1.52 paragraph K response in section 1.8.

J. Position C.4.c

The control building systems do not have permanently-installed aerosol-injection ports upstream of the fan. Instead of the aerosol-injection ports, access panels were provided upstream of the fan.

K. Position C.5.d

See Regulatory Guide 1.52 paragraph M response in section 1.8.

L. Position C.6.a

See Regulatory Guide 1.52 paragraphs N and O response in section 1.8. 4

QUESTION 6A.7 (NRC Question 450.3) (6.4)

In your description of the control room's protective features, provide the time interval between the time the chlorine concentration exceeds 5 ppm at the isolation dampers and the time the dampers are completely closed.

RESPONSE: ~~The response is given in amended section 6.4.2.2.2.K.~~ INSERT (B)

QUESTION 6A.8 (NRC Question 450.4) (6.4)

List the areas, equipment and materials to which the control room operator has access during emergency operation, i.e., during the time the control room is serviced by the emergency ventilation system.

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INSERT (B) TO THE RESPONSE TO QUESTION 6A.7 (Pg. 6A-5)

Refer to the response to NUREG-0737, ~~Item~~ section 18. III.D.3.4.(2).(j).

sump chemistry control, in kg or moles, and the location of the TSP baskets. Calculate the post-injection sump pH. Preliminary staff calculations indicate that the long-term pH should be at least 8.0 to meet 10 CFR 100 dose guidelines for the DBA
6 LOCA.

RESPONSE: The response will be provided on the CESSAR docket. Also refer to the response to QUESTION 6A.30 (NRC Question 281.5).

7 <u>Question 6A.29</u> (NRC Question 450.18) (6.4)

Provide the following information required for the control room habitability evaluation:

(1) control room shielding including radiation streaming from penetrations, doors, ducts, stairways, etc.:

(2) Self-contained breathing apparatus availability (number)

(3) bottled air supply (hours supply)

(4) control room personnel capacity (normal and emergency)

(5) potassium iodide drug supply

(6) control room emergency filtration system including
6 the capability to maintain the control room pressurization at 1/8-inch water gauge, verification of isolation by test signals and damper closure times, and filter testing requirements.

<u>RESPONSE</u>:

1) The required information is provided in sections 6.4.2.5 ~~and LLIR Section II.B.2.~~ and 12.3.2.2.7.

2) The required information is provided in paragraph M of section 6.4.2.2.2.

3) The required information is provided in paragraph M of section 6.4.2.2.2.

Normal HVAC System--Control, Computer and Associated Rooms

The control room complex is located on elevation 140' 0". The normal HVAC system provided for the control and computer room includes cooling by an air washer (evaporative) for the outside air supply which is common for the total control building and by a recirculating air conditioning system with cooling coils served by the normal chilled water system described in section 9.2.

Heating is provided by the use of electric zone heaters located in the supply air ducts.

P Refer to section 18.III.D.3.4 for TMI related information pertaining to "Control Room Habitability Requirements".

9.4.1.2.1 Design Bases

9.4.1.2.1.1 Safety Design Bases. The normal HVAC system provided for the control and computer room has no safety design bases. Protection of the operator from radioactivity and poisonous gases is described in section 6.4. The isolation is treated as a part of the essential control room HVAC system.

9.4.1.2.1.2 Power Generation Design Bases. The normal HVAC system provided for the control and computer room complex has one power generation design basis:

A. Power Generation Design Basis One

The normal HVAC system shall supply conditioned air to the control and computer room during normal plant operating conditions to provide personnel comfort and to maintain a suitable operating environment for equipment.

9.4.1.2.1.3 Codes and Standards. The normal HVAC system provided for the control and computer room is designed in accordance with codes and standards set forth in table 3.2-1.