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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

NOV 2 0 1978

Docket Nos. STN 50-592
and STN 50-593

APPLICANT: Arizona Public Service Company

FACILITY: Palo Verde Nuclear Generating Station, Units 4 and 5

SUBJECT: SUMMARY OF MEETING HELD ON NOVEMBER 14, 1978 REGARDING THE SAFETY REVIEW OF PALO VERDE, UNITS 4 AND 5

A meeting was held between NRR staff members and representatives of the Arizona Public Service Company, Combustion Engineering Inc., and Bechtel Power Corporation to discuss the safety review of the Palo Verde Nuclear Generating Station, Units 4 and 5. The meeting agenda and list of attendees are attached as Enclosures 1 and 2.

Summary of Meeting

Enclosure 3 provides a compilation of some of the questions discussed during the meeting. The questions in Enclosure 3 which are under the headings "Conduct of Operations" and "Radiation Protection" did not require any discussion during the meeting. In addition to the responses documented in Enclosure 3, the following points were made:

1. Regarding the commitment to 10 CFR 50.55a, APS assured us that the applicable code addenda to be applied for Units 4 and 5 components within the reactor coolant system pressure boundary would be based on the purchase order date for such components and not related to addenda used for Units 1, 2 and 3. APS will make a clarification to this effect in the PSAR.

2. With res[illegible]t to Quality Assurance question #1 in Enclosure 3, we requeste[illegible] APS maintain the revised commitment to ANSI N45.2.13 (Draf[illegible]v. 4, April '74) which was included in the question response and su[illegible]ntly scratched.

3. With res[illegible] to Fire Protection question A in Enclosure 3, we informed the applicant that Mr. Vassallo's letter of August 29, 1977 required the designation of the upper level offsite management position which has responsibility for the fire protection program. APS said they would review this requirement.

4. Concerning Instrumentation and Control Systems question #1, we requested that APS advise us concerning the extent to which safety-related instruments do not have setpoint adjustment mechanism securing devices.

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5. Concerning Instrumentation and Control Systems question #3, we informed APS that preliminary design information should be included in PSAR section 7.6. A preliminary logic diagram was discussed as a possible means to provide this information.

6. Concerning Instrumentation and Control Systems question #4, we informed APS that we would review their response and that if additional clarification was required, they would be so informed. APS said they would review the tables in question and determine if clarifying footnotes would be helpful.

7. Regarding Reactor Systems question #1, the applicant referenced Appendix A of CESSAR. We informed CE that, as a minimum, a commitment to the basic criteria for the overpressure mitigation system, which are included in the question to Palo Verde, must be included in an acceptable CP level review. Details of system design and implementation of the basic criteria can be resolved during the CESSAR FSAR review.

8. Regarding Reactor Systems question #2, we stated that a commitment to perform tests or document results of occurances at operating plants was required at the CP stage. The tests are required to show adequate mixing and cooldown under natural circulation. After the meeting it was noted that part b of the question should read "cooldown to RHR initiation in less than 36 hours" rather than "cooldown to cold shutdown in less than 36 hours". Also after the meeting, we determined that additional discussion on this subject with APS and CE would be beneficial in clarifying how we are reviewing the positions of RSB 5-1.

9. Concerning Reactor Systems question #3, we requested that a schedule for conducting the reactor coolant pump test described in CENPD-201 be provided. CE said the test would be done in calendar year 1979 and that the schedule would be documented.

10. Concerning Reactor Systems question #6, we informed CE that we believed other applicants with CE plants had given satisfactory responses to this issue.

11. The containment isolation system was discussed at length. Several inconsistencies and required clarifications in Table 6.2-16 and Figure 6.2-22 of the Palo Verde PSAR were noted. Appropriate revisions to the PSAR will be made. Also discussed was the preliminary plans for the testing required by Appendix J of 10 CFR 50. We provided APS with a copy of our proposed changes to Appendix J and discussed several places where recent OL applicants have had difficulty meeting our Appendix J requirements.

12. APS gave a presentation on their proposed off-site power system emphasizing their undervoltage protection scheme, the use of load shedding while power is being supplied by the diesels, and the use of the sequencer for off-site power loading. A copy of the slides used by APS and their responses to the applicable Power Systems Branch questions are attached as Enclosure 4.

Concerning the undervoltage protection provided for degraded grid conditions, we stated that although we had not completed our review, the proposed design appeared to be acceptable. Concerning the use of load shedding while on emergency power, we said that the proposed design appeared to meet the original intent of the staff position in that load shedding was prevented during the sequencing of loads. APS was requested to provide a description in the PSAR of the logic for this system including the 60 second timer that accomplishes the above. With respect to the use of the sequencer with offsite power, we decided after the meeting to maintain our position. If APS wishes to pursue the matter further with the staff, an appeal should be requested. If an appeal is conducted, APS should be prepared to discuss the status of procurement of associated equipment and details of the proposed sequencer along with the rationale for their design.

13. Concerning the response provided to a Power Systems Branch question during our October 19 meeting (and regarding Regulatory Guide 1.63) we noted: (a) that testing of molded case breakers would be required and (b) Revision 2 of the guide should be addressed.

14. Concerning the CE response to Regulatory Guide 1.99, we stated that a more positive commitment to the guide was necessary. CE stated that they had an approved topical report which addresses the same issues as the guide. It was suggested that CE include a reference to this topical report in their revised response.

15. Concerning the CE response to Regulatory Guides 1.124 and 1.121, it was determined that additional discussions between the staff and would be necessary.

16. Site drawings were reviewed to clarify exclusion area and site boundary distances.

17. APS stated that, contrary to their letter of 9/25/78 on the unit 1, 2, and 3 docket, the smoke detectors were not eliminated from the control room air intake.

18. The subject of PSAR Tables 7.1-1 and 8.1-1 was broached. These tables provide the interface acceptance criteria for instrumentation, controls, and electric power systems. There are some differences between the PSAR tables and similar ones provided in the CESSAR SER. APS reminded us that they requested we review this matter back in February, 1977.

Robert Stright

Robert Stright, Project Manager
Light Water Reactors Branch
Division of Project Management

Enclosures:
1. Agenda
2. Attendees
3. Questions and Responses
4. Off-Site Power Presentation

cc w/enclosures:
See next page

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ENCLOSURE 1

NRC - APS - CE

MEETING AGENDA

11/14/78 P-110

Time	Subject
9:00	Reactor Systems - Glen Kelly
10:00	Reg. Guide 1.99 - Dave Sellers
10:15	CESSAR/commitment to RG 1.124 & 1.121 - Jim Brammer
10:30	Instrumentation & Control Systems - Bob Stevens, C. Miller
1:00	Offsite Power Systems - Om Chopra, Faust Rosa
2:00	QA - Jack Spraul
2:30	Smoke Detectors - Jerry Wermiel Excl. Area Distances - Al Brauner
3:00	Containment Isolation System - Stuart Brown

ENCLOSURE 2

LIST OF ATTENDEES

NRC

R. Stright
R. Kirkwood
I. Villalva
J. Spraul
J. Wermiel
R. Stevens
C. Miller
G. Kelly
A. Brauner
A. Bennert
O. Chopra
F. Rosa
S. Brown
J. Shapaker
J. Pulsipher
C. Sellers
H. Brammer

Arizona Public Service Co.

D. Karner
J. Allen

Bechtel Power Corp.

D. Keith
W. Bingham

Combustion Engineering

J. Goldberg
C. Ferguson

ENCLOSURE 3

PALO VERDE NUCLEAR GENERATING STATION

UNITS 4 AND 5

DOCKET NOS. STN 50-592 AND STN 50-593

REQUEST FOR ADDITIONAL INFORMATION

Reactor Systems

Question 1

The staff requires that a permanent overpressure mitigation system (OMS) be provided prior to first fuel cycle operation to prevent reactor vessel pressures in excess of those allowed in Appendix G. Specific criteria for system performance to be addressed are:

a. Operator Action: No credit can be taken for operator action for 10 minutes after the operator is aware of a transient.

b. Single Failure: The system must be designed to relieve the pressure transient given a single failure in addition to the failure that initiated the pressure transient.

c. Testability: The system must be testable on a periodic basis consistent with the system's employment.

d. Seismic and IEEE 279 Criteria: Ideally, the system should meet seismic Category I and IEEE 279 criteria. The basic objective is that the system should not be vulnerable to a common failure that would both initiate a pressure transient and disable the overpressure mitigating system. Such events as loss of instrument air and loss of offsite power must be considered. The staff requires that the OMS be capable of withstanding an operational basis earthquake.

In demonstrating that the mitigation system meets these criteria, the applicant should include the following information in his submittal:

a. Identify and justify the most limiting pressure transients caused by mass input and heat input.

b. Show that overpressure protection is provided (do not violate Appendix 6 limits) over the range of conditions applicable to shutdown/heatup operation.

c. Identify and justify that the equipment will meet pertinent parameters assumed in the analyses (e.g., valve opening times, signal delay, valve capacity).

d. Provide a description of the system including relevant P&I drawings and electrical schematics.

e. Discuss how the system meets the criteria.

f. Discuss all administrative controls required to implement the protection [illegible]tem.

g. Assure that [illegible]ign pressures do not exceed 110% of design press[illegible]

Response:

A discussion of the consequences of inadvertent overpressurization resulting from a malfunction or operator error when the reactor coolant system is water solid during startup or shutdown is within the scope of the CESSAR. Refer to CESSAR FSAR Appendix A for resolution of this matter.

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REQUEST FOR ADDITIONAL INFORMATION

Reactor Systems

Question 2

The Regulatory Requirements Review Committee, in a memorandum from E. Case, Committee Chairman, to L. Gossick, Executive Director for Operations (dated February 16, 1978), has approved a new staff position (BTP BSB 5-1) for the Residual Heat Removal System (RHR). The technical requirements for your plant are described below. Please respond to these requirements in sufficient detail to enable the staff to review your compliance in an expeditious fashion.

a. Provide safety-grade steam generator pump valves, operators, air and power supplies which meet the single failure criterion.

b. Provide the capability to cool down to cold shutdown in less than 36 hours assuming the most limiting single failure and loss of offsite power, or show that manual actions inside or outside containment or return to hot standby until the manual actions or maintenance can be performed to correct the failure provides an acceptable alternative.

c. Provide the capability to depressurize the reactor coolant system with only safety-grade systems assuming a single failure and loss of offsite power, or show that manual actions inside or outside containment or remaining at hot standby until manual actions or repairs are complete provides an acceptable alternative.

d. Provide the capability for borating with only safety-grade systems assuming a single failure and loss of offsite power, or show that manual actions inside or outside containment or remaining at hot standby until manual action or repairs are completed provides an acceptable alternative.

e. Commit to conduct, or reference appropriate prototypical tests to study both the mixing of the added borated water and the cooldown under natural circulation conditions with and without a single failure of a stear generator atmospheric dump valve. These tests and analyses will be used to obtain information on cooldown times and the corresponding AFW requirements.

f. Commit to providing specific procedures for cooling down using natural circulation and submit a summary of this procedures.

g. Provide or require a seismic Category I AFW supply for at least 4 hours at hot shutdown plus cooldown to the DHR system cut-in based on the longest time (for only onsite or offsite power and assuming the worst single failure), or show that an adequate alternate seismic Category I source will be available.

Response

Refer to CESSAR FSAR Appendix A for a discussion of the CESSAR design.

a. The Palo Verde design will include safety grade steam generator dump values, operators, air and power supplies.

b. The Palo Verde design has the capability to cool down to cold shutdown in less than 36 hours assuming the loss of offsite and the loss of one atmospheric dump valve.

c. The Palo Verde design has the capability to depressurize the reactor coolant system with safety grade systems and a loss of offsite power. However, single failure of a valve in the charging line or spray line would require manual operation of that valve.

d. The Palo Verde design has the capability for borating with safety grade systems and a loss of offsite power. However, single failure of a valve in the charging line would require manual operation of that valve.

e. Performance of tests to study both the mixing of the added borated water and the cooldown under natural circulation conditions with and without a single failure of a steam generator atmospheric dump valve will be discussed in the PVNGS FSAR.

f. Procedures for cooling down using natural circulation will be provided in the FSAR.

g. The seismic Category I Condensate Storage Tank provides for at least 4 hours at hot standby plus cooldown to 350^{o}F based on a loss of offsite power plus a single failure.

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Reactor Systems

Question 3

As part of our concern over adequate cooling of the reactor coolant pumps, and the ability of these pumps to withstand a loss of component cooling flow, we require the following:

a. Provide a reference for the RCP test results, associated with CENPD-201, designed to provide assurance that the RCPs can continue to function without excessive seal leakage or damage to the thrust bearings that could affect the pump coastdown characteristics in the event of loss of cooling flow.

b. Provide administrative controls that are defined in the plant technical specifications and which will require that: (1) the affected reactor coolant pumps be shutdown 30 minutes after loss of component cooling water, (2) the affected reactor coolant pumps will not be restarted until component cooling water is restored and pump thermal conditions are normal, and (3) prior to shutdown of the last reactor coolant pump, a sufficient amount of boron will be introduced into the reactor coolant system to facilitate cooldown to residual heat removal system operating conditions.

Response:

a. CENPD-201 commited Combustion Engineering to performing a protypical test to substantiate the analysis of the pump performance when component cooling water is lost. This test has not, as yet, been completed. When the results are available, they will be provided as an appendix to CENPD-201. A reference to CENPD-201 will be provided in the PVNGS 1, 2, & 3 PSAR.

b. The plant technical specifications, as provided in CESSAR PSAR, do not contain these requirements on the reactor coolant pump's operation. As part of the work being done on these pumps, a test is being performed by Combustion Engineering to demonstrate the availability of a full 30 minutes for operator action (see above). Additional technical specifications will be considered, if necessary, on the CESSAR docket after completion of this test. The final resolution of this matter will be addressed in the Palo Verde FSAR.

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REQUEST FOR ADDITIONAL INFORMATION

Reactor Systems

Question 4

Mitigation of a passive failure during long-term cooling following a LOCA requires additional information. Provide the following:

a. Identify and justify the maximum leak rate assumed in your passive failure analysis.

b. Justify the leak in an ECCS' pump room of 50 fpm can be mitigated such that no vital equipment is flooded given operator action no sooner than 30 minutes following the first leak detection alarm in the control room. Identify the alarm (and time of its annunciation) which alerts the operator to the leak. The alarm considered must be in the control room and be IEEE-279 except for single failure requirements.

c. Justify that the operator can identify and isolate the faulted line given the information available to him in the control room.

Response #4

a. The maximum expected seal leakage rates are given in PVNGS 1, 2 & 3 PSAR Table 15.4-7A. The radioactive drain system for the auxiliary building ESF pumprooms is conservatively sized for a 50 gpm leakage rate.

b. As the room drains are sized for a 50 gpm leakage rate, no operator action is required to prevent flooding by a leak of this magnitude sooner than 30 minutes following the first leak detection alarm in the control room. A leak during long-term cooling following a LOCA would be mitigated by virtue of multiple level alarms designed to annunciate in the control room upon one or more of the following level indications:

1. The drain in each ESF pump room is provided with a high level alarm with control room annunciation.
2. Each ESF pump is also provided with a high-high level switch with control room alarm. This switch is located on a compartment wall 9 inches above the floor elevation.
3. Each ESF sump is provided with a low-low and a high-high level switch which alarms in the control room.
4. Similar high and high-high level switches with control room alarm are provided in other ESF equipment areas including the shutdown cooling heat exchanger rooms and the pipe chase area rooms.

These level switches and alarms are designed to provide highly reliable operation. Multiple levels of alarming assure detection of leakage in the event of an alarm or switch malfunction.

c. Since multiple level switches and alarms monitoring fluid levels in each train-related compartment and sump are provided, the control room operator is able to identify the faulted train and isolate the appropriate ESF train using the information available to him in the control room.

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Reactor Systems

Question 5

Provide assurance that no manually operated valves in the ECCS, if incorrectly positioned, can significantly degrade ECCS performance. If any such valves exist, the staff requires that position indication be provided for them in the control room. Commit to provide this indication for any valves meeting this criterion.

Response

Manual valves which could jeopardize the operation of the ECCS are mechanically locked in the correct position. These valves are not expected to be operated more than once per year. Based on the guidance of Regulatory Guide 1-47, control room indication of these valves is, therefore, not required.

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Reactor Systems

Question 6

A steam line break causes cooldown of the primary system, with shrinkage of RCS inventory and depletion of pressurizer fluid. Subsequent to plant trip, ECCS actuation, and main steam system isolation the RCS inventory increases and expands, refilling the pressurizer. Without operator action (which is usually the termination of ECCS injection at 10 minutes) expansion at low temperature could repressurize the reactor to an unacceptable pressure-temperature regime compromising reactor vessel integrity. Analyses are required to show that following a break in a main steam line and not assuming operator action before 10 minutes (a) no additional fuel failures result from the accident and (b) the pressures following the initiation of the break, considering the changes in coolant and material temperature, will not compromise the integrity of the reactor coolant pressure boundary. If this issue is totally within the scope of CESSAR, commit to adopting the resolution between the staff and Combustion Engineering.

Response

This issue is totally within the scope of CESSAR. Refer to CESSAR FSAR Appendix A for resolution of this matter.

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REQUEST FOR ADDITIONAL INFORMATION

Instrument and Control Systems

Question 1

Qualification Review Item (E-17) - R. G. 1.105 - Instrument Setpoints.

The response to qualification review item E-17 does not meet our requirements for conforming to the recommendations of Regulatory Guide 1.105. It is the staff's position that all portions of R. G. 1.105 be applied to all instrumentation systems important to safety.

We request that you provide a commitment that all instrumentation in systems important to safety will conform to the recommendations of R. G. 1.105, and that each exception will be identified and justified.

Response 1

Refer to CESSAR FSAR Appendix A for instruments in the CE scope of supply. All other instruments in systems important to safety will meet the requirements of Regulatory Guide 1.105 except that securing devices will only be used when required to meet drift specifications and when available.

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REQUEST FOR ADDITIONAL INFORMATION

Instrument and Control

Question 2

Qualification Review Item (E-27) - Environmental Control and Qualification Outside Containment.

This response does not satisfy our concerns as stated in qualification review item E-27. Specific problem areas of the response are:

1. Only a few plant areas are identified,

2. Only the temperature environment is mentioned,

3. Portable temperature recorders are used to ensure that limits established by the environmental qualification program for Class 1E equipment are not exceeded but they are placed in the area only after alarms indicate they are needed,

4. Abnormal environmental conditions for the control room are not defined,

5. It is not described how operating personnel are to detect abnormal normal environmental conditions in the control room, and

6. No justification is given to describe why it might be necessary to shutdown all environmental control equipment in a safety related area.

It is the Staff's position that you should resolve the concerns of qualification review item E-27. Specifically:

A. Provide a commitment that all safety related equipment will be qualified to the extreme environmental conditions (as specified in Paragraph 3(7) of IEEE 279-1971) that can occur during plant shutdown or hot standby, or

B. Specify that the environmental control systems in safety-related plant areas will operate continuously to maintain the environmental conditions within the qualified limits of the safety-related equipment. As a minimum, you are requested to concentrate on the problem areas previously described and:

   1. Identify appropriate plant areas and describe environmental control systems,

2. Identify the environmental conditions (temperature, humidity, radiation, etc.) that will be experienced in each plant area,

3. Identify environmental monitoring equipment for each environmental condition per the concerns of qualification review item E-27.

4. Describe administrative procedures that will ensure environmental control systems continue to be operable during plant shutdown or hot standby,

5. Justify the use of portable temperature recorders and describe how their use meets the requirements of qualification review item E-27 as it pertains to environmental monitoring equipment (including control room alarm),

6. Describe the method of analysis to be provided according to qualification review item E-27, and

C. Identify and justify each exception to the Staff's requirements A and B above.

Environmental Control & Qualification Outside Containment

Response #2

A. The following plant areas contain safety related equipment:

1. Auxiliary Building
   a. HPSI pump rooms
   b. LPSI pump rooms
   c. CSS pump rooms
   d. ECW pump rooms
   e. ESF electrical penetration areas
   f. CEDM control cabinet room
2. Main .'eam Support Structure
   a. Motor driven auxiliary feed pump room
   b. Turbine driven auxiliary feed pump room
   c. Area above elevation 100'
3. Diesel Generator Building
   a. DG rooms
   b. DG control area
4. Control Building
   a. Control room
   b. ESF switchgear rooms
   c. ESF equipment rooms
   d. Essential battery rooms
   e. Remote shutdown panel rea
5. Fuel Building Essential Exhaust Filter
6. Containment

   The safety related equipment inside containment (Item 6) is qualified, as a minimum, to LOCA conditions as defined in PVNGS 1, 2, & 3 PSAR Tables 3.11-1 & 3.11-2. This qualification is more severe than any temperature, humidity or radiation conditions expected during normal operation, shutdown or standby. Two 100% capacity normal environmental control systems are provided in containment with automatic failover from the operating system to the standby system. In addition, the containment is equipped with temperature and humidity monitors which alarm in the control room.

Equipment located in the turbine driven auxiliary feed pump room located in the Main Steam Support Structure (Item 2b) is qualified to 122°F. The maximum temperature in the pump room twenty-four hours after failure of the normal environmental control system, with the pump running, is conservatively calculated to be less than this qualification temperature. Therefore, no monitoring is required.

The Main Steam Support Structure above elevation 100' (Item 2c) is open to natural circulation of outside air. All safety related equipment in this area is qualified for a main steam line break in the Main Steam Support Structure. This qualification is sufficient to ensure that the safety related equipment in the Main Steam Support Structure will not be exposed to environmental conditions during normal operation, shutdown or standby for which it has not been qualified.

B1. The environmental control systems for the areas listed in Response A are described in detail in PVNGS 1, 2 & 3 PSAR Sections 6.4 and 9.4. In general, for areas containing safety related equipment two separate environmental control systems are provided. One system operates during normal plant conditions. The second (essential) system operates during emergency conditions.

B.2 PVNGS 1, 2 & 3 PSAR Table 9.4-2 summarizes the inside design conditions for the buildings, areas and rooms listed in Response A. The normal and essential environmental control systems are designed to maintain these conditions during normal plant operation and during shutdown or standby conditions. Either the normal or the essential environmental control system is capable of maintaining these environmental conditions.

In order to assure the function of the safety related equipment listed in Response A during accident conditions, the equipment is qualified to perform its safety related function in the environmental conditions listed in PVNGS 1, 2, & 3 PSAR Tables 3.11-1 & 3.11-2.

Three factors are considered in defining the environmental conditions for safety related equipment: temperature, humidity and radiation exposure.

As shown in Tables 3.11-1 & 3.11-2 all safety related equipment is qualified to the radiation exposure expected during the plant life. Therefore, no monitoring of this environmental factor is required.

B.3 The following plant areas (as defined in Response A) contain temperature switches which alarm in the control room:

1. Auxiliary Building
   a. HPSI pump rooms
   b. LPSI pump rooms
   c. CSS pump rooms
   d. ECW pump rooms
   e. ESF electrical penetration areas
   f. CEDM control cabinet room
2. Main Steam Support Structure
   a. Motor driven auxiliary feed pump room
3. Diesel Generator Building
   a. DG rooms
   b. DG control areas

The following plant areas (as defined in Response A) are monitored by a temperature switch (alarmed in the control room) in the supply duct or exhaust plenum of the normal environmental control system:

4. Control Building
   b. ESF switchgear rooms
   c. ESF equipment rooms
   d. Essential battery rooms
   e. Remote shutdown panel area
5. Fuel Building Essential Exhaust Filter

Temperature switches located in individual rooms alarm at a setpoint below the qualification temperature of the equipment located in that room. Upon receipt of an alarm, the essential environmental control system for that room would be started to maintain room temperature below the equipment qualification temperature. Operation of either the normal or essential environmental control system will ensure control of humidity below that for which the equipment is qualified. Therefore, only temperature switches are provided in each room.

Temperature switches located in the supply duct of normal environmental control systems monitor the performance of that system. As there are no significant heat sources in the control building areas monitored by these temperature switches, monitoring of individual rooms is not required. Proper operation of the normal environmental control system will ensure control of temperature and humidity below that for which the equipment is qualified. Should a high temperature be alarmed, the essential environmental control system for these areas would be started.

The control room area of the Control Building (Item 4.a of Response A) is maintained at 70°-$80^{\circ}F$ by its normal environmental control system (PVNGS 1, 2 & 3 PSAR Table 9.4-2). Safety related equipment located in this area is qualified for a maximum of $104^{\circ}F$ (PVNGS 1, 2 & 3 PSAR Table 3.11-2. As this area is continuously manned, failure of the normal environmental control system would be detected by operating personnel before the temperature exceeded the equipment qualification temperature. The essential environmental control system would then be started. The differential of 20°-$30^{\circ}F$

between normal room temperature and equipment qualification temperature provides ample margin for detection of a failure of the normal environmental control system.

B.4 Temperature alarms described in Response B.3 will alarm whenever temperatures exceed the inside design conditions for the monitored area. The setpoint of these alarms is below the qualification temperature of the equipment they are monitoring. In addition per Regulatory Guide 1.47, whenever an essential environmental control system is bypassed or inoperable, this condition will be alarmed in the control room. Administrative procedures will require operating personnel to respond to these alarms and ensure that either an environmental control system is operating or if all systems are shutdown that portable temperature monitoring equipment is installed nearby the affected safety related equipment. Administrative shutdown of all environmental control systems in an area will not be allowed.

B.5 Temperature alarm setpoints are set below the qualification temperature of the equipment affected to allow sufficient time for corrective action or installation of portable monitoring equipment prior to qualification temperatures being exceeded. The monitoring equipment will be of high quality, continuously powered while installed and will provide a continuous temperature record during the time that temperatures exceed the equipment's qualification. It should be noted that it is highly unlikely that both normal and essential environmental control systems will simultaneously be inoperable requiring installation of the protable monitoring equipment. The monitoring equipment will be periodically tested and calibrated.

B.6 The record of the abnormal environmental conditions in an area will be compared to the original equipment qualification records of the equipment in the affected areas and analyzed. If required, appropriate action will be taken to restore the equipment qualification.

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Instrument and Control Systems

Question 3

Qualification Review Item - (E-33) - Cooling Water Supply to Reactor Coolant Pumps.

Your response to this item does not satisfy the Staff's requirements. Based on Combustion Engineering's commitment to demonstrate that the reactor coolant pumps will operate longer than 30 minutes without cooling water, it is the staff's position that safety grade instrumentation with alarms in the control room should be provided to detect the loss of cooling water to the pumps to ensure that the operator will have sufficient time to initiate manual protection of the plant. The entire instrumentation system, including audible alarms and visual status indicators, should meet the requirements of IEEE Standard 279-1971, and the preliminary design should be documented in accordance with the requirements of Regulatory Guide 1.70, Section 7.6.

Response:

Safety grade instrumentation with alarms in the control room will be provided to detect loss of cooling water to the reactor coolant pumps. This will consist of two redundant sensor to alarm circuits, including audible alarms and visual status indicators, that will meet the requirements of IEEE 279-1971.

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REQUEST FOR ADDITIONAL INFORMATION

Instrument and Control Systems

Question #4:

Ref. Qualification Review Item B. - Changes from Documented Base Plant Design

Tables 7.3-7 and 7.3-14 give design basis events and monitored variables for emergency safety feature system protective action. In Amendment 17 these tables show design changes that have occurred since the documented design of the base plant. Initiation of the containment purge isolation system (CPIS) in the event of a loss-of-coolant-accident (LOCA) was deleted from Table 7.3-7, and initiation of the CPIS and the control room essential ventilation system (CREVS) upon a pressurizer pressure signal or containment pressure signal was deleted from Table 7.3-14. There is a contradiction between the tables with respect to the CREVS since Table 7.3-7 shows the CREVS will be actuated in the event of a LOCA whereas Table 7.3-14 does not show this. Also, Table 6.2-16 shows the containment purge isolation system to be actuated by various signals described in Chapter 7.

It is the staff's position that in the event of a loss-of-coolant-accident the containment purge isolation system and control room essential ventilation system should be initiated as required by Branch Technical Position CSB 6-4 and General Design Criteria 19 and 20 of Appendix A to 10 CFR Part 50, "General Design Criteria for Nuclear Power Plants".

We request that as a minimum you:

A. Describe how your design satisfies the staff's requirements as stated in Branch Technical Position CSB 6-4 and the requirements of GDC 19 and GDC 20.

B. Identify and justify each exception, and

C. Clarify the contradictions between Tables 7.3-7, 7.3-14, and 6.2-16.

Response #4

The Containment Purge Isolation System and Control Room Essential Ventilation System are designed to fully satisfy the requirements of BTP CSB 6-4 and GDC 19 and 20 of 10 CFR 50, Appendix A.

Refer to PSAR Section 7.3.1.2.8 for a description of how the output signals from either a Containment Purge Isolation Actuation Signal (CPIAS) or a Containment Isolation Actuation Signal (CIAS) are combined in the control circuits of the valving common to both the containment purge isolation system and the containment isolation system such that either high radiation due to a refueling accident in the containment or a LOCA will isolate the valves used for containment purge.

Refer to PSAR Section 7.3.1.2.9 for a description of how the Control Room Essential Filtration Actuation Signal (CREFAS) signal is initiated and how the output signals from either CREFAS or

Safety Injection Actuation Signal (SIAS) are combined in the control circuits of the control room essential filtration system such that either high airborne activity in the control building intake, in the fuel building, or in the containment after a fuel handling accident, or a LOCA, will start the control room essential filtration system and isolate the normal control room HVAC system.

Tables 7.3-9 and 7.3-14 show the primary variables used to provide the CREFAS and CPIAS. After a LOCA the control room essential ventilation system will be actuated by a SIAS and the containment purge isolation system will be actuated by a SIAS and CIAS.

PALO VERDE NUCLEAR GENERATING STATION

UNITS 4 AND 5

DOCKET NOS. STN 50-592 AND STN 50-593

REQUEST FOR ADDITIONAL INFORMATION

Conduct of Operations

Question 1

In view of the added responsibilities for Units 4 and 5, describe your plans for expanding the capability of your Nuclear Services Department. This may be done by updating Tables 13.1-1 and 13.1-2 of your application.

Response:

As each of the PVNGS units are scheduled for operaion two years apart, the peak number of personnel required to support Units 4 and 5 during the construction phase is not expected to exceed those shown in PVNGS 1, 2 and 3 PSAR Table 13.1-1 and 13.1-2. As the work load required for the design, construction and start-up for Units 1, 2 and 3 decreases, the workload for offsite engineering support and construction for Units 4 and 5 will be increasing, hence the overall effect being a manpower leveling. Inherent in this logic is the fact that Engineering design will be complete and the bulk of the procurement work will be complete once these tasks are completed for Unit 1. In any case, the manpower requirements are periodically reviewed and adjustments made as the workload dictates.

PALO VERDE NUCLEAR GENERATING STATION

UNITS 4 AND 5

DOCKET NOS. STN 50-592 AND STN 50-593

REQUEST FOR ADDITIONAL INFORMATION

Conduct Of Operations

Question 2

Describe the functions and responsibilities of the Assistant Plant Manager and the Day Shift Supervisor.

Response:

The responses are given in amended Section 13.1.2.2.1 and 13.1.2.2.10.6, respectively.

PALO VERDE NUCLEAR GENERATING STATION

UNITS 4 AND 5

DOCKET NOS. STN 50-592 AND STN 50-593

REQUEST FOR ADDITIONAL INFORMATION

Conduct of Operations

Question 3

Describe the functional differences between the responsibilities of the Unit Maintenance Supervisors and the General Service Manager.

Response:

The response is given in amended Section 13.1.2.2.4.

PALO VERDE NUCLEAR GENERATING STATION

UNITS 4 AND 5

DOCKET NOS. STN 50-592 AND STN 50-593

REQUEST FOR ADDITIONAL INFORMATION

Conduct of Operations

Question 4

Describe the shift crew compositions for the operation of the fourth and fifth units.

Response:

During normal, five-unit operation, each unit shift complement will be as described in the PVNGS 1, 2 and 3 PSAR Section 13.1.2.3 as referenced in the PVNGS 4 and 5 PSAR.

PALO VERDE NUCLEAR GENERATING STATION
UNITS 4 AND 5

DOCKET NOS. STN 50-592 AND STN 50-593
REQUEST FOR ADDITIONAL INFORMATION

Conduct of Operations

Question 5

Describe the qualification requirements for the positions of Assistant Plant Manager, Supervising Nuclear Engineer, Day Shift Supervisor, Nuclear Plant Equipment Operator, I&C Foreman and Technicians, Maintenance Supervisor, Maintenance Foreman, Plant Electricians and Mechanics, Chemistry Technicians, Radiation Protection Foreman and Technicians, and the General Services Manager.

Response: The qualification requirements for these positions are referenced to the appropriate section of ANSI N18.1-1971 in the following table:

Position	N18.1 Section Number
Assistant Plant Manager	4.2.1
Supervising Nuclear Engineer	4.4.1
Day Shift Supervisor	4.3.1
Nuclear Plant Equipment Operator	4.5.1 (unlicensed operator)
I&C, Maintenance, & Rad. Protection Foreman	4.3.2
I&C, Chemistry, and Rad. Protection Technicians	4.5.2
Plant Electricians and Mechanics	4.5.3
Maintenance Supervisor	4.2.3
General Services Manager	4.3.2

for their assigned duties in accordance with ANSI N18.1-1971, Selection and Training of Nuclear Power Plant Personnel.

13.1.2.1.1 Operating Unit

The operating unit is responsible for the operation of all plant equipment and systems in a safe, reliable and efficient manner under the direction of the Unit Superintendent. Reporting to the Unit Superintendent are five departments. The unit departments are unit operating; unit instrumentation and control; unit maintenance; unit chemistry and unit radiological protection.

13.1.2.1.2 Personnel and Administrative Unit

The personnel and administrative unit provides and/or administers services such as station security, material supply and storage, clerical, record retention, budgets and safety.

13.1.2.1.3 Engineering and Technical Department

The engineering and technical department consists of engineers and technicians performing onsite support functions in the areas of reactor physics, radiological and environmental protection, computer systems, fuel management, nuclear technology, reactor safety and licensing activity. Personnel with appropriate experience and specialized training will be assigned in these areas of responsibility.

13.1.2.1.4 General Services Department

The general services department provides trained personnel and supervision to perform building services, radwaste transportation and water reclamation plant operation; and maintenance support to the units when requested by Unit Superintendents.

13.1.2.2 Station Personnel Responsibilities and Authorities

13.1.2.2.1 Station Manager and Assistant Station Manager

The Station Manager reports to the Vice President of Electrical Operations and has direct responsibility for the safe, reliable and efficient operation of PVNGS. He is responsible for control of onsite personnel activities and for complying with the station's operating license, applicable federal and state regulations, and policies established by APS. It is the policy of APS to maintain occupational exposures at the lowest practicable level. The Assistant Station Manager provides administrative and technical assistance to the Station Manager. In the absence of the Station Manager, his responsibilities for the day-to-day operation, maintenance and performance of the station are assigned to the Assistant Station Manager, or he may delegate other qualified subordinates. The PVNGS Station Manager and Assistant Station Manager shall satisfy the qualification requirements of ANSI N18.1-1971 Section 4.2.1 prior to the initial core loading of the PVNGS Unit 1.

13.1.2.2.2 Engineering and Technical Manager

The Engineering and Technical Manager is responsible to the Station Manager for the onsite engineering and technical work required to ensure proper functioning of the nuclear plant. He directs the electrical, nuclear, mechanical, radiological and environmental supervisc and the technical supervisor in the performance of their duties. In the absence of the Engineering and Technical Manager, his responsibilities shall be assigned to the Technical Supervisor and/or a qualified supervising engineer. The Engineering and Technical Supervisor shall satisfy the qualification requirements of the ANSI N.18.1-1971 Section 4.2.4 prior to initial fuel loading of the PVNGS Unit 1.

13.1.2.2.3 Personnel and Administrative Manager

The personnel and administrative manager is responsible to the station manager for maintaining the station central personnel file, technical file and record file. He directs the activities of the security supervisor, office supervisor and safety director.

13.1.2.2.4 General Services Manager

The General Services Manager is responsible to the Station Manager for the operation and maintenance of mechanical and electrical equipment outside the unit's power block. He will also provide maintenance support to the unit power blocks as needed. This includes maintaining of required equipment and manpower records, overall upkeep of the station, and the operation of the central station support shop.

The General Services Manager shall satisfy the qualification requirements of ANSI N18.1-1971 Section 4.3.2 prior to the initial core loading of PVNGS Unit 1.

13.1.2.2.5 Unit Superintendents

The Unit Superintendents are responsible to the Station Manager for the conduct of all operating, maintenance and testing activities in their respective unit plants. They direct the activities of the unit operating supervisor, instrumentation and controls supervisor, maintenance supervisor, chemistry supervisor and radiation protection supervisor. They coordinate with the other area managers and supervisors for plant support services.

The Unit Superintendents shall satisfy the qualification requirements of ANSI N18.1-1971, Section 4.2.2 prior to the initial core loading of the respective unit. In the absence of the unit superintendent, his responsibilities shall be assigned to the operating supervisor and/or other qualified subordinates.

13.1.2.2.6 Nuclear Engineering Supervisor

The Nuclear Engineering Supervisor is responsible to the Engineering and Technical Manager for core monitoring and incore fuel management programs and for core physics test programs. Prior to initial core loading of PVNGS Unit 1, he shall satisfy the qualification requirements of ANSI N18.1-1971, Section 4.4.1.

13.1.2.2.7 Quality Supervisor

The Quality Supervisor is responsible to the Station Manager. He is responsible for the Operating Quality Assurance Program and reviews, audits and observes station operations to ensure compliance with licensing requirements. Prior to initial core loading of PVNGS Unit 1, the qualifications of the Quality Supervisor will include a minimum of 6 years experience in quality assurance, of which 6 months shall be nuclear quality assurance. A minimum of 2 years experience should be related technical training. A maximum of 4 years of this 6 years experience may be fulfilled by related technical or academic training.

13.1.2.2.8 Radiological and Environmental Supervisor

The Radiological and Environmental Supervisor is responsible to the engineering and technical manager for the preparation and coordination and conduction of the station radiological and environmental monitoring programs. The unit chemistry supervisor and radiation protection supervisor have the responsibility for the conduct of the program within their respective units. The objective of the radiological protection program will be to delineate the operating philosophy and procedures for maintaining occupational radiation exposures as low as reasonably achievable. Prior to initial core loading of PVNGS Unit 1, the qualifications of the radiological and environmental engineer will be in compliance with Regulatory Guide 1.8, Section C, as it pertains to Radiation Protection Manager.

13.1.2.2.9 Training Supervisor

The Training Supervisor is responsible to the Station Manager for the preparation, coordination and conduct of the station training program. He directs the activities of the simulator supervisor and the nuclear plant instructors.

13.1.2.2.10 Unit Department Supervisors

13.1.2.2.10.1 Unit Maintenance Supervisor. The Unit Maintenance Supervisor is responsible to the Unit Superintendent for organizing and conducting preventive maintenance and repairs of the respective unit's mechanical and electric equipment. This includes maintaining all required records for equipment, the overall upkeep of the unit and an adequate inventory of spare parts and consumables. Prior to initial core loading of the respective unit, he will satisfy the qualification requirements of Section 4.2.3 of ANSI N18.1-1971.

13.1.2.2.10.2 Unit Chemistry Supervisor. The Chemistry Supervisor is responsible for the conduct of the unit water chemistry program including analysis and execution of analytical procedures, with emphasis on primary, secondary, and circulating water chemistry. He trains personnel in the use of equipment necessary to control water chemistry and coordinates with the radiation protection supervisor, radiation exposures and contamination problems associated with the primary plant. Prior to initial core loading of the respective unit, the qualifications for this position will meet or exceed those described in Section 4.4.3 of ANSI N18.1-1971.

13.1.2.2.10.3 Unit Radiation Protection Supervisor. The Unit Radiation Protection Supervisor is responsible to the unit superintendent for the conduct of the radiation protection program in the respective unit. His responsibilities include analysis and execution of analytical procedures, with special emphasis on control of radiation exposures to all personnel; including preparation of related records, maintenance of accumulated doses, conduct of the environmental monitoring requirements, surveillance and approval of radioactive waste disposal activities. In the absence of the radiation protection supervisor, his responsibilities are assigned to a qualified subordinate or other plant staff member. Prior to initial core loading of the respective unit, the qualifications for this position will be in compliance with Section 4.4 of ANSI N18.1-1971.

13.1.2.2.10.4 Instrumentation and Control Supervisor. The Instrumentation and Control Supervisor is responsible to the unit superintendent for the proper functioning and repair of the unit's instrumentation and control system. This includes preventive maintenance calibration and testing of instruments and controls, supervision of assigned technicians, establishing procedures for instrument and control repair, test and calibration, and maintaining all required records for equipment under his jurisdiction. The instrumentation and control supervisor will satisfy the qualification requirements of ANSI N18.1-1971, Section 4.4.2 prior to initial core loading.

13.1.2.2.10.5 Unit Operating Supervisor. The Unit Operating Supervisor is responsible to the unit superintendent for the conduct of the unit operations in a safe and efficient manner in accordance with technical specification and station instructions. He supervises the activities of the unit's operating personnel. The Unit Operating Supervisor shall satisfy the qualification requirements of ANSI N18.1-1971, Section 4.3.1 prior to initial core loading of the respective unit.

13.1.2.2.10.6 Day Shift Supervisor. The Day Shift Supervisor assists the Unit Operating Supervisor with his responsibility as he directs. He provides on-shift coverage as required (e.g., during refueling outages) and, as directed by the Unit Operating Supervisor, assists the Shift Supervisors with their responsibilities. He acts for the Unit Operating Supervisor during his absence. Prior to initial core loading of the respective unit, the Day Shift Supervisor shall satisfy the qualification requirements of Section 4.3.1 of ANSI N18.1-1971.

13.1.2.2.10.7 Shift Supervisor. The Shift Supervisor is responsible for the safe, reliable and efficient operation of the unit plant during his assigned shift. He directs the activities of the operators on his shift, coordinates all maintenance activities performed while he is on duty, and ensures compliance with all required radiological control procedures. Should the

13.1.2.3 Shift Crew Composition

During normal operation, each unit shift complement consists of the following:

1 - Shift Supervisor

1 - Supervising Nuclear Plant Operator

1 - Nuclear Plant Operator

2 - Nuclear Plant Equipment Operators

The shift supervisors and supervising nuclear plant operators are Senior Reactor Operator licensed. The nuclear plant operators will hold a Reactor Operator's license.

13.1.3 QUALIFICATION REQUIREMENTS FOR NUCLEAR PLANT PERSONNEL

13.1.3.1 Minimum Required Qualifications

The minimum qualifications of all managerial and supervisory technical personnel will meet the requirements of ANSI N18.1-1971, Standard for Selection and Training of Personnel for Nuclear Power Plants.

shift supervisor be absent or incapacitated, the supervising Nuclear Plant Operator will assume his responsibilities. Shift Supervisors shall satisfy the qualification requirements of ANSI N18.1-1971, Section 4.3.1 prior to initial core loading of the respective unit.

13.1.2.2.10.8 Supervising Nuclear Plant Operator. The Supervising Nuclear Plant Operator is responsible for operation of the unit from the control room. He directs the nuclear plant operator and nuclear plant equipment operators, on the operation of the unit, maintains the log and other required records. The supervising nuclear plant operator shall satisfy the qualification requirements of ANSI N18.1-1971, Section 4.5.1 prior to initial core loading for the respective unit.

13.1.2.2.10.9 Nuclear Plant Operator. The Nuclear Plant Operator maniuplates the reactor plant controls. The nuclear plant operator shall satisfy the qualification requirements of ANSI N18.1-1971, Section 4.5.1 prior to initial core loading for the respective unit.

13.1.2.2.10.10 Nuclear Plant Equipment Operator. The Nuclear Plant Equipment Operator is responsible under the direction of the shift supervisor and supervising nuclear plant operator, for operating auxiliary systems and assisting in the refueling of the plant as directed. He shall satisfy the unlicensed operator qualification requirements of Section 4.5.1 of ANSI N18.1-1971 prior to initial core loading for the respective unit.

PALO VERDE NUCLEAR GENERATING STATION

UNITS 4 AND 5

DOCKET NOS. STN 50-592 AND STN 50-593

REQUEST FOR ADDITIONAL INFORMATION

Radiation Protection

Question 1

Provide information concerning action taken to maintain occupational radiation exposure as low as is reasonably achievable by minimizing and controlling the buildup, transport and deposition of activated corrosion products in reactor coolant and auxiliary systems. Include as a minimum information on the following steps taken to minimize Co-58 and Co-60, including:

a. The use of reduced nickel in primary coolant systems alloys.

b. Low cobalt impurity specifications in primary coolant systems alloys.

c. The minimization of high cobalt, hard facing wear materials in the primary coolant system.

d. The use of high flow rate/high temperature filtration.

e. The selection of valves and packing materials to minimize crud buildup and maintenance.

f. Provisions of decontamination of reactor coolant components and systems.

Response:

This question is generally within the CESSAR scope. In the design of the NSSS, consideration for minimizing and controlling the buildup, transport and deposition of activated corrosion products is a major concern. Therefore, this matter is addressed in CESSAR on its docket.

PALO VERDE NUCLEAR GENERATING STATION

UNITS 4 AND 5

DOCKET NOS. STN 50-592 AND STN 50-593

REQUEST FOR ADDITIONAL INFORMATION

Radiation Protection

Question 2

Provide information concerning action taken to maintain occupational radiation exposure as low as is reasonably achievable during the eventual decommissioning of the reactor plants.

Response:

Procedures for decommissioning, currently, are not detailed enough to incorporate specific provisions for maintaining exposures ALARA. The NRC is presently developing procedures for decommissioning reactors. It is anticipated that this development will result in more definitive policies and procedures for decommissioning within 2-4 years.

In general, decommissioning of reactors is a process which can use several radiation reduction methods for the protection of both the public and plant workers involved in the decommissioning effort. The process of decommissioning may, for example, include decontamination of the reactor system. This is but one of the several methods (e.g., use of shielding) available to reduce radiation levels. In addition, the specific ALARA considerations for operation and maintenance (refer to CESSAR and PVNGS 1, 2 & 3 PSAR Chapter 12) will lower the activity levels during operation and result in greatly reduced man-rem exposures for decommissioning.

PALO VERDE NUCLEAR GENERATING STATION

UNIT NOS. 4 AND 5

DOCKET NUMBERS 50-592/593

REQUEST FOR ADDITIONAL INFORMATION

Auxiliary Systems

Question 1

In Amendment 17 to the Palo Verde, Unit Nos. 1, 2, and 3 PSAR, Section 6.4.1 was revised to delete the smoke detectors located in the control room ventilation system supply duct at the outside air intakes, without providing a justification. Appendix A of NRC Branch Technical Position ASB 9.5-1, Item D.2, Control Room (Page 36) recommends smoke detectors in the control room ventilation intake which alarm in the control room and alert the operators to manually isolate the control room ventilation system thus preventing smoke from entering the control room.

It is our position that smoke detectors be provided in the control room ventilation system air intakes, as they were in the originally approved system design.

Response 1

Smoke detectors are provided in the control room ventilation system air intakes. These detectors alarm in the control room.

PALO VERDE NUCLEAR GENERATING STATION

UNITS 4 AND 5

DOCKET NOS. STN 50-592 AND STN 50-593

REQUEST FOR ADDITIONAL INFORMATION

Fire Protection

Question A

In Part III, Position A.1. (PVNGS Fire Protection Evaluation), you state that overall responsibility for the fire protection will be assigned to the Palo Verde Nuclear Station Plant Manager. This is an onsite position. Describe the upper level offsite management position which will have overall responsibility for the formulation and implementation of the fire protection program.

Response

As required by Appendix A of NRC Branch Technical Position Paper 9.5-1, responsibility for the overall fire protection program is assigned to a designated person in the upper level of management. This designated person is the PVNGS Plant Manager. No further assignment of responsibility is required.

11/8/78
Rev. 1

PALO VERDE NUCLEAR GENERATING STATION
UNITS 4 AND 5
DOCKET NOS. STN 50-592 AND STN 50-593
REQUEST FOR ADDITIONAL INFORMATION
QUALITY ASSURANCE PROGRAM

We have compared the description of the QA program for Palo Verde 4 and 5 with the QA program previously accepted (7/25/75) for Palo Verde 1, 2 and 3. Listed below are items related to quality assurance for Palo Verde Units 4 and 5 that require clarification.

1. Inconsistencies between Appendix 3J, "Conformance with AEC Regulatory Guides", and lists of similar guidance documents on pages 17.1A-18 and 19 and pages 17.1B-2 and 3 must be corrected. The inconsistencies are shown on blue pages indicating changes by the applicant in the PVNGS-1, 2 and 3 PSAR pages. The inconsistencies are illustrated on the enclosed table. The commitments must be at least as up-to-date as given in the response to Question 17A.35 on page 17A-22 of the PVNGS-1, 2 and 3 PSAR.

Response:

The inconsistencies will be resolved by changing Section 17.1A and 17.1B to reflect the dates as expressed in Appendix 3J. In addition, Paragraph 17.1A.2.6, "Applicability of Codes, Standards and Regulatory Guides", will be revised to include the following:

A. ANSI N45.2.5 (Draft 3, Revision 1, January 1974): "Supplementary Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Steel During the Construction Phase of Nuclear Power Plants"

B. ANSI N45.2.8 (Draft 3, Revision 3, April 1974): "Supplementary Quality Assurance Requirements for Installation, Inspection and Testing of Mechanical Equipment and Systems for the Construction Phase of Nuclear Power Plants"

~~C. Item I, ANSI N45.2.13 will be revised to read, (Draft 2, Revision 4, April 1974)~~.

2. The three questions on fire protection quality assurance for PVNGS-1, 2 and 3 forwarded in John Stolz's letter of August 14, 1978 must be also satisfactorily resolved for Units 4 and 5.

Response:

The response submitted in our letter ANPP-11734-JAR, dated September 13, 1978, also applies to Units 4 and 5.

3. The commitment to Combustion Engineering, Inc.'s Topical Report must be in the PSAR, not in the FSAR. (This is a follow-on of Item B.2 to the

Qualification Review Letter, dated December 12, 1977.)

Response:

The PSAR will be revised to reflect the commitment to the Combustion Engineering, Inc.'s Topical Quality Assurance Program CENPD-210A, Revision 3.

4. Amendment 17 deleted the following sentence from 17.1A.13 of the PVNGS-1, 2 and 3 PSAR: "C-E has responsibility for establishing requirements for shipping, handling, and storage of NSSS components at the site." Clarify who now has this responsibility.

Response:

Bechtel has the responsibility for the shipping, handling, and storage of NSSS components, in accordance with the criteria established by Combustion Engineering, after they have arrived at the Palo Verde site. Paragraph 17.1A.13 of the PSAR, "Handling, Storage and Shipping", will be revised to reflect this responsibility.

ENCLOSURE 4

PVNGS OFFSITE POWER SYSTEM

I. ESF Loads Approximately 5300 KW per Train

II. ESF Transformer Sized 10/12.5 MVA, OA/FA

- A. Twice the Size Required for a Single Train Loading
- B. Capable of Supplying Both ESF Trains With Manual Interconnection
- C. Impedance 5% at OA Rating

III. Startup Transformer Sized 42/56/70 MVA, OA/FOA/FOA

- A. Sized for Accident Loads of One Unit and Normal Shutdown Loads of Another Unit Plus Margin
- B. Impedance 6.5% at OA Rating

SEQUENCING WITH OFFSITE POWER AVAILABLE

I. Maintains Voltage Levels Above Minimum Starting Requirements

II. Accepted Electrical Industry Practice

II. At Least Twelve Other Applicants Have Proposed Similar Sequencing Schemes

III. Sequencer Design Minimizes Potential for Malfunction

   A. Utilizes Solid State Circuitry

   B. Incorporates Self-Test Features

IV. Complies With The Requirements of GDC 17

VOLTAGE DROP CALCULATIONS

I. ASSUMPTIONS

A. 95% GRID VOLTAGE

B. SIMULTANEOUS STARTING OF ALL LOCA LOADS

II. VOLTAGE DROPS

A. 4160 V LEVEL: 79%

B. 480 V LOAD CENTER: 70%

C. 480 V MCC: 67%

III. MINIMUM VOLTAGE REQUIRED TO START AND ACCELERATE LOADS

A. 4160 V LEVEL: 75%

B. 480 V LEVEL: 75%

ELIMINATION OF SEQUENCING FEATURE

I. Requires Enlargement of ESF and Load Center Transformers

II. Requires Increased Fault Duty Ratings for Class IE Switchgear

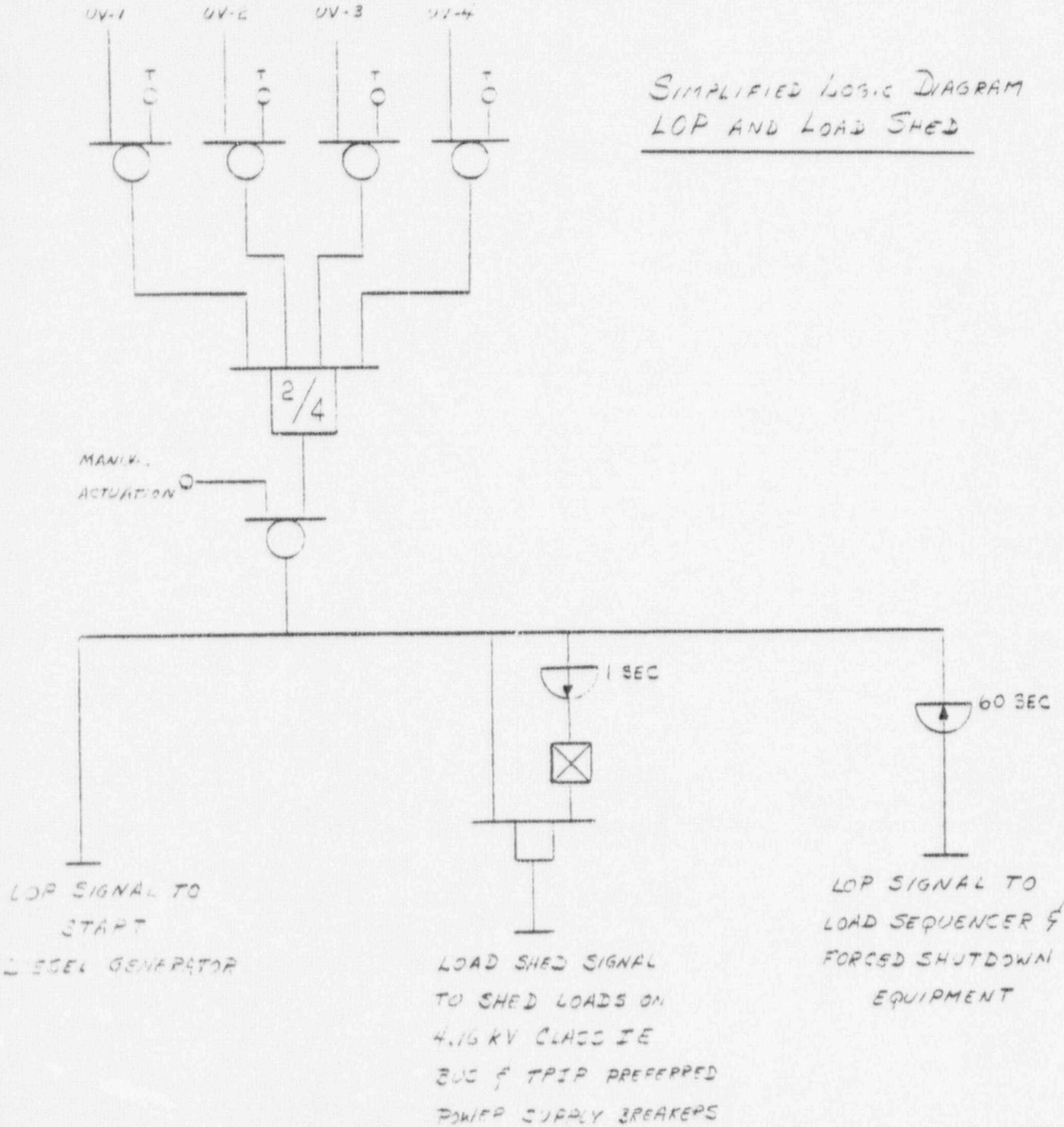
III. Requires Redesign of Entire Class IE Power System

IV. Delays Involved:

A. Engineering Effort

B. Equipment Procurement

SIMPLIFIED LOGIC DIAGRAM
LOP AND LOAD SHED



PVNGS LOAD SHED FEATURES

I. Relaying Provides Reliable Indication of LOP

   A. 2 Out of 4 Coincidence Logic

   B. Induction Disc Relays

   C. Relays Set to Maintain Connection to Offsite Power As Long as Possible Without Jeopardizing Function of Class IE Devices

   D. Relays Operate off the Class IE 4160 V Bus to Monitor Either Offsite or Onsite Power

II. Load Shed During Diesel Generator Load Sequencing Prevented By:

   A. Proper Setting of Induction Disc Undervoltage Relays

   B. Proper Sequencer Design, Minimizing Voltage Drops

III. Load Shed During Fault Conditions Prevented by Inverse Time Characteristic of Induction Disc Relays

IV. Retention of Load Shed Feature with Onsite Power Prevents Equipment Breaker Lockout on Undervoltage Conditions

V. Retention of Load Shed Feature with Onsite Power Allows Operator Action to Provide Alternate Power Source Upon Diesel Generator Malfunction

PALO VERDE NUCLEAR GENERATING STATION

UNITS 4 AND 5

DOCKET NOS. STN 50-592 AND STN 50-593

REQUEST FOR ADDITIONAL INFORMATION

Power Systems

Question 2

It is not clear from the information provided in response to qualification review item E-32 position 1, if a second level of under voltage protection is provided (in addition to the existing under voltage protection) that will automatically perform the function of switching from offsite power (the preferred source) to the onsite power sources in case of sustained degraded voltage conditions at the offsite power source. This second level of protection is a requirement. Therefore, provide a modified response to this item and include the setpoints and time delays associated with the first and second levels of undervoltage protection.

Response:

A second level of voltage protection with time delay would be required for use with an undervoltage protection scheme that utilized instantaneous relays with time delays. The primary undervoltage protection for such a scheme would be typically set to operate at about 70% of nominal voltage with appropriate time delays. Such a scheme would not detect a degraded grid condition between 70% and 100% of the nominal voltage. Monitoring this range of voltage would require the use of a second level of voltage protection.

The undervoltage protection scheme provided for the PVNGS is adequate to detect loss of offsite power at 4160 volt ESF busses and to protect the onsite power system from any adverse effects that could result from a sustained degraded voltage condition on the offsite power system. The PVNGS design for the undervoltage protection utilizes induction disc relays with inverse voltage-time relationship measured from approximately 90% nominal bus voltage down to zero voltage. This system inherently provides a second level of undervoltage protection. Since the Class IE motors for PVNGS are specified to start and accelerate their loads at 75% rated voltage and to operate continuously at 90% of rated voltage, any bus voltage that falls below 75% voltage for short time periods and below 90% for long time periods will be considered a degraded condition. Such degraded conditions will be monitored by induction disc relays and will generate a loss of voltage signal. Section 8.3.1.1.2.11 B of the PVNGS 1, 2 & 3 PSAR summarizes the setpoint and design criteria for these relays.

Reliability of the undervoltage detection is assured by the use of four relays, with 2 out of 4 coincidence logic, on each of the 4160 volt Class IE busses.

PALO VERDE NUCLEAR GENERATING STATION

UNITS 4 AND 5

DOCKET NOS. STN 50-592 AND STN 50-593

REQUEST FOR ADDITIONAL INFORMATION

Power Systems

Question 3

Your response to position 2 of item E32 is unacceptable because 1) your design does not prevent load shedding of the emergency buses once the onsite sources are supplying power to all sequenced loads on the emergency busses and, 2) your design does not include the capability of the load shedding feature to be automatically reinstated if the onsite source supply breakers are tripped. Provide a modified response to this item that meets our requirements or provide full justification for your proposed design on some other defined bases.

Response:

1) Prevention of the load shedding feature for the Class IE 4.16kv buses would be required if an undervoltage protection system design were based on the utilization of undervoltage relays with instantaneous characteristics. The undervoltage relays utilized for PVNGS as described in Section 8.3.1.1.2. 11.B of the PVNGS PSAR have inverse voltage-time characteristics. On a two-out-of-four coincidence logic a single pulse load shed signal is generated for a degraded bus voltage condition or for a loss of offsite power.

The undervoltage relays are adjusted to initiate a load shed when conditions are present such that there is insufficient bus voltage available to accelerate a motor (less than 75% of rated voltage for short period of time) or a sustained undervoltage condition exists (less than 90% of rated voltage for long duration). The diesel generators are designed to be consistent with the recommendations of Regulatory Guide 1.9. The undervoltage relays are not expected to generate an undervoltage signal during normal bus sequencing due to the time delay inherent in their inverse voltage-time operation.

2) Section 8.3.1.1.3.6 of the Unit 1, 2 & 3 PSAR states: "After load shed, tripping of the Class IE 4.16 kV bus offsite supply breaker and subsequent closing of diesel generator breaker to the Class IE 4.16kV bus, the undervoltage relays monitor the standby (onsite) power supply for an undervoltage occurrence. Should an undervoltage occur, the Class IE 4.16kV loads are shed and the loading sequence restarted."
This indicates that there is no need for reinstatement of the load shed feature since it is never disconnected from the Class IE 4.16kV busses. If the onsite source supply breakers are tripped, the undervoltage relays detect the subsequent bus undervoltage and initiate a load shed.

The load shed feature is retained in the PVNGS design when the onsite (standby) source is supplying power to the Class IE 4.16kV bus for the following reasons:

A. If a load shed is not generated during a degraded voltage condition, then a condition could exist where there is insufficient voltage on the bus for either accelerating Class IE motors or for the continuous operation of Class IE motors. If the load shedding feature was prevented, the affected motors and subsequent motors sequenced onto the bus would remain at approximately a locked rotor current condition, eventually tripping their associated circuit breakers. Allowing the load shed feature to clear the bus will enable the diesel generator to recover and be reloaded through the load sequencer. Without this feature tripping of the motor circuit breakers on locked rotor current conditions would lock-out the circuit breakers so they would have to be manually reclosed. This would result in their not being readily available when needed and could result in damage to the motors.

B. A load shed will not be generated if a short circuit occurs on motor feeders since the resultant voltage dip will not be detected by the load shed undervoltage relays by the time the fault is cleared by the motor circuit breaker and the voltage returns to normal on the bus. For complete description of electrical circuit protection refer to PVNGS PSAR Section 8.3.1.1.2.11.

PALO VERDE NUCLEAR GENERATING STATION

UNITS 4 AND 5

DOCKET NOS. STN 50-592 AND STN 50-593

REQUEST FOR ADDITIONAL INFORMATION

Power Systems

Question 4

Section 8.3.1.1.2.8 of the PSAR states that, "If preferred power is available to the Class IE bus following an engineered safety feature actuation signal, the required Class IE loads will be started through a sequencer." Provide your basis and justification for sequencing safety loads when preferred (offsite) power is available during the accident.

Provide a comparison on a bus by bus basis for all emergency buses of the voltage and motor starting transients associated with sequenced versus instantaneous loading for the condition of grid voltage at the low end of its normal range and maximum plant auxiliary load. In addition, address the loss of one startup transformer and the capability to fast transfer to the other startup transformer during this transient.

Provide a description of what would be required to remove this non-standard design feature from your design and the associated safety implications, if any.

Response:

Justification of sequencing the LOCA loads with offsite power available is that the simultaneous starting of these loads will depress the voltage at motor terminals below 75% on 480 volt loadcenters and MCC's. This is below the minimum to start and accelerate the motors. This conclusion is based on the attached calculations of voltage drop on starting all safety loads simultaneously.

We have investigated the use of this sequencing feature with fifteen other applicants, representing forty PWR's which are either under construction or have begun commercial operation within the last several years. Of these, twelve applicants, representing twenty-five units, sequence with offsite power available, either through the use of a sequencer or with individual time delay relays, while three applicants, representing fifteen units, do not sequence. This demonstrates that this should not be considered a non-standard design feature, but rather a standard alternative way of starting emergency loads with offsite power available.

The PVNGS design does not provide for automatic fast transfer from one start-up transformer source to another. This transfer is performed manually. Should one start-up transformer source be lost the ESF loads of the redundant train will be available to mitigate the consequences of an accident. This is in accordance with GDC 17. Per R.G. 1.93, operation is allowed for 72 hours in this LCO before proceeding to cold shutdown.

The PVNGS ESF transformers are sized to simultaneously handle the loads of both ESF trains. It is, therefore, sized twice as large as required for steady state operation. In addition, the transformer impedance has been specified as 5% to minimize the voltage drop effects.

The sequencer for PVNGS utilizes solid state logic with self-test features to enhance the reliability of sequencer operation.