CATEGORY 1

REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

ACCESSION NBR:9910210001 DOC.DATE: 99/10/08 NOTARIZED: YES DOCKET #
FACIL:STN-50-528 Palo Verde Nuclear Station, Unit 1, Arizona Publi 05000528
STN-50-530 Palo Verde Nuclear Station, Unit 3, Arizona Publi 05000530
AUTH.NAME AUTHOR AFFILIATION
MAULDIN,D. Arizona Public Service Co. (formerly Arizona Nuclear Power
RECIP.NAME RECIPIENT AFFILIATION
Records Management Branch (Document Control Desk)

SUBJECT: Application for amends to licenses NPF-41 & NPF-74,waiving requirement to perform SR 3.8.4.8 for Unit 1 channels A,B & C,on one time basis.Regulatory commitments & TS encl.

DISTRIBUTION CODE: A001D COPIES RECEIVED:LTR 1 ENCL 1 SIZE: 15+4
TITLE: OR Submittal: General Distribution

NOTES:STANDARDIZED PLANT 05000528
Standardized plant. 05000530

	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL
	JAMERSON,C	1	1	KALYANAM,N	1	1
	SC	1	1			
INTERNAL:	ACRS	1	1	FILE CENTER 01	1	1
	NRR/DSSA/SPLB	1	1	NRR/DSSA/SRXB	1	1
	NUDOCS-ABSTRACT	1	1	OGC/RP	1	0
EXTERNAL:	NOAC	1	1	NRC PDR	1	1

C A T E G O R Y 1 D O C U M E N T

NOTE TO ALL "RIDS" RECIPIENTS:
PLEASE HELP US TO REDUCE WASTE. TO HAVE YOUR NAME OR ORGANIZATION REMOVED FROM DISTRIBUTION LISTS OR REDUCE THE NUMBER OF COPIES RECEIVED BY YOU OR YOUR ORGANIZATION, CONTACT THE DOCUMENT CONTROL DESK (DCD) ON EXTENSION 415-2083

TOTAL NUMBER OF COPIES REQUIRED: LTTR 11 ENCL 10

AA2

APS

10 CFR 50.90
10 CFR 50.91

Palo Verde Nuclear Generating Station

David Mauldin
Vice President
Nuclear Engineering and Support

TEL (623) 393-5553
FAX (623) 393-6077

Mail Station 7605
P.O. Box 52034
Phoenix, AZ 85072-2034

102-04356-CDM/SAB/RKR
October 8, 1999

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Station P1-37
Washington, DC 20555-0001

References: 1. Letter 102-04053-JML/SAB/RMW, dated December 17, 1997, from J. M. Levine, APS, to NRC, "Request for Amendment to Technical Specification (TS) 3/4.8.2, Electrical Power Systems, D.C. Sources, and Bases Section 3/4.8."

2. Letter dated March 16, 1998, from J. W. Clifford, NRC, to J. M. Levine, APS, "Issuance of Amendments for the Palo Verde Nuclear Generating Station Unit No. 1 (TAC No. MA0298), Unit No. 2 (TAC No. MA0299), and Unit No. 3 (TAC No. MA0300)." Reference TS Amendments No. 116, 109 and 88 for Units 1, 2, and 3, respectively.

Dear Sirs:

Subject: Palo Verde Nuclear Generating Station (PVNGS)
Units 1 and 3
Docket Nos. STN 50-528/530
Proposed Amendment to Technical Specification Section 3.8.4, "DC Sources - Operating" Under Exigent Circumstances

Pursuant to 10 CFR 50.90 and 10 CFR 50.91(a)(6), Arizona Public Service Company (APS) requests an amendment to Technical Specification Section 3.8.4, "DC Sources - Operating" under exigent circumstances. As discussed in references 1 and 2, Palo Verde Nuclear Generating Station (PVNGS) is planning to replace the existing Class 1E, 125 volt DC high specific gravity round cell batteries with low specific gravity rectangular cell batteries. The Unit 2 battery replacement was completed during the last refueling outage (Spring 1999). The Units 1 and 3 battery replacements were scheduled to be completed during the next refueling outage for each unit (Fall 1999 and Spring 2000, respectively). Due to problems experienced by the vendor of the low specific gravity rectangular cell batteries, four acceptable batteries are not available and battery replacement will not be able to be completed during the Fall 1999 Unit 1 refueling outage. Therefore, Unit 1 will need to operate for one more cycle with the currently installed Class 1E, 125 volt DC high specific gravity round cell batteries.

A001

Technical Specification 3.8.4, surveillance requirement (SR) 3.8.4.8, requires a performance discharge test or a modified performance discharge test on a 60 month frequency to verify battery capacity. The specified frequency for this surveillance requirement, including the additional time allowed by SR 3.0.2 (1.25 times the interval specified in the frequency), for three of these batteries (channels A, B, and C) will be exceeded starting in December 1999. The battery channel D performance discharge test is not due until the next Unit 1 refueling outage (1R09, Spring 2001). Therefore, the Unit 1 battery channels A, B, and C require a performance discharge test during the current Unit 1 refueling outage, which started October 2, 1999.

The proposed amendment would waive, on a one time basis, the requirement to perform SR 3.8.4.8 for Unit 1 channels A, B, and C. The surveillance requirement would be waived until entry into Mode 4 coming out of the ninth refueling outage for Unit 1 (1R09). At that time, the requirement for the performance discharge test of the high specific gravity round cell batteries would no longer apply since the round cell batteries will have been replaced with low specific gravity rectangular cell batteries. The battery service test in SR 3.8.4.7 will be performed during the current Unit 1 outage (1R08) in lieu of the battery performance discharge test.

The waiver of the requirement to perform a performance discharge test on the Unit 1 batteries was discussed with the NRC staff in a September 22, 1999 phone call. During the phone call the NRC staff concurred with this course of action.

Provided in the enclosure to this letter are the following sections which support the proposed Technical Specification amendment:

A. Explanation of the Exigent Circumstances
B. Description of the Proposed Technical Specification Amendment
C. Purpose of the Technical Specification
D. Safety Analysis of the Proposed Technical Specification Amendment
E. No Significant Hazards Consideration Determination
F. Compensatory Measures to be Taken
G. Environmental Consideration
H. Marked-up Technical Specification Pages
I. Retyped Technical Specification Pages

Even though the maximum extension date for SR 3.8.4.8 does not occur during the current Unit 1 refueling outage, this surveillance requirement can only be performed when the unit is shutdown. Therefore, it is requested that this Exigent Technical Specification

change be issued by October 31,1999 in order to support startup from the current Unit 1 refueling outage.

In accordance with the PVNGS Quality Assurance Program, the Plant Review Board and Offsite Safety Review Committee have reviewed and concurred with this proposed amendment. By copy of this letter this request is being forwarded to the Arizona Radiation Regulatory Agency (ARRA) pursuant to 10 CFR 50.91(b)(1).

The following regulatory commitments are being made to the NRC by this letter.

- The Class 1E, 125 volt DC high specific gravity round cell batteries currently installed in Units 1 and 3 will be replaced with low specific gravity rectangular cell batteries during the next refueling outage for each of these units. The next Unit 1 refueling outage is the ninth refueling outage (1R09, Spring 2001) and the next Unit 3 refueling outage is the eighth refueling outage (3R08, Spring 2000).
- The battery service test, surveillance requirement 3.8.4.7, will be performed on the Unit 1 Battery Channels A, B, and C during the Unit 1 eighth refueling outage (1R08, Fall 1999) in lieu of the performance discharge test.

In accordance with SECY-98-224, "Staff and Industry Activities Pertaining to the Management of Commitments Made by Power Reactor Licensees to the NRC," dated September 28, 1998 and NEI 99-04, "Guidelines for Managing NRC Commitment Changes," Revision 0, dated July 1999, PVNGS considers this to be a regulatory commitment that does not warrant a legally binding requirement or inclusion in the UFSAR or a program subject to a formal regulatory change mechanism. Control of these commitments with PVNGS controlled programs is deemed to be acceptable since these programs include controls for evaluating changes and reporting them to the NRC to maintain the accuracy of the licensing basis.

Should you have any questions, please contact Scott A. Bauer at (623) 393-5978.

Sincerely,

David Mauldin

CDM/SAB/RKR/

Enclosure

STATE OF ARIZONA)
) ss.
COUNTY OF MARICOPA)

I, David Mauldin, represent that I am Vice President Nuclear Engineering and Support, Arizona Public Service Company (APS), that the foregoing document has been signed by me on behalf of APS with full authority to do so, and that to the best of my knowledge and belief, the statements made therein are true and correct.

David Mauldin

David Mauldin

Sworn To Before Me This 8th Day Of October, 1999.

A K Krainik

Notary Public

My Commission Expires

"OFFICIAL SEAL"
A. K. Krainik
Notary Public-Arizona
Maricopa County
My Commission Expires 5/6/2000

cc: E. W. Merschoff (all w/Enclosure)
N. Kalyanam
J. H. Moorman
A. V. Godwin

ENCLOSURE

PROPOSED AMENDMENT TO TECHNICAL SPECIFICATION

SECTION 3.8.4, "DC Sources - Operating"

UNDER EXIGENT CIRCUMSTANCES

A. EXPLANATION OF THE EXIGENT CIRCUMSTANCES

Battery replacement in Unit 1 was scheduled to be completed during the current refueling outage (1R08, Fall 1999). Due to problems experienced by the vendor of the low specific gravity rectangular cell batteries, four acceptable batteries are not available and the planned battery replacement will not be completed as planned. Unit 1 will, therefore, need to operate for one more cycle with the high specific gravity round cell batteries.

Since the high specific gravity round cell batteries will remain in Unit 1 for an additional cycle, a performance discharge test or a modified performance discharge test is required to be performed in accordance with Technical Specification SR 3.8.4.8. Technical Specification SR 3.8.4.8 requires that a performance discharge test be performed to verify battery capacity on a 60 month frequency. The specified frequency for this surveillance requirement including the additional time allowed by SR 3.0.2 (1.25 times the interval specified in the frequency) for three of the Unit 1 batteries (channels A, B, and C) will be exceeded starting in December 1999. The channel D performance discharge test is not due until the next Unit 1 refueling outage (1R09, Spring 2001).

This condition could not be avoided since Palo Verde had planned on replacing the batteries during the current Unit 1 refueling outage and the battery vendor was not able to provide four qualified batteries in time for the outage. As late as September 10, 1999, the vendor was still confident that they could provide the replacement batteries for Unit 1. Only two of the four replacement batteries have been received on site.

B. DESCRIPTION OF THE PROPOSED TECHNICAL SPECIFICATION AMENDMENT REQUEST

Arizona Public Service Company (APS) proposes to waive, on a one time basis, the requirement to perform SR 3.8.4.8 for Unit 1 channels A, B, and C until entry into Mode 4 coming out of the ninth refueling outage for Unit 1 (1R09). The battery service test in surveillance requirement SR 3.8.4.7 will be performed in the current Unit 1 refueling outage (1R08) in lieu of the battery performance discharge test.

C. PURPOSE OF THE TECHNICAL SPECIFICATION

Technical Specification 3.8.4 ensures that at least one train of DC power will be available to mitigate the consequences of an abnormal operating occurrence (AOO) or a postulated design basis accident (DBA). Surveillance Requirement 3.8.4.8 requires a performance discharge test or a modified performance discharge test of the batteries at a 60 month frequency. The test frequency changes to 12 months if the battery shows degradation or has reached 85 percent of the expected life with the capacity less than 100 percent of the manufacturer's rating. The test frequency changes to 24 months if the battery has reached 85 percent of the expected life with the capacity greater than or equal to 100 percent of the manufacturer's rating.

Bases

The Bases for Technical Specification 3.8.4 provides the following description and purpose for the LCO and SR 3.8.4.8 and SR 3.8.4.7.

The Bases for Technical Specification 3.8.4 states that the DC electrical power subsystems, each subsystem consisting of two batteries, battery charger for each battery (the backup battery charger, one per train, may be used to satisfy this requirement), and the corresponding control equipment and interconnecting cabling supplying power to the associated bus within the train are required to be OPERABLE to ensure the availability of the required power to shut down the reactor and maintain it in a safe condition after an AOO or a postulated DBA. Loss of any train of DC electrical power subsystem does not prevent the minimum safety function from being performed.

The initial conditions of DBA and transient analyses in the UFSAR, Chapter 6 and Chapter 15, assume that Engineered Safety Feature (ESF) systems are OPERABLE. The DC electrical power subsystem provides normal and emergency DC electrical power for the DGs, emergency auxiliaries, and control and switching during all MODES of operation. The OPERABILITY of the DC sources is consistent with the initial assumptions of the accident analyses and is based upon meeting the design basis of the unit. This includes maintaining the DC sources OPERABLE during accident conditions in the event of:

a. An assumed loss of all offsite AC power or all onsite AC power; and

b. A worst case single failure.

Bases for SR 3.8.4.8

The surveillance requires that a performance discharge test or a modified performance discharge test of the batteries be performed at a frequency of 60 months. The battery performance discharge test is a test of constant current capacity of a battery, normally done in the "as found" condition, after having been in service, to detect any change in the capacity determined by the acceptance test. The test is intended to determine overall battery degradation due to age and usage.

The acceptance criteria for this surveillance are consistent with IEEE-450, "IEEE Recommended Practice for Maintenance Testing and Replacement of Large Lead Storage Batteries for Generating Stations and Substations, 1980" and IEEE-485, "Recommended Practice for Sizing Large Lead Storage Batteries for Generating Stations and Substations, 1983."

The Surveillance Frequency for this test is normally 60 months. If the battery shows degradation, or if the battery has reached 85% of its expected life and capacity is < 100% of the manufacturer's rating, the surveillance test interval is reduced to 12 months. However, if the battery shows no degradation but has reached 85% of its expected life, the surveillance test interval is only reduced to 24 months for batteries that retain capacity ≥ 100% of the manufacturer's rating. Degradation of high specific gravity round cell batteries is indicated when the battery capacity drops by more than 5% relative to its capacity on the previous performance test, or when it is ≥ 5% below the manufacturer's rating.

Bases for SR 3.8.4.7

A battery service test is a special test of battery capability, as found, to satisfy the design requirements (battery duty cycle) of the DC electrical power system. The discharge rate and test length should correspond to the design duty cycle requirements as specified in the UFSAR section 8.3.2, "DC Power Systems."

Updated Final Safety Analysis Report (UFSAR)

The UFSAR provides the following design basis and discussion for the onsite electrical distribution system, including the batteries.

UFSAR section 8.1.2, "Onsite Power System Description" states that the onsite power system of each unit is divided into two separate systems: the non-Class 1E power system and the Class 1E power system which is divided into two separate load groups. Power is supplied to the auxiliaries at 13.8 kV, 4.16 kV, and 480V levels. The onsite power system includes the Class 1E power system which provides auxiliary AC and DC power for equipment used to shut down the reactor safely following a design basis event. The Class 1E buses of each unit must be energized in order to provide preferred or standby power to the safety-related loads of each unit. The Class 1E power systems are designed in accordance with IEEE 308-1974. A Class 1E DC system provides four channels of 125 V-DC control power for Class 1E switchgear, essential AC power inverters, and other engineered safety feature (ESF) equipment.

The design basis of the onsite power system is described in UFSAR section 8.1.4.2, "Onsite Power System." The UFSAR states that the onsite electric power system for each unit is split into two independent load groups, each with its own offsite and onsite power supplies, buses, transformers, loads, and associated 125 V-DC control power; either load group is independently capable of safely shutting down the unit; the onsite power system includes two redundant Class 1E electric systems for each unit; there is one independent diesel generator provided for each Class 1E AC load group; and the Class 1E electric systems are designed to satisfy the single failure criterion. There is complete independence of onsite electric systems between units. The Class 1E electric systems are designed to satisfy the single failure criterion. For each protection and control channel, one independent 125 V-DC power source and one 120V vital AC power source are provided. Batteries are sized for a minimum of 2 hours of operation without support of a battery charger.

The four Class 1E DC power subsystems are described in UFSAR section 8.3.2, "DC Power Systems." The battery capacity and sizing are discussed in UFSAR section 8.3.2.1.2. In accordance with IEEE Standard 450-1980 battery replacement criteria, initial battery capacity is at least 25% greater than required. UFSAR section 8.3.2.2.1.21, "IEEE 450-1980, Recommended Practice for Maintenance, Testing, Replacement of Large Stationary Type Power Plant and Substation Lead Storage Batteries," describes the implementation of the recommended practices of IEEE 450 for maintenance, testing, and replacement of batteries.

D. SAFETY ANALYSIS OF THE PROPOSED TECHNICAL SPECIFICATION AMENDMENT

The design basis of the onsite power systems, including the batteries, is to provide redundant electric power to equipment required to safely shut down the plant assuming a single failure. Technical Specifications provide surveillance requirements to ensure the operability of required equipment including batteries.

The proposed amendment would waive, on a one time basis, the requirement to perform SR 3.8.4.8 for Unit 1 channels A, B, and C. The surveillance requirement would be waived until entry into MODE 4 coming out of the ninth refueling outage for Unit 1 (1R09). The battery service test in SR 3.8.4.7 will be performed in the current Unit 1 refueling outage (1R08) in lieu of the battery performance discharge test.

As discussed in the Bases for Technical Specification 3.8.4, the purpose of SR 3.8.4.8 is to determine overall battery degradation due to age and usage. This information is then used to determine the expected service life of the battery and when the battery needs to be replaced. The last performance discharge test of the batteries showed that the Unit 1 batteries were capable of supplying over 100 percent of their capacity. The highest design basis load demand for these batteries is less than 50 percent of the actual rated capacity of the batteries. There is over 100 percent margin for these batteries. Therefore, the batteries currently have a high capacity and a large margin above the needed capacity.

Since the battery capacity has remained well over 100 percent for two performance discharge tests for channels A, B, and C and for a third performance discharge test for channel D, and the batteries have been installed for less than eight years, deferring the performance discharge test for 18 months will not result in overestimating the expected service life of the batteries.

Since the batteries will be replaced during the next (ninth) refueling outage, the remaining life of the existing batteries is 18 months. To demonstrate design basis capability and operability for this period, the service test in SR 3.8.4.7, in addition to the other surveillance tests required by Technical Specification 3.8.4 and Technical Specification 3.8.6, “Battery Cell Parameters,” will be performed in lieu of the performance discharge test.

E.. NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

The Commission has provided standards for determining whether a significant hazards consideration exists as stated in 10 CFR 50.92. A proposed amendment to an operating license for a facility involves a no significant hazards consideration if operation of the facility, in accordance with the proposed amendment, would not: (1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) Involve a significant reduction in a margin of safety. A discussion of these standards as they relate to this amendment request follows:

Standard 1 -- Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

No. The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The DC power sources are required to ensure that sufficient power is available to supply safety-related equipment required for safe plant shutdown and the mitigation and control of accident conditions. Since the batteries are not accident initiators and are intended to mitigate the consequences of an accident, the delay of the performance discharge test does not involve a significant increase in the probability of an accident previously evaluated.

The purpose of SR 3.8.4.8 is to determine overall battery degradation due to age and usage. This information is then used to determine the expected service life of the battery and when the battery needs to be replaced. The last performance discharge test of the batteries showed that the Unit 1 batteries were capable of supplying over 100 percent of their rated capacity. The highest design basis load demand for these batteries is less than 50 percent of the actual rated capacity of the batteries. There is over 100 percent margin for these batteries. Therefore, the batteries currently have a high capacity and a large margin above the needed capacity.

Since the battery capacity has remained well over 100 percent for two performance discharge tests for channels A, B, and C and for a third performance discharge test for channel D, and the batteries have been installed for less than eight years, deferring the performance discharge test for 18 months will not result in overestimating the expected service life of the batteries.

Since the batteries will be replaced during the next (ninth) refueling outage the remaining installed life of these batteries is 18 months. To demonstrate design basis capability and operability for this period, the service test in SR 3.8.4.7, in addition to the other

surveillance tests required by Technical Specification 3.8.4 and Technical Specification 3.8.6, "Battery Cell Parameters," will be performed in lieu of the performance discharge test.

The proposed change does not result in any hardware changes or changes to plant operating practices, nor does it affect plant operation. Therefore, since the batteries have high capacity and significant margin and will perform their design function as intended, this change does not involve a significant increase in the consequences of an accident previously evaluated.

Standard 2 --Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

No. The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The DC power sources are required to ensure that sufficient power is available to supply safety-related equipment required for safe plant shutdown and the mitigation and control of accident conditions. The purpose of SR 3.8.4.8 is to determine overall battery degradation due to age and usage. This information is then used to determine the expected service life of the battery and when the battery needs to be replaced. The last performance discharge test of the batteries showed that the Unit 1 batteries were capable of supplying over 100 percent of their rated capacity. The highest design basis load demand for these batteries is less than 50 percent of the actual rated capacity of the batteries. There is over 100 percent margin for these batteries. Therefore, the batteries currently have a high capacity and a large margin above the needed capacity.

Since the battery capacity has remained well over 100 percent for two performance discharge tests for channels A, B, and C and for a third performance discharge test for channel D, and the batteries have been installed for less than eight years, deferring the performance discharge test for 18 months will not result in overestimating the expected service life of the batteries.

Since the batteries will be replaced during the next (ninth) refueling outage the remaining installed life of these batteries is 18 months. To demonstrate design basis capability and operability for this period, the service test in SR 3.8.4.7, in addition to the other surveillance tests required by Technical Specification 3.8.4 and Technical Specification 3.8.6, "Battery Cell Parameters," will be performed in lieu of the performance discharge test.

The proposed change does not change the plant design or configuration (no new or different type of equipment will be installed), or change the method of operation of the plant. The batteries have high capacity and significant margin and will perform their design function as intended. Therefore, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

Standard 3 -- Does the proposed change involve a significant reduction in a margin of safety.

No. The proposed change does not involve a significant reduction in a margin of safety.

The proposed amendment would waive, on a one time basis, the requirement to perform SR 3.8.4.8 for Unit 1 channels A, B, and C. The surveillance requirement would be waived until the next refueling outage for Unit 1 (1R09 Spring 2001). The purpose of the battery performance test required by this surveillance requirement is to determine overall battery degradation due to age and usage. This information is then used to determine the expected service life of the battery and when the battery needs to be replaced. The last performance discharge test of the batteries showed that the Unit 1 batteries were capable of supplying over 100 percent of their capacity. The highest design basis load demand for these batteries is less than 50 percent of the actual rated capacity of the batteries. There is over 100 percent margin for these batteries. Therefore, the batteries currently have a high capacity and a large margin above the needed capacity.

Since the battery capacity has remained well over 100 percent for two performance discharge tests for channels A, B, and C and for a third performance discharge test for channel D, and the batteries have been installed for less than eight years, deferring the performance discharge test for 18 months will not result in overestimating the expected service life of the batteries.

Since the batteries will be replaced during the next (ninth) refueling outage the remaining installed life of these batteries is 18 months. To demonstrate design basis capability and operability for this period, the service test in SR 3.8.4.7, in addition to the other surveillance tests required by Technical Specification 3.8.4 and Technical Specification 3.8.6, "Battery Cell Parameters," will be performed in lieu of the performance discharge test.

The batteries have demonstrated that they have a high capacity, they have been installed for only a short duration of their expected service life, they have a large margin above the needed capacity, and will perform their design function as intended. Therefore, this proposed change does not involve a significant reduction in a margin of safety.

F. COMPENSATORY MEASURES TO BE TAKEN

The battery service test, SR 3.8.4.7, will be performed in the current Unit 1 refueling outage (1R08) in lieu of the performance discharge test, SR 3.8.4.8.

G. ENVIRONMENTAL IMPACT DETERMINATION

The proposed amendment waives the performance discharge test for the Unit 1 channels A, B, and C high specific gravity round cell batteries on a one time basis. APS has determined that the proposed amendment involves no changes in the amount or type of effluent that may be released offsite, and results in no increase in individual or cumulative occupational radiation exposure. As such, operation of PVNGS Units 1, 2, and 3, in accordance with the proposed amendment, does not involve an unreviewed environmental safety question.

H. MARKED-UP TECHNICAL SPECIFICATION PAGES

Page 3.8.4-4

I. RETYPED TECHNICAL SPECIFICATION PAGES

Page 3.8.4-4

9910210001

50-528
3/31/2000

Distribution Sheet

Priority: Normal

From: Stefanie Fountain

Action Recipients:	Copies:	
RidsNrrPMMFields	0	OK
RidsNrrLACJamerson	0	OK
RidsNrrDlpmLpdiv2	0	OK
Internal Recipients:		
RidsRgn4MailCenter	0	OK
RidsOgcRp	0	OK
RidsNrrWpcMail	0	OK
RidsNrrDssaSrxb	0	OK
RidsManager	0	OK
RidsAcrsAcnwMailCenter	0	OK
FILE CENTER 01	1	Paper Copy
ACRS	1	Paper Copy
External Recipients:		
NOAC	1	Paper Copy

Total Copies:	3	

Item: ADAMS Document
Library: ML_ADAMS^HQNTAD01
ID: 003699376:1

Subject:
Palo Verde 1,2 & 3 response to NRC RAI & revised request for amend to Technical Specification 3.3.1,reactor protective sys (RPS) instrumentation - Operating

Body:
ADAMS DISTRIBUTION NOTIFICATION.

Electronic Recipients can RIGHT CLICK and OPEN the first Attachment to View the Document in ADAMS. The Document may also be viewed by searching for Accession Number ML003699376.

APR 03 2000

AA2

A001 - OR Submittal: General Distribution

Docket: 05000528
Docket: 05000529
Docket: 05000530

APS

10 CFR 50.90
10 CFR 50.91

Palo Verde Nuclear Generating Station

David Mauldin
Vice President
Nuclear Engineering and Support

TEL (623) 393-5553
FAX (623) 393-6077

Mail Station 7605
P.O. Box 52034
Phoenix, AZ 85072-2034

102-04425-CDM/SAB/RKR
March 31, 2000

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Station P1-37
Washington, DC 20555-0001

Reference: Letter 102-04293-JML/SAB/RKR, dated May 26, 1999, from J. M. Levine, APS, to NRC, "Request for Amendment to Technical Specification 3.3.1, Reactor Protective System (RPS) Instrumentation - Operating."

Dear Sirs:

Subject: Palo Verde Nuclear Generating Station (PVNGS)
Units 1, 2 and 3
Docket Nos. STN 50-528/529/530
Response to NRC Request for Additional Information and Revised Request for Amendment to Technical Specification 3.3.1, Reactor Protective System (RPS) Instrumentation - Operating

In the referenced letter, Arizona Public Service Company (APS) requested an amendment to Technical Specification 3.3.1, Reactor Protective System (RPS) Instrumentation - Operating, for each Palo Verde Nuclear Generating Station (PVNGS) Unit. In phone calls on August 17, 1999 and October 5, 1999 the NRC and APS staffs discussed the specific methodology used in the reactor coolant pump (RCP) sheared shaft event analysis affected by the amendment request. At the end of the October 5 phone call, the NRC staff requested that APS formally submit the information discussed in the phone calls. Enclosure 2 provides the requested information.

RCP Sheared Shaft Analysis

In the preparation of Enclosure 2, a review of the guidance provided by NRC Office Letter Number 803, "Technical Specification Review Procedures," was performed. APS engineering reviewed the methodology used for the sheared shaft analysis discussed in the amendment request to determine if the methodology was different than the methodology described in Updated Final Safety Analysis (UFSAR) section 15.3.4, "Reactor Coolant Pump Shaft Break with Loss of Offsite Power." This review identified that an assumption used in the dose calculation portion of the analysis had been changed. The UFSAR analysis methodology assumes that the atmospheric dump valves (ADVs) are opened 30 minutes after the reactor coolant pump shaft breaks and that one ADV is stuck open for the duration (90 minutes) of the analyzed event.

A001

003699376

The review found that the supporting dose consequence analysis for the amendment request assumed that the stuck open ADV was manually closed after thirty minutes. This change in dose consequence methodology for the subject event was made by the Palo Verde fuel vendor in 1993. The revised methodology was used to calculate the UFSAR section 15.3.4.3.1.C threshold fuel failure of 25 percent based on the current licensing basis 2-hour site boundary thyroid dose limit of 240 rem.

APS engineering evaluated the assumption change to close the stuck open ADV after 30 minutes under 10 CFR 50.59 and determined the methodology change would require NRC approval. As a result, the analysis was redone in accordance with the UFSAR methodology (i.e., without the stuck open ADV being manually closed). A review of reload analyses since 1993 was performed to ensure that the combination of predicted fuel failures and bounding radial peaking factor due to a reactor coolant pump shaft break would not have resulted in the 2-hour site boundary thyroid dose exceeding the 240 rem limit. For example, based on the methodology described in UFSAR section 15.3.4 and a dose limit of 240 rem, with a bounding radial peaking factor of 2.0, the maximum fuel failure would be approximately 15 percent. Likewise if the radial peaking factor were 1.4, then the maximum fuel failure would be approximately 21.5 percent. For each reload design, the 240 rem 2-hour site boundary thyroid dose was verified based on an assessment of calculated fuel failure and the corresponding radial peaking factor. Section D of enclosure 1 has been revised to describe the licensing basis methodology results. These results do not change the conclusions reached in the original submittal.

Large Steam Line Break Analysis

Subsequent to the submittal of the referenced letter, APS engineering personnel determined that the analysis for the large steam line break inside containment with a concurrent loss of offsite power used a nonconservative assumption. Specifically, work being done for the Unit 2 steam generator replacement and power uprate project identified that the analysis for the large steam line break inside containment with a concurrent loss of offsite power discussed in the referenced letter used the most negative moderator temperature coefficient (MTC). Since this large steam line break analysis assumes a simultaneous loss of offsite power, the event also becomes a loss of reactor coolant system (RCS) flow event. Since loss of RCS flow is a heatup event, the most positive or least negative MTC would result in the worst consequences.

In the evaluation of this incorrect assumption, APS engineering determined that the low reactor coolant flow trip was not required to mitigate a large steam line break event. In 1994, as part of the Palo Verde Reload Process Improvement Project (RPI), a large steam line break event that credited the low reactor coolant flow trip was evaluated due to reliability questions related to the environmental qualification of the RCP low shaft speed trip which normally would provide the protection for this event. This event assumed a simultaneous steam line break inside containment and loss of offsite power. This event was added to the safety analysis basis engineering documents, but was not added to the UFSAR since the large steam line break events described in UFSAR section 15.1.5 continued to be the bounding events

In light of the nonconservative MTC assumption and its affect on the analysis results, the reliability of the RCP low shaft speed trip was reevaluated. The evaluation determined that the RCP low shaft speed trip would reliably provide the protection function as required. This validated the original design basis and safety analysis basis for Palo Verde and negated the need to include a steam line break event that relied on the low reactor coolant flow trip in the design basis and the amendment request. Therefore, the reference to the large steam line break has been removed from the discussion supporting the proposed Technical Specification amendment in enclosure 1.

The problems discussed above were entered into and evaluated in accordance with the Palo Verde corrective action program. A review of other analyses is being performed in light of these issues.

Enclosure 1 has been revised to reflect the changes discussed above. Change bars have been added to identify the changes to enclosure 1 to the referenced letter. The technical specification pages submitted in reference 1 are not affected by these changes. Provided in enclosure 1 to this letter are the following sections which support the proposed Technical Specification amendments:

A. Need for the Amendment
B. Description of the Proposed Technical Specification Amendment
C. Purpose of the Technical Specification
D. Safety Analysis of the Proposed Technical Specification Amendment
E. No Significant Hazards Consideration Determination
F. Environmental Consideration
G. Revised Technical Specification Pages
H. Retyped Technical Specification Pages

In accordance with PVNGS Quality Assurance Program, the Plant Review Board and Offsite Safety Review Committee have reviewed and concurred with this proposed amendment. By copy of this letter this request is being forwarded to the Arizona Radiation Regulatory Agency (ARRA) pursuant to 10 CFR 50.91(b)(1).

APS requests 60 days to implement the approved Technical Specification amendment. The 60 days is required to complete procedure changes, and complete and schedule modification packages for the setpoint changes in all three units.

The proposed amendment modifies the allowable values for the surveillance requirements associated with the steam generator low reactor coolant flow reactor protection system trips. Therefore, APS requests that the following condition be added to the amendment issuance letter: "For surveillance requirements associated with the revised allowable values for functions 12 and 13 in technical specification Table 3.3.1-1, the first performance is due at the end of the first surveillance interval that began on the date the surveillance was last performed prior to the date of implementation of this amendment." This is consistent with the license condition issued with technical specification amendment 117 to the Palo Verde operating license.

No commitments are being made to the NRC by this letter.

Should you have any questions, please contact Scott A. Bauer at (602) 393-5978.

Sincerely,

David Mauldin

CDM/SAB/RKR/mah

Enclosures

cc: E. W. Merschoff (all w/Enclosures)
M. B. Fields
J. H. Moorman
A. V. Godwin

STATE OF ARIZONA)
) ss.
COUNTY OF MARICOPA)

I, David Mauldin, represent that I am Vice President Nuclear Engineering and Support, Arizona Public Service Company (APS), that the foregoing document has been signed by me on behalf of APS with full authority to do so, and that to the best of my knowledge and belief, the statements made therein are true and correct.

David Mauldin

David Mauldin

Sworn To Before Me This 31st Day Of March, 2000.

Nora E. Meador

Notary Public

My Commission Expires

April 6, 2003

"OFFICIAL SEAL"
Nora E. Meador
Notary Public-Arizona
Maricopa County
My Commission Expires 4/6/2003

ENCLOSURE 1

Proposed Amendment to Units 1, 2 and 3
Technical Specification 3.3.1

Proposed Amendment to Units 1, 2 and 3 Technical Specification 3.3.1

A. NEED FOR THE AMENDMENT

The Reactor Coolant Flow, Steam Generator #1-Low and Reactor Coolant Flow, Steam Generator #2-Low reactor protection system trips provide protection against a reactor coolant pump (RCP) sheared shaft event described in the UFSAR Chapter 15 "Accident Analysis." A reactor trip is initiated when the differential pressure across the primary side of either steam generator decreases below a variable setpoint. This variable setpoint normally stays below the indicated differential pressure by a preset value called the Step function, unless limited by a preset maximum decreasing rate determined by the Ramp function, or by a preset minimum value called the Floor function. The Step function is the amount by which the trip setpoint remains below the input signal unless limited by Ramp or Floor functions. The Ramp function is the maximum permitted rate of decrease of the trip setpoint. There are no technical restrictions on the rate of increase of the trip setpoint. The Floor function is the enforced minimum value of the trip setpoint. The combined action of these functions (settings) determines the actual trip setpoint at any moment. The trip setpoint ensures that a reactor trip occurs to prevent violation of the peak linear heat rate (LHR) or departure from nucleate boiling ratio (DNBR) safety limits. There is a separate trip for each steam generator. Pre-trip alarms are also provided.

Since the variable trip setpoint will track the indicated differential pressure upwards very quickly, but is reduced very slowly, normal process noise will keep the setpoint much closer to the mean differential pressure signal than the Step function alone would indicate. This action is conservative with respect to the safety analysis assumptions, but it can result in a trip hazard depending on the magnitude of the noise. A large amount of noise on this process signal is to be expected since the signal is the difference of two pressures taken across a steam generator (a large and complex device) with the high flow rates that exist. Even if overall flow was constant, significant turbulence would still be expected where reactor coolant exits the steam generator.

In 1986 the PVNGS units experienced two plant trips caused by spurious operation of the low reactor coolant flow variable trip. The system vendor, Combustion Engineering determined that the steam generator differential pressure signal includes a random noise component. The source of the noise is believed to be related to the large fluid system acoustic waves propagating throughout the RCS, and randomly initiated by the natural turbulence of flow. The frequency character is a function of the fluid properties and the geometry of the system.

Technical Specification amendments 10 and 5 for Units 1 and 2, respectively, were subsequently issued. The Technical Specification amendments changed the variable trip setpoint (Step, Floor, and Ramp functions) so that process noise could be accommodated without tripping the units. Combustion Engineering also recommended that a small amount of additional filtering be added to the process instrumentation to eliminate spurious trips. Since the filter modification and Technical Specification amendments were implemented there have been no spurious full unit trips associated with the differential pressure signal.

However, since 1992, all three PVNGS units have experienced multiple-channel pre-trip alarms and/or single-channel trips which are attributed to the differential pressure signal. Palo Verde believes that this is a result of the random noise component discussed above. Recent investigation shows that the differential pressure signal periodically rises approximately three psid in six to eight seconds and then immediately drops by as much as six psid in about two seconds. During this sequence the variable setpoint will increase and then hold at the increased setpoint when the process signal drops back down. This often results in the average value of the process signal falling close to the setpoint. PVNGS data indicates that such pressure changes occur every 10 to 20 minutes. Depending on the magnitude of the pressure change, a pre-trip alarm or even a channel trip signal may occur. This process is seen in all three PVNGS units.

Although there is limited data, the frequency of these spurious pretrips appears to be increasing. This is attributed to the slowly increasing differential pressure across the steam generators over time, primarily due to steam generator tube plugging. As the differential pressure increases, the magnitude of the signal excursions due to the random noise component also increases. Therefore, as more steam generator tubes are plugged the potential for a spurious trip increases.

Palo Verde has determined that these excursions are not a result of hardware or instrumentation problems, but are fundamental to the system design. PVNGS Engineering has concluded that a change to the Technical Specification allowable values for the Ramp, Floor, and Step functions (i.e., lowering the effective setpoint) will directly increase the operating range and reduce the trip hazard associated with the random noise component. Additional filtering is not considered an option, since its effect on system response is relatively imprecise. Therefore, any changes to hardware or instrumentation are impractical.

B. DESCRIPTION OF THE PROPOSED TECHNICAL SPECIFICATION AMENDMENT

The allowable values in Technical Specification section 3.3.1, Table 3.3.1-1, Item 12 "Reactor Coolant Flow, Steam Generator #1-Low" and Item 13 "Reactor Coolant Flow, Steam Generator #2-Low," will be changed from ≤0.118 psid/sec. to ≤0.115 psid/sec. for Ramp, from ≥11.7 psid to ≥12.49 psid for Floor, and from ≤10.2 psid to ≤17.2 psid for Step. This change is required to reduce the demonstrated spurious trip hazard associated with this setpoint.

C. PURPOSE OF THE TECHNICAL SPECIFICATION

The low reactor coolant flow trip function is part of the Reactor Protective System (RPS). The RPS initiates a reactor trip to protect against violating the core specified acceptable fuel design limits and breaching the reactor coolant pressure boundary (RCPB) during anticipated operational occurrences (AOOs). Specifically, the low reactor coolant flow trip function ensures that a reactor trip occurs to prevent violation of the peak LHR or DNBR safety limits. The protection and monitoring systems have been designed to ensure safe operation of the reactor.

D. SAFETY ANALYSIS OF THE PROPOSED TECHNICAL SPECIFICATION AMENDMENT

The changes to the allowable values for the low reactor coolant flow trip function settings will provide a larger Step function between the process signal (indicated differential pressure) and the variable trip setpoint, while making the Floor and Ramp more restrictive. The overall effect of these changes will delay the RPS initiated low reactor coolant flow reactor trip. UFSAR Chapter 15 "Accident Analysis," identifies one event that relies on the low reactor coolant flow trip. This event involves a single RCP sheared shaft with a loss of offsite power (UFSAR 15.3.4). Therefore, the single RCP sheared shaft with a loss of offsite power event was reanalyzed to determine the effect of the delayed reactor trip on the analysis described in UFSAR Chapter 15.

In addition, other UFSAR events were evaluated to verify that these events were not affected by this change. The evaluation determined that the results of the bounding analyses for the other UFSAR events were not affected by this change.

RCP Sheared Shaft

The RCP sheared shaft event is a limiting fault event that results in a decrease in reactor coolant flow. Violation of the specified acceptable fuel design limits (SAFDLs) and resulting fuel failure is permissible. The dose consequences are the limiting factor for this event and are limited to the 10 CFR 100 limit (less than 300 Rem thyroid dose and 25 Rem whole body dose at the EAB). UFSAR Section 15.3.4.3 "Analysis of Effects and Consequences," currently states that "The resultant radiological consequences are a 2 hour site boundary thyroid dose of less than 240 Rem. This is within 10CFR100 guidelines."

For decreasing reactor coolant flow events, the major parameter of concern is the minimum hot channel DNBR. This parameter establishes whether a SAFDL has been violated and thus whether fuel damage could be anticipated. Those factors that cause a decrease in local DNBR are:

- increasing reactor coolant system (RCS) temperature,
- decreasing RCS pressure,
- increasing local heat flux (including radial and axial power distribution effects), and
- decreasing RCS flow.

During the first few seconds of the RCP sheared shaft transient, the combination of decreasing RCS flow and increasing RCS temperature results in a decrease in the fuel pins' DNBR. Minimum DNBR is reached at approximately 2 seconds when the RCS flow approaches the flow for three RCPs operating. The decrease in DNBR is reversed as a result of negative reactivity feedback via doppler and void coefficients. Following the reactor trip, a drop in power and heat flux results in rapid recovery of DNBR.

The doppler and void coefficients are primarily responsible for turning DNBR around once three RCP flow has been reached. The time of control rod insertion (i.e., timing of the reactor trip) primarily influences the rate of DNBR recovery and thus relates to DNBR propagation. These two distinct cause-and-effect relationships are fundamental to the sheared shaft event.

The reanalysis of the sheared shaft event using the same methodology as the original analysis, determined that the overall effect of the changes to the allowable values for the low reactor coolant flow trip function was to delay the RPS initiated low reactor coolant flow reactor trip for this event from the current value of approximately 1.2 seconds after event initiation to approximately 2.5 seconds after event initiation. The reanalysis of the sheared shaft event concluded that delaying the reactor trip would result in approximately

the same minimum DNBR as previously analyzed. This is expected since the RCP coastdown characteristics are not being changed. Furthermore, although the time-in-DNB-condition (DNBR propagation) increases from approximately 2.6 seconds to approximately 3.9 seconds as a result of the delay in reactor trip, it remains below the limiting time (4.5 seconds) for the strain limit to be reached. Thus, DNBR propagation is also not a concern.

The reanalysis also evaluated the impact of extending the total trip time from approximately 1.2 seconds to approximately 2.5 seconds and assuming a LOP at approximately 3 seconds after the trip. The reanalysis showed that the minimum DNBR was relatively unchanged. This is expected because at 2 seconds into the event - close to the time of minimum DNBR - flow reaches the flow for three RCPs operating. Therefore, the reanalysis concluded that since minimum DNBR was relatively unchanged, the UFSAR section 15.3.4.3 EAB dose consequences of 240 Rem remains bounding for the sheared shaft event.

The methodology used for the reanalysis is consistent with the original methodology used in the CESSAR and UFSAR. A detailed description of the methodology is included in enclosure 2.

E. NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

The Commission has provided standards for determining whether a significant hazards consideration exists as stated in 10 CFR 50.92. A proposed amendment to an operating license for a facility involves no significant hazards consideration if operation of the facility in accordance with a proposed amendment would not: (1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) Involve a significant reduction in a margin of safety. A discussion of these standards as they relate to this amendment request follows:

Standard 1 -- Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

No. The proposed change will change the Reactor Protection System (RPS) reactor coolant flow trip setpoints. The RPS functions to mitigate the consequences of an accident. The changes to the low reactor coolant flow trip setpoints will reduce or eliminate unnecessary challenges to the RPS. Therefore, the proposed change will not involve a significant increase in the probability of an accident previously evaluated.

These changes will result in an increased time delay for the RPS low reactor coolant flow trip. The reanalysis of the affected Updated Final Safety Analysis Report (UFSAR) Chapter 15 event (UFSAR 15.3.4, Reactor Coolant Pump Shaft Break with Loss of Offsite Power), with the increased time delay, shows that the dose consequences for this event remains bounded by the UFSAR analysis. Therefore, this change does not involve a significant increase in the consequences of an accident previously evaluated.

<u>Standard 2</u> -- Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

No. The proposed change will change the RPS reactor coolant flow trip setpoints. The RPS functions to mitigate the consequences of an accident. The changes to the low reactor coolant flow trip setpoints will reduce or eliminate unnecessary challenges to the RPS. The proposed change only changes the mitigating actions of the RPS, without changing the required function of the RPS. Therefore, the change to the low reactor coolant flow trip setpoints does not create the possibility of a new or different kind of accident from any accident previously evaluated.

<u>Standard 3</u> -- Does the proposed change involve a significant reduction in a margin of safety?

No. The proposed change will change the RPS reactor coolant flow trip setpoints. The reanalysis of the affected UFSAR Chapter 15 event (UFSAR 15.3.4, Reactor Coolant Pump Shaft Break with Loss of Offsite Power), with the revised reactor coolant flow trip setpoints, shows that the minimum departure from nucleate boiling ratio (DNBR) and specified acceptable fuel design limits (SAFDLs) for this event remains bounded by the UFSAR analysis. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the responses to these three criterion, APS has concluded that the proposed amendment involves no significant hazards consideration.

F. ENVIRONMENTAL CONSIDERATION

APS has determined that the proposed amendment involves no changes in the amount or type of effluent that may be released offsite, and results in no increase in individual or cumulative occupational radiation exposure. As described above, the proposed TS amendment involves no significant hazards consideration and, as such, meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9).

G. REVISED TECHNICAL SPECIFICATIONS PAGES

Units 1, 2, and 3: Page 3.3.1-9

H. RETYPED TECHNICAL SPECIFICATION PAGES

Units 1, 2, and 3: Page 3.3.1-9

Table 3.3.1-1 (page 2 of 3)
Reactor Protective System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
8. Steam Generator #1 Level – Low	1,2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.13	≥ 43.7%
9. Steam Generator #2 Level – Low	1,2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.13	≥ 43.7%
10. Steam Generator #1 Level – High	1,2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.13	≤ 91.5%
11. Steam Generator #2 Level – High	1,2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.13	≤ 91.5%
12. Reactor Coolant Flow, Steam Generator #1-Low	1,2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.13	Ramp: ≤ 0.118 psid/sec. Floor: ≥ 11.7 psid Step: ≤ 10.2 psid
13. Reactor Coolant Flow, Steam Generator #2-Low	1,2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.13	Ramp: ≤ 0.118 psid/sec. Floor: ≥ 11.7 psid Step: ≤ 10.2 psid

(continued)

Ramp: ≤ 0.115 psid/sec.
Floor: ≥ 12.49 psid
Step: ≤ 17.2 psid

Table 3.3.1-1 (page 2 of 3)
Reactor Protective System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
8. Steam Generator #1 Level - Low	1,2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.13	≥ 43.7%
9. Steam Generator #2 Level - Low	1,2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.13	≥ 43.7%
10. Steam Generator #1 Level - High	1,2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.13	≤ 91.5%
11. Steam Generator #2 Level - High	1,2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.13	≤ 91.5%
12. Reactor Coolant Flow, Steam Generator #1-Low	1,2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.13	Ramp: ≤ 0.115 psid/sec. Floor: ≥ 12.49 psid Step: ≤ 17.2 psid
13. Reactor Coolant Flow, Steam Generator #2-Low	1,2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.13	Ramp: ≤ 0.115 psid/sec. Floor: ≥ 12.49 psid Step: ≤ 17.2 psid

(continued)

ENCLOSURE 2

Requested Information from August 17, 1999
and October 5, 1999 Phone Conversations

The PVNGS reload analysis methodology, including an overview of the seized rotor/sheared shaft analysis, was reviewed and approved by NRC in the PVNGS Reload Analysis Methodology Report (dated June 14, 1993). The sheared shaft methodology for this Technical Specification (T. S.) submittal is based on the approved methods described in the UFSAR (Refer to 1.6, "Material Incorporated by Reference," 4.3.3, "Analytical Methods," 4.3.4, "References," 4.4.7, "References," and 15.3.4, "Reactor Coolant Pump Shaft Break with Loss of Offsite Power."), but includes a revision to selected input assumptions. This description is provided to clarify previously approved methods and inputs.

I. Sheared Shaft/Seized Rotor (SS/SR) Analysis Methodology

Beginning with Unit 2 Cycle 7, a limited long-term scenario (50 seconds duration) was performed to evaluate the impact of stretch power on the sheared shaft analysis.

Subsequently, in Unit 1 Cycle 7, a short-term scenario for the SS/SR event was performed as part of Reload Process Improvement (RPI) in order to develop a bounding analysis for evaluating fuel failure. The following text details the methodology employed in the bounding RPI analysis, and in the analysis for the requested change to the Technical Specifications.

<u>Initial Power Operating Limit (POL) Calculation:</u>

The selection of initial conditions is based upon the criterion of preserving as much subcooling as possible while maintaining reasonable power operating limit (POL) conditions from the operational standpoint. Preserving initial subcooling minimizes negative reactivity feedback due to voiding associated with loss of flow through the core. Hence, initializing the transient from these conditions maximizes predicted fuel failure. The thermal-hydraulics code CETOP is utilized to calculate the POL conditions. The Core Operating Limit Supervisory System (COLSS) is utilized to preserve initial margin. The calculated POL conditions are listed in Table A-1 to illustrate the selection of initial conditions. Section III details the selection of initial plant conditions.

<u>MTC Tuning:</u>

HERMITE[1] is utilized to perform moderator temperature coefficient (MTC) calculations and determine the soluble boron concentration that corresponds to the limiting MTC value (See Section III).

1. CENPD-188, "HERMITE, a Multi-Dimensional Space-time Kinetics Code for PWR Transients," March 1976 (Proprietary).

Transient Simulations:

Initiating from the POL conditions and boron concentrations noted above, the HERMITE/GENI/CETOP codes are utilized to simulate the transient response of the Sheared Shaft and Seized Rotor events. The main difference between these two transients is the Reactor Coolant Pump (RCP) coastdown curve (independently generated by the COAST code) and the credited Reactor Protection System (RPS) response (e.g. reactor trip and associated delays).

For the Seized Rotor event, a Core Protection Calculator System (CPCS) RCP Shaft Speed trip is credited. The CPCS will generate a signal to the trip breakers 0.71 seconds after the RCP seizes. The trip breakers response time is set at 0.15 seconds. Thus, the trip breakers are credited to open at 0.86 seconds.

For the Sheared Shaft event, the Steam Generator (SG) Low RCS Flow trip, also referred to as the "SG dP Low RCS Flow Trip", is credited to occur at 90% loop flow (corresponds to ~95% RCS flow). The trip is based on differential pressure across the SG primary side as the measurement input[1]. The total response time, including the trip breakers response time of 0.15 seconds, is 1.20 seconds (See Table A-3).

The "SG dP Low RCS Flow Trip" was analyzed with a total trip time changed from 1.20 seconds to 2.50 seconds in order to support the proposed Plant Protection System (PPS) setpoint for the "SG dP Low RCS Flow" reactor trip. The "SG dP Low RCS Flow Trip" has been responsible for a long series of spurious pre-trips and trip signals. Because significant extra time was found in the safety analysis, the option of a Technical Specification change was evaluated which resulted in a RPS total trip time of 2.50 secs, rather than previously credited 1.20 secs (see Tables A-3 and A-4). The reanalysis of the Sheared Shaft Event showed that the results (i.e., the minimum DNBR and transient time in DNB) are in close agreement and acceptable with those established in the RPI Analysis of Record (AOR).

The purpose of the HERMITE/GENI/CETOP transient simulations is to determine plant parameters that correspond to the time of minimum DNBR. Table A-2 lists the bounding time-of-minimum DNBR conditions resulting from the RPI SS/SR simulations. Note that these conditions were based on the Sheared Shaft event, which proved to be slightly more adverse than the Seized Rotor.

Table A-3 lists a typical Sequence of Events for the Short-Term Sheared Shaft event for the RPI AOR and Table A-4 lists a typical sequence of events for the Short-Term Sheared

1. A PVNGS Design I&C calculation translates the T.S. values and hardware settings to support this assumption.

Shaft event for the proposed T. S. change. Note that the transient simulations do not include the Loss of Offsite Power.

Thermal Hydraulic Calculation:

The core thermal-hydraulics code TORC[1] was utilized to calculate DNBR values at various integrated radial peaking factors (Fr), and to account for the effects of the 3-pump flow and time of minimum DNBR conditions.

The result of these 3-Pump TORC cases were a set of Fr versus DNBR data which was subsequently used to calculate fuel failure. Table A-5 lists a set of "typical" data resulting from the 3-Pump TORC cases.

Fuel Failure Calculation:

The calculated fuel failure associated with the reload pin census is based on the Fr versus DNBR data using statistical convolution. Typical values range from 5% to 16%.

In future reload cycles, the fuel failure will be calculated based upon the cycle specific pin census, DNBR statistics, and the Fr versus DNBR data generated in the bounding analysis (Table A-5).

Assessment of DNBR Propagation:

Under severe local conditions, channel blockage due to fuel rod ballooning may potentially impact the heat transfer of adjacent rods sufficiently to produce DNB Propagation. A mechanistic evaluation of fuel clad ballooning in fuel rods experiencing severely degraded heat transfer (i.e. DNB) with internal pressure exceeding RCS pressure was documented. The objective was to provide a means of evaluating the potential of DNB Propagation in accordance with the NRC approved Topical Report CEN-372-P-A, "Fuel Rod Maximum Allowable Gas Pressure," May 1990.

This methodology was utilized to evaluate the fuel clad strain for a wide range of local conditions. The principle result of the evaluation was a determination of the minimum time in DNB which would allow fuel rods to reach the NRC approved maximum strain limit.

In accordance with the general procedure for applying the DNB propagation methodology to Non-LOCA transients, the local conditions were extracted from the HERMITE/GENI/CETOP transient simulation. Since this simulation provides the dynamic thermal hydraulic response to the transient, it was chosen over the detailed, static TORC DNBR calculation (which analyzed the time of minimum DNBR only). Note that the CETOP

1. CENPD-161-P-A, "TORC Code, A Computer Code for Determining the Thermal Margin of a Reactor Core," April, 1986.

calculated DNBR values utilized in the analysis originate from the 4-Pump CETOP Model, not the 3-Pump model. However, local conditions from the 4-Pump model are valid for determining DNB propagation provided the appropriate hot channel flow factor is applied to the core average mass flux.

For the proposed T.S. change, the duration of time in DNB (DNBR < 1.30) experienced by the SS/SR event is less than the minimum time required to reach the NRC approved strain limit. Therefore, the fuel clad will not balloon enough to impede the channel flow sufficiently to propagate DNB to adjacent fuel rods.

Table A-1 Typical Initial POL Conditions

Parameter	Selection Strategy	Input to 4-Pump SAFDL Calculation	4-Pump POL Conditions
Core Average Heat Flux (E6 Btu/hr-ft2)	Maximize	0.224805 (119% ROPM)	0.188912
Core Mass Flux (E6 lbm/hr-ft2)	Maximize	3.0015	
Inlet Temperature (F)	Minimize	548	
RCS Pressure (psia)	Maximize	2242	
Fr[a]	Maximize	2.00	
Axial Power Distribution (ASI)	Maximum Bottom Peaked	+0.19535	
Calculated DNBR	- -	1.300	1.6507

a. A value of 1.70 was used in the dose calculation. This does not impact the POL calculation.

Table A-2 Typical Sheared Shaft Transient Response at Time of Minimum DNBR Condition

Parameter	Initial 4-Pump POL Conditions	Typical Change	RPL Bounding Change	Final Conditions for 3-Pump TORC
Core Average Heat Flux (E6 Btu/hr-ft2)	0.188912	0.976	1.00	0.188912
Core Mass Flux (E6 lbm/hr-ft2)	3.0015	0.746	0.70	2.10105
Inlet Temperature (F)	548	- -	- -	548
RCS Pressure (psia)	2242	- -	- -	2242
Fr	2.00	+0.025	+0.05	2.05
Axial Power Distribution (ASI)	+0.195	+0.002	+0.002	+0.197

Table A-3 Typical Sheared Shaft Short-Term Sequence of Events

(Loss of Offsite Power Not Included in Short-Term scenario)

Time (seconds)	Transient	RPS Response	CETOP DNBR
0.0	Initiate Sheared Shaft Event - POL Conditions		1.650
0.2	RCS Flow ~ 95.4%	Low RCS Flow Trip Set-point Reached (S.G. ΔP)	1.573
0.90	RCS Flow ~ 81.9%	RPS Delays	1.30
1.20	RCS Flow ~ 78.5%	Trip Signal Generated	1.278
1.54	RCS Flow ~ 75.7%	CEDM Hold Coil Delay	1.230
2.00	RCS Flow ~ 74.6%	CEAs Inserted ~ 6.9%	1.210 Minimum DNBR
3.00	RCS Flow ~ 74.3%	CEAs Inserted ~ 33.2%	1.251
3.48	RCS Flow ~ 74.3%	CEAs Inserted ~ 48.0%	1.30
4.00	RCS Flow ~ 74.3%	CEAs Inserted ~ 64.1%	1.502
5.00	RCS Flow ~ 74.3%	CEAs Inserted ~ 86.7%	1.782

Table A-4 Typical Sheared Shaft Short-Term Sequence of Events with Increased RPS and CEDMCS Holding Coil Delay Time
(Loss of Offsite Power Not Included in Short-Term scenario)

Time (seconds)	Transient	RPS Response	CETOP DNBR
0.00	Initiate SS Event	None	1.650
0.90	RCS Flow ~ 81.9%	Fuel begins DNB Condition	~1.34
2.00	RCS Flow ~ 74.6%	RPS Delays, Minimum DNBR for the transient	1.209 Minimum DNBR
2.50	RCS Flow ~74.6%	Trip Signal Generated	1.211
3.10	RCS Flow ~ 74.3%	CEDMCS HC Delay, Rods begin to insert	1.212
3.50	RCS Flow ~ 74.3%	CEAs Inserted ~5%	1.214
5.50	RCS Flow ~ 72.2%	CEAs Inserted ~70%	1.520
6.00	RCS Flow 69.3%	CEAs Inserted ~80%	1.598
7.00	RCS Flow ~ 63.8%	CEAs Inserted ~ 100%	1.720
10.00	RCS Flow ~ 59.0%	None	2.242

Table A-5 Typical Fr versus DNBR Data

Integrated Radial Peak	Minimum DNBR
2.10	0.606
2.00	0.891
1.90	1.128
1.80	1.306
1.70	1.442
1.60	1.590

II. Historical Background

PVNGS Cycle 5

Following the PVNGS-2 Cycle 5 Reload, an expert team review was performed as part of the validation of the reload team concept. This detailed review concluded that the selection

of initial POL conditions should be reevaluated. Specifically, the team determined that selection of a minimal RCS pressure (along with a minimal integrated radial peak) results in voiding in the upper region during the transient which introduces negative reactivity. To minimize this negative void reactivity, the selection of initial conditions was changed to preserve as much subcooling as possible (refer to typical values in Section III of this attachment and to Table A-6).

One result of the change in selection of initial conditions was that the Seized Rotor event was no longer bounding relative to the Sheared Shaft event. The Sheared Shaft yielded slightly higher fuel failure. This revised methodology was first applied to the PVNGS-1 Cycle 5 Reload. The calculated fuel failure for this reload, 5.76%, exceeded the previously reported UFSAR value of 4.5%. A new SS/SR 2-hour thyroid dose calculation was performed to demonstrate that the previously reported UFSAR 2-hour thyroid dose of 240 REM was not exceeded. The calculation documented a 2-hour thyroid dose of less than 200 REM, based on 12% fuel failure.

The improved inlet flow distribution, a new DNBR SAFDL (1.30 versus 1.24), and DNB statistics were incorporated into the PVNGS-3 Cycle 5 Reload. A 3-Pump TORC Model based upon the improved inlet flow distribution was also developed. The result of incorporating these improvements was a significant decrease in calculated fuel failure. The PVNGS-3 Cycle 5 SS/SR Analysis calculated less than 0.85% fuel failure.

PVNGS Cycle 6

Following PVNGS-3 Cycle 5, the Cycle 6 Reloads calculated fuel failures less than 1.0%. The PVNGS-3 Cycle 6 Reload adopted the "No Clad Lift-Off" topical and was required to demonstrate that no DNB Propagation would occur. Future SS/SR analyses would also need to demonstrate no DNB Propagation.

PVNGS Cycle 7

Stretch Power and the associated plant changes (i.e. increased tube plugging, inlet temperature LCO, etc.) only slightly impacted the PVNGS-2 Cycle 7 SS/SR Analysis. For this reload (U2C7), both the short-term and long-term scenarios were analyzed. A revised UFSAR write-up was submitted which documented less than 0.2% fuel failure (hence, less than 240 REM 2-hour site boundary thyroid dose). For RPI (beginning with Unit 1 Cycle 7), a revised source term was also developed (see Section IV).

Post Cycle 7 Reloads

Reload Process Improvement attempted to bound the transient and thermal-hydraulic response of the SS/SR events for future reloads. Specifically, a cycle-independent set of Fr versus DNBR data based upon "bounding" physics and plant data are verified for each

reload during the reload analysis process. This bounding set of Fr versus DNBR data is then used, along with the cycle specific reload pin census file, to calculate fuel failure. Reload fuel failure is then compared against the "bounding" fuel failure which yields the reported dose limits. If necessary, a cycle specific analysis is performed.

Table A-6 PVNGS Sheared Shaft Transient-Selection of Initial Conditions[a]

Case	CAHF	Mass Flux	T-Inlet	Pressure	Fr	AXPD	Transient ROPM
Limiting Axial Power Distribution							
1	Max-Iterated	Maximum	Minimum	Minimum	Minimum	Top	1.2226
2						Bottom	1.2325
Limiting RCS Pressure							
3	Max-Iterated	Minimum	Minimum	Minimum	Minimum	Bottom	1.206
4							1.237
Limiting Core Mass Flux							
5	Max-Iterated	Minimum	Minimum	Maximum	Minimum	Bottom	1.237
6		Maximum					1.2552
Limiting Radial Peak							
A maximum radial peaking factor allows for the use of maximum pressure, minimum temperature, and maximum mass flux when determining the POL conditions. This combination of initial condition minimizes the void reactivity feedback effects and yields lower transient DNBR values. Note that no credit is taken of a 'peakier' pin census file (the pin census associated with higher radial peaks, like those associated with rodded operation, would yield beneficial affects on calculated fuel failure)'							
Original U2C5 Seized Rotor	Maximum	Minimum	Minimum	Minimum	Minimum	Top	1.194 (2.16% fuel failure)
Revised U2C5 Seized Rotor	Maximum	Maximum	Minimum	Maximum	Maximum	Bottom	1.245 (9.28% fuel failure)

a. Blank cells indicate that there is no change in the equivalent parameter between comparative cases.

III. Plant Initial and Other Event Dependent Conditions

Inputs to the SS/SR analysis should be based upon the criterion of preserving as much subcooling as possible while maintaining reasonable POL conditions from the operational standpoint. The "typical" values listed below were obtained from the Bounding RPI SS/SR Analysis.

Licensed Power Limit (LPL):

A maximum rated core power (i.e. 100% T.S. LPL) increases the consequences of the SS/SR events. Since the event is initiated from a POL, power measurement uncertainties (i.e. 2% secondary calorimetric uncertainty) need not be added to the initial power level. Typically, the initial power level is set to 3876 MWt. Note - while not explicitly included in the transient, a 2% power measurement uncertainty is accounted for in the source term used to determine dose.

Core Average Heat Flux (CAHF):

A maximum CAHF increases the consequences of the SS/SR events. CAHF is calculated based upon the power level and number of fuel pins. Due to the displacement of B_4C shims by Erbium fuel rods, the number of fuel rods have been increasing (until all erbium fuel management). This growing trend in the number of fuel rods reduces the calculated CAHF. Typically, a smaller than expected number of fuel rods would be used to calculate CAHF. This quantity of fuel rods would be calculated as: 241 assemblies x 236 pins - 752 non-fuel pins = 56,124 fuel pins.

Primary Coolant Flow:

A maximum primary coolant flow preserves the largest subcooling in the upper region of the core which minimizes negative void reactivity feedback during the transient. Typically, a maximum coolant flow of 115% of design (maximum RCP capacity) would be utilized. This corresponds to a core mass flux of 3.0015 E6 lbm/hr-ft^2 (115% of 2.61 E6 lbm/hr-ft2) and a core mass flow of 182.9 E6 lbm/hr (115% of (164 E6 lbm/hr - 3% core bypass)).

Inlet Temperature:

A minimal inlet temperature preserves the largest subcooling in the upper region of the core which minimizes negative void reactivity feedback during the transient. The inlet temperature is set to the lower LCO minus "monitoring" uncertainty. Typically, the inlet temperature is set to 548 °F (LCO 550 °F - 2.0 °F monitoring uncertainty).

RCS Pressure:

A maximum RCS pressure preserves the largest subcooling in the upper region of the core which minimizes negative void reactivity feedback during the transient. The proper selection of other initial parameters (especially a large integrated radial peak) will ensure a maximum RCS pressure. An iterated pressure between 2025 - 2300 psia was used in the bounding analysis.

Integrated Radial Peak (Fr):

A maximum radial peaking factor preserves the largest subcooling in the upper region of the core, which minimizes negative void reactivity feedback during the transient. Hence, the initial POL conditions are governed by the hot channel which is far different from the average channel. The radial peaking factor is set to the cycle maximum value predicted for the reload. This allows the use of a maximum RCS pressure in the initial POL conditions. Typically, a radial peaking factor of 2.0 is utilized. A value of 1.70 was used in the dose calculation. This does not impact the POL calculation.

Axial Power Distribution:

A bottom peaked axial power distribution delays the power decrease due to scram. In addition, generation of a majority of the power in the region where the core is most subcooled (bottom of the core) reduces the negative void reactivity feedback. The axial power distribution is set to the "limiting" axial shape within the Analysis Range. The Analysis Range is the COLSS LCO ± uncertainty. A typical analysis value is +0.20 ASI at full power.

Moderator Temperature Coefficient:

A more positive (less negative) MTC increases the positive reactivity insertion due to moderator temperature feedback during the flow coastdown. The MTC is set to the most positive value allowed by the COLR. Typically, an MTC value of 0.0 E-4 $\Delta\rho/^{\circ}F$ is utilized.

Fuel Temperature Coefficient:

A more positive (less negative) FTC decreases the negative reactivity insertion due to fuel temperature feedback. The FTCs are tuned to bounding values in the HERMITE Models and verified during the reload analysis process.

Kinetics:

A maximum Beta fraction (β) delays the core power decrease after reactor trip which results in a later DNBR turn-around and a lower flow at the time of minimum DNBR. The kinetics parameters (β, λ, l^*) are tuned to bounding values in the HERMITE Models and verified during the reload analysis process.

Net Scram Worth:

A minimum scram worth delays the core power decrease after reactor trip which results in a later DNBR turn-around and a lower flow at the time of minimum DNBR. The net scram worth (with Worst Rod Stuck Out) calculated based on the T.S. Power Dependent Insertion Limits (PDILs), is tuned to the bounding value in the HERMITE Models and verified during the reload analysis process. Typically, a minimum net scram worth of -7.0%$\Delta\rho$ is utilized.

Fuel Pellet/Clad Gap Conductance (Hgap):

A minimum gap conductance (Hgap) delays the core heat flux decrease after reactor trip which results in a later DNBR turn-around and a lower flow at the time of minimum DNBR. The minimum Hgap values are verified during the reload analysis process.

Flow Coastdown:

A maximum coastdown (1 RCP seized rotor/sheared shaft) minimizes the RCS flow at the time of minimum DNBR. The flow coastdown values are verified during the reload analysis process. Typical coastdowns for the SS/SR events are listed in Table A-7.

Table A-7 Typical RCS Flow Coastdown

Time (seconds)	Sheared Shaft (fraction)	Seized Rotor (fraction)
0.0	1.0	1.0
0.5	0.885	0.839
1.0	0.803	0.773
1.5	0.759	0.753
2.0	0.746	0.749
9.0	0.743	0.748

Reactor Protection System (RPS) Response:

For the Seized Rotor event, a Core Protection Calculator System (CPCS) RCP Shaft Speed trip is credited as soon as the RCP seizes. The CPCS will generate a signal to the trip breakers in 0.71 second (trip breakers response time equals 0.15 second). Thus, the trip breakers are credited to open at 0.86 second.

For the Sheared Shaft event, an "SG dP Low RCS Flow Trip" is credited at 90% loop flow (corresponds to ~95% RCS flow). The trip is based on differential pressure across the SG primary side as the measurement input. The total response time, including the trip breakers response time of 0.15 second, is 1.20 seconds (See Table A-3).

For this submittal, the "SG dP Low RCS Flow Trip" was reanalyzed with the total trip time changed from 1.20 seconds to 2.50 seconds in order to support the proposed Plant Protection System (PPS) setpoint for the "SG dP Low RCS Flow Trip." The "SG dP Low RCS Flow Trip" function has been responsible for a long series of spurious pre-trips and trip signals. Significant extra time was found in the safety analysis space, therefore, the option of a Technical Specification change was evaluated which resulted in a RPS total trip time of 2.50 seconds rather than the previously credited 1.20 seconds (see Tables A-3 and A-4). The reanalysis of the sheared shaft event showed that the fuel failure results are acceptable with respect to those established in the RPI AOR.

CEA Scram Curve:

An extended CEA drop time (scram position versus time) delays the core power decrease after reactor trip which results in a later DNBR turn-around and a lower flow at the time of minimum DNBR. The CEA scram curve is verified during the reload analysis process. A typical scram curve is listed in Table A-8.

Table A-8 Typical Scram Curve

Time (seconds)	Scram Insertion (%)
0.0	0
0.6	0
1.0	5
1.01	10
1.39	20
1.70	30
2.02	40
2.34	50
2.66	60
3.00	70
3.40	80
4.00	90
4.78	100

DNBR Probability Statistics:

The DNBR Probability Statistics are utilized in the fuel failure calculation using approved statistical convolution methods. Typical values used in the convolution technique are listed below.

- DNBR SAFDL[1] = 1.34
- Mean = 1.0605

1. This value has varied during recent reload analyses and its impact on fuel protection was incorporated into the cycle specific CPC addressable constants while leaving the DNBR SAFDL at 1.30.

IV. Dose Calculations for SS/SR Event

A revised dose calculation was generated as part of the Reload Process Improvement program. The calculation was notable in providing an alternate methodology to integrate the activity release based on activity transport and an emergency plant cooldown profile (100 F°/hour) for the unaffected steam generator. The activity release rate was modeled with the equation:

$$\frac{d}{dt}A = i(t) - K \cdot A(t)$$

where A is the activity in the steam generator at time t, i(t) is the rate at which Iodine enters the steam generator, and K is the effective steaming rate including partitioning. The solution is:

$$A(t) = e^{-K \cdot t}\left(\int i(t) \cdot e^{K \cdot t} dt + C\right)$$

and the activity release at any time t is:

$$Release = \int \frac{K \cdot A(t)}{M(t)} dt$$

where M(t) is the steam generator water mass at any time t.

Other inputs and assumptions are summarized below:

- The primary-to-secondary leakage rate is 720 gpd per SG, or 0.5 gpm per SG. These values are based on the Technical Specifications which were in effect at the time the RPI analysis was performed.[1]
- The initial activity concentrations are based on the Technical Specification limits on primary and secondary activity (dose equivalent iodine-131 limits of 1.0 µCi/gm primary and 0.1 µCi/gm secondary).
- The steaming rate is based on the steaming required for plant cooldown, which in turn is dependent on reactor decay heat (including actinides) and stored energy. The activity source term was based on the revised source term developed for RPI, beginning with Unit 1 Cycle 7.
- Offsite power is not available.
- An ADV is assumed to stick open from 1800 seconds for the duration of the event for the 2 hour EAB thyroid dose calculation.
- A partitioning factor of 100 for bulk boiling was used whenever a steam/water interface exists. A plant cooldown rate of 100 F°/hr was used which is more adverse than adminis-

1. These limits bound the current T.S. 3.4.14 limit of 150 gpd for total primary-to-secondary leakage, and 720 gpd is deemed to remain valid for this event.

trative control limit of 75 F°/hr. The RSB SER - CESSAR SYSTEM 80 documents acceptance of the use of an iodine partition factor of 100 for use in the SS/SR event for releases from the SG without the stuck open ADV.[1]

- Iodine is assumed to be released to the atmosphere with a partition coefficient of 1.0 for fluid leaked from primary to the SG with a stuck open ADV.
- Parameters and values used to evaluate dose consequences include:

  a. Power level - 102% Rated Thermal Power.

  b. Source Term developed for U2C7 Stretch Power submittal.

  c. Percent of fuel (bounding value) assumed to experience DNB - 17%.

  d. A bounding radial peaking factor (Fr) of 1.70

  e. RCS activity before event - 1.0 µCi/gm

  f. Secondary system activity before event - 0.1 µCi/gm

  g. ICRP 30 dose conversion factors as stated in PVNGS Technical Specifications.

  h. Atmospheric dispersion factors, X/Q, based on 1986-1991 site specific meteorological data.

The threshhold used for reload evaluation has been set at bounding value of 17% fuel failure, based on the assumptions noted above. This value results in an 2 hour EAB thyroid dose of 240 REM, which is less than the 10CFR Part 100 requirements.

V. Conservatisms

The following conservatisms exist in the RPI SS/SR bounding analysis.

a. The RPI bounding analysis assumed a transient change in core mass flux of 0.70 (fractional) when determining the time of minimum DNBR conditions. This conservative value (relative to the actual change of 0.745) may be used to justify future changes in the flow coastdown or future changes in the RPS setpoints and/or response times.

b. The RPI bounding analysis does not credit the core average heat flux decrease during the transient when determining the time of minimum DNBR conditions. This conservative assumption (relative to the actual 2% decrease) may be used to justify that future changes in physics (i.e. FTCs, MTC, kinetics, etc.) and/or fuel performance (i.e. Hgap) parameters will not affect the results of the bounding analysis.

c. The RCP coastdown curves are based on 90% pump inertia.

1. Docket No. 50-470, October 17, 1981.

50-528/529/530
3/3/2000
See Proposed Change To Tech Specs.

Distribution Sheet

Priority: Normal

From: Stefanie Fountain

Action Recipients:	Copies:	
M Fields	1	Paper Copy

Internal Recipients:		
RidsRgn4MailCenter	0	OK
RidsManager	0	OK
OGC/RP	1	Paper Copy
FILE CENTER 01	1	Paper Copy
ACRS	1	Paper Copy

External Recipients:

Total Copies: 4

Item: ADAMS Package
Library: ML_ADAMS^HQNTAD01
ID: 003689895

Subject:
Palo Verde Nuclear Generating Station, Units 1, 2 and 3 - Application by Public Service Company of New Mexico for Consent to Indirect Transfers of Control and Approval of Li cense Amendments to Reflect Licensee's Name Change

Body:
ADAMS DISTRIBUTION NOTIFICATION.

Electronic Recipients can RIGHT CLICK and OPEN the first Attachment to View the Document in ADAMS. The Document may also be viewed by searching for Accession Number ML003689895.

*** -
AP01 - Proprietary Review Distribution - Pre Operating License & Operating Reactor

AA-2

Docket: 05000528
Docket: 05000529
Docket: 05000530

Public Service Company
of New Mexico
Alvarado Square MS-2824
Albuquerque, NM 87158
505 241-2754
Fax 505 241-2322

Benjamin F. Montoya
Chairman
President & CEO

PNM

March 3, 2000

BY HAND DELIVERY

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555

Re: **Palo Verde Nuclear Generating Station, Units 1, 2 and 3 (Docket Nos. STN 50-528/529/530, Facility Operating License Nos. NPF-41, NPF-51, NPF-74) --Application By Public Service Company of New Mexico for Consent to Indirect Transfers of Control and Approval of License Amendments to Reflect Licensee's Name Change**

Ladies and Gentlemen:

Public Service Company of New Mexico ("PNM") submits this application under Section 184 of the Atomic Energy Act as amended, 22 U.S.C. § 2234, and 10 C.F.R. § 50.80 ("Application") for NRC consent to the indirect transfer of control of PNM's licenses to hold minority interests (both owned and leased) in the Palo Verde Nuclear Generating Station, Units 1, 2 and 3 ("PVNGS" or "the PVNGS Units") to a holding company (Manzano Corporation, hereinafter "Manzano") created to implement the public utility restructuring requirements of the New Mexico Electric Utility Industry Restructuring Act of 1999, SB 428, NMSA 1978, §§ 62-3A-1 through 23 (1999) (the "Restructuring Act").[1] As further described in Attachment B to this Application, the restructuring encompasses the formation of Manzano, the transfer by PNM of its electric and gas transmission and distribution businesses to an affiliated company (with PNM and such affiliated company being under common control by Manzano) and a change in PNM's name to a new name (Manzano Energy Corporation, hereinafter "Manzano Energy").[2] This letter also constitutes

---

[1] A copy of the Restructuring Act is included as Exhibit 1 to Attachment B to this Application.

[2] Upon restructuring, the name "Public Service Company of New Mexico" will be assigned to the new affiliated company (referred to hereinafter as the "UtilityCo").

APO1

0036898990

Change: NRC PDR ltr 1 encl w/o prop

an application under 10 C.F.R. § 50.90 for NRC approval of amendments to the operating licenses for the PVNGS Units to reflect the change in PNM's name.[3]

A description of the proposed transactions, including information provided pursuant to 10 C.F.R. §§ 50.80 and 50.90 and NRC Administrative Letter 96-02, is provided in the Information Submittal in Support of Application for NRC Consent to Proposed Indirect Transfers of Control and Approval of License Amendments to Reflect Name Change ("Information Submittal"), enclosed as Attachment B. That Attachment includes financial information for PNM that is confidential business information. Accordingly, Attachment B includes separate sets of certain proprietary (marked "PR") and non-proprietary (marked "NPR") Exhibits. PNM requests, pursuant to 10 C.F.R. § 2.790, that the proprietary information be withheld from public disclosure. The affidavit of Terry R. Horn, Vice-President and Treasurer of PNM, enclosed as Attachment A hereto, provides the basis for non-disclosure under the regulations.

As the Information Submittal demonstrates, the proposed indirect transfers of control and administrative license amendments will not be inimical to the common defense and security or result in any undue risk to public health and safety. The transfers will also be consistent with all applicable provisions of law, regulations, and orders issued by the Commission.

The proposed restructuring requires approval of other regulatory agencies in addition to consent from the NRC. The NRC Project Manager for PVNGS will be kept informed of the progress made by the other regulatory agencies in granting their approvals.

In order to fully comply with the requirements of the Restructuring Act, PNM respectfully requests that the NRC review this Application on a schedule that will permit the issuance of NRC's consent, and approval of the conforming administrative license amendments, as promptly as possible, and in any event before July 1, 2000. PNM also requests that NRC's consent to the indirect transfers of control of PNM's interests in the PVNGS Units be made effective upon issuance of the NRC order to that effect, and that the

---

[3] A formal license amendment application will be submitted separately by Arizona Public Service Company, as Agent and operator of the PVNGS units.

NRC's consent for the transfer remain effective for a period of one year from the date of the issuance of the order.

If the NRC requires additional information concerning this Application, please contact Terry R. Horn, Vice President and Treasurer, (505) 241-2119. Service upon the applicant of comments, hearing requests, intervention petitions, or other docket entries should be made to PNM's outside counsel, as follows:

Matias F. Travieso-Diaz, Esq.
SHAW PITTMAN
2300 N Street, N.W.
Washington, D.C. 20037
(202) 663-8142
matias.travieso-diaz@shawpittman.com

Very truly yours,

Benjamin F. Montoya
Chairman, President & Chief Executive Officer
Public Service Company of New Mexico
Alvarado Square
Albuquerque, New Mexico 87158

Enclosures

cc

Ellis W. Merschoff, Regional Administrator, NRC Region IV
Mel B. Fields, NRC Project Manager, PVNGS
David E. Corporandy, NRC Resident Inspector, PVNGS
Robert S. Wood, NRC Division of Licensing and Program Mgm't
Steven R. Hom, Esq., NRC Office of General Counsel
James M. Levine, Arizona Public Service Company
(as PVNGS Operating Agent)

ATTACHMENT A

AFFIDAVIT

STATE OF NEW MEXICO)
) ss
COUNTY OF BERNALILLO)

Terry R. Horn, upon being first duly sworn according to law, under oath, deposes and states:

1. I am the Vice President and Treasurer of Public Service Company of New Mexico ("PNM"). I have reviewed the Exhibits to the document entitled "Information Submittal In Support Of Application For NRC Consent To Proposed Indirect Transfers Of Control And Approval Of License Amendments To Reflect Name Change" and believe them to be true and correct.

2. Certain of the said Exhibits, specifically Exhibits 4, 5, 6, 7 and 8, contain commercial and financial information relating to the current and estimated future business operations of PNM and its affiliates that is considered confidential by PNM. Other than its disclosure in the cited Exhibits, the information in question is held in confidence and is generally not disclosed to parties outside PNM. Internal distribution of this information within PNM is limited to essential PNM personnel.

3. Public disclosure of the information contained in the cited Exhibits would be likely to cause substantial harm to PNM's competitive position in the electric power generation market because the information includes PNM's planning estimates of its economic performance, market penetration estimates and other sensitive business data.

4. For these reasons, I believe that the information contained in the cited Exhibits qualifies for withholding from public disclosure pursuant to 10 C.F.R. § 2.790(a)(4) and should be treated as confidential.

Terry R. Horn

SIGNED this 3rd day of MARCH, 2000.

NOTARY PUBLIC IN AND FOR
THE STATE OF NEW MEXICO

My Commission Expires:

5/3/00

12/21/99

Distri25.txt
Distribution Sheet

Priority: Normal

From: Esperanza Lomosbog

Action Recipients:	Copies:	
NRR/DLPM/LPD4-2	1	Not Found
M Fields	1	Not Found
C Jamerson	1	Not Found

Internal Recipients:		
RidsRgn1..MailCenter	1	Not Found
RidsNrrDssaSrxb	1	OK
RidsNrrDssaSplb	1	OK
RidsManager	1	OK
RidsAcrsAcnwMailCenter	1	OK
OGC/RP	1	Not Found
NRR/DSSA/SRXB	1	Not Found
~~NRR/DSSA/SPLB~~	1	Not Found
File Center 01	1	Not Found
ACRS	1	Not Found

External Recipients:		
NRC PDR	1	Not Found
NOAC	1	Not Found

Total Copies:	15

Item: ADAMS Document
Library: ML_ADAMS^HQNTAD01
ID: 003671150

Subject:
Licensee request that when 981216 request for amendment to Technical Specifications (TS) 3.1.8 and TS 3.3.7 is ready for issuance, that 90 days be allowed for implementation of the amendment at PVNGS.

Body:
PDR ADOCK 05000528 P

Docket: 05000528, Notes: STANDARDIZED PLANT

Docket: 05000529, Notes: Standardized plant.

Docket: 05000530, Notes: Standardized plant.

AA-2

APS

Angela K. Krainik
Director
Regulatory Affairs
Palo Verde Nuclear
Generating Station

Tel: 623/393-5421
Fax: 623/393-5442

Mail Station 7636
P.O. Box 52034
Phoenix, AZ 85072-2034

102-04386–AKK/SAB
December 21, 1999

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Station P1-37
Washington, DC 20555-0001

Dear Sirs:

Subject: Palo Verde Nuclear Generating Station (PVNGS)
Units 1, 2 and 3
Docket Nos. STN 50-528/529/530
Request for Amendment to Technical Specifications (TS) 3.8.1, AC Sources – Operating and 3.3.7, Diesel Generator (DG) – Loss of Voltage Start (LOVS)

On December 16, 1998, Arizona Public Service Company (APS) requested an amendment to Technical Specifications (TS) 3.8.1, AC Sources – Operating and 3.3.7, Diesel Generator (DG) – Loss of Voltage Start (LOVS). It is requested that when this amendment is ready for issuance that 90 days be allowed for implementation of the amendment at PVNGS. This time will allow for associated procedures to be changed and for appropriate training to be conducted.

No commitments are being made to the NRC in this letter.

Should you have any questions, please contact Scott A. Bauer at (623) 393-5978.

Sincerely,

Scott A. Bauer for

AKK/SAB/kg

cc: E. W. Merschoff
M. B. Fields
J. H. Moorman

0036711150

A001

PDR ADOCK 05000528

12/16/99

Distribution Sheet

Priority: Normal

From: Elaine Walker

Action Recipients:	Copies:	
NRR/DLPM/LPD4-2	1	Not Found
M Fields	1	Not Found
C Jamerson	1	Not Found

Internal Recipients:		
RidsRgn4 MailCenter	1	Not Found
RidsNrrDssaSrxb	1	OK
RidsNrrDssaSplb	1	OK
RidsManager	1	OK
RidsAcrsAcnwMailCenter	1	OK
OGC/RP	1	Not Found
NRR/DSSA/SRXB	1	Not Found
NRR/DSSA/SPLB	1	Not Found
File Center 01	1	Not Found
ACRS	1	Not Found

External Recipients:		
NRC PDR	1	Not Found
NOAC	1	Not Found

Total Copies: 15

Item: ADAMS Document
Library: ML_ADAMS^HQNTAD01
ID: 993630431

Subject:
Palo Verde Nuclear Generating Station (PVNGS) Units 1, 2, and 3 Docket Nos. STN 50-52
8/529/530 Request for Common Operating License Amendment Numbers

Body:
PDR ADOCK 05000528 P

Docket: 05000528, Notes: STANDARDIZED PLANT

AA2

Docket: 05000529, Notes: Standardized plant.

Docket: 05000530, Notes: Standardized plant.

APS

Angela K. Krainik
Director
Regulatory Affairs
Palo Verde Nuclear
Generating Station

Tel: 623/393-5421
Fax: 623/393-5442

Mail Station 7636
P.O. Box 52034
Phoenix, AZ 85072-2034

102-04384–AKK/SAB/GAM
December 16, 1999

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Station P1-37
Washington, DC 20555-0001

Dear Sirs:

Subject: Palo Verde Nuclear Generating Station (PVNGS)
Units 1, 2, and 3
Docket Nos. STN 50-528/529/530
Request for Common Operating License Amendment Numbers

By letter dated May 20, 1998, the NRC issued Amendment No. 117 to Facility Operating License Nos. NPF-41, NPF-51, and NPF-74 for PVNGS Units 1, 2, and 3, respectively. These amendments, which converted the PVNGS technical specifications to the Improved Technical Specifications, were issued as a single document applicable to all three units with a common amendment number for all three units. Consistent with the issuance of Amendment No. 117, APS requests that all future common operating license amendments issued to the PVNGS units be issued with the same amendment number for all three units. This may require skipping amendment numbers for one or two of the units (as was done when Amendment No. 117 was issued) if previously issued amendment(s) applied to only one or two of the units.

No commitments are being made to the NRC by this letter.

Should you have any questions, please contact Scott A. Bauer at (623) 393-5978.

Sincerely,

Scott A. Bauer for

AKK/SAB/GAM/kg

cc: E. W. Merschoff (NRC Region IV)
M. B. Fields (NRR Project Manager)
J. H. Moorman (NRC Resident Inspector)

A001

PDR ADOCK 05000528 9912300431

DEC 2 8 1999

12/1/99

See Proposed Change To Tech Specs.

Distri91.txt
Distribution Sheet

Priority: Normal

From: Esperanza Lomosbog

Action Recipients:	Copies:	
NRR/DLPM/LPD4-2	1	Not Found
M Fields	1	Not Found
C Jamerson	1	Not Found
Internal Recipients:		
RidsNrrDssaSrxb	1	OK
RidsNrrDssaSplb	1	OK
RidsManager	1	OK
RidsAcrsAcnwMailCenter	1	OK
OGC/RP	1	Not Found
NRR/DSSA/SRXB	1	Not Found
NRR/DSSA/SPLB	1	Not Found
File Center 01	1	Not Found
ACRS	1	Not Found
External Recipients:		
NRC PDR	1	Not Found
NOAC	1	Not Found

Total Copies:	14	

Item: ADAMS Document
Library: ML_ADAMS^HQNTAD01
ID: 993430261

Subject:
Proposed Amendments for Administrative Changes to Update the PVNGS Operating Licenses

Body:
PDR ADOCK 05000528 P

Docket: 05000528, Notes: STANDARDIZED PLANT

Docket: 05000529, Notes: Standardized plant.

Docket: 05000530, Notes: Standardized plant.

AA2

APS

Palo Verde Nuclear Generating Station

Gregg R. Overbeck
Senior Vice President
Nuclear

TEL (623) 393-5148
FAX (623) 393-6077

Mail Station 7602
P.O. Box 52034
Phoenix, AZ 85072-2034

10 CFR 50.90

102-04378 – GRO/AKK/GAM
December 1, 1999

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Station P1-37
Washington, DC 20555-0001

Dear Sirs:

Subject: Palo Verde Nuclear Generating Station (PVNGS)
Units 1, 2, and 3
Docket Nos. STN 50-528/529/530
Proposed Amendments for Administrative Changes to Update the PVNGS Operating Licenses

Arizona Public Service Company (APS) is requesting NRC review and approval of administrative changes to update the PVNGS Operating Licenses. These proposed amendments would (1) delete or update operating license references to outdated administrative information, (2) delete license conditions that were complied with and are no longer applicable to the current operating environment, and (3) delete license conditions that were one-time requirements and have been completed.

Provided in the Enclosure to this letter are the following sections which support the proposed changes:

A. Description of the Proposed Operating License Amendments
B. Purpose of the Operating License Requirements
C. Need for the Operating License Amendments
D. Safety Analysis for the Proposed Operating License Amendments
E. No Significant Hazards Consideration Determination
F. Environmental Consideration
G. Marked-up Operating License Pages

In accordance with the PVNGS quality assurance program, the Plant Review Board and the Offsite Safety Review Committee have reviewed and concurred with this request. By copy of this letter, this submittal is being forwarded to the Arizona Radiation Regulatory Agency (ARRA) pursuant to 10 CFR 50.91(b)(1).

No commitments are being made to the NRC by this letter.

A001

Should you have any questions, please contact Scott A. Bauer of my staff at (623) 393-5978.

Sincerely,

Gregg R. Overbeck

GRO/AKK/SAB/GAM
Enclosure

cc:	E. W. Merschoff	(NRC Region IV)
	M. B. Fields	(NRR Project Manager)
	J. H. Moorman	(NRC Resident Inspector)
	A. V. Godwin	(ARRA)

STATE OF ARIZONA)
) ss.
COUNTY OF MARICOPA)

I, G. R. Overbeck, represent that I am Vice President - Nuclear Production, Arizona Public Service Company (APS), that the foregoing document has been signed by me on behalf of APS with full authority to do so, and that to the best of my knowledge and belief, the statements made therein are true and correct.

Gregg R. Overbeck
G. R. Overbeck

Sworn To Before Me This 1st Day Of December, 1999.

Nora E. Meador
Notary Public

"OFFICIAL SEAL"
Nora E. Meador
Notary Public-Arizona
Maricopa County
My Commission Expires 4/6/2003

My Commission Expires

April 6, 2003

11/24/99

Distribution Sheet

Priority: Normal

From: Elaine Walker

Action Recipients:	Copies:	
NRR/DLPM/LPD4-2	1	Not Found
M Fields	1	Not Found
C Jamerson	1	Not Found
Internal Recipients:		
RidsNrrDssaSrxb	1	OK
RidsNrrDssaSplb	1	OK
RidsManager	1	OK
RidsAcrsAcnwMailCenter	1	OK
OGC/RP	1	Not Found
NRR/DSSA/SRXB	1	Not Found
NRR/DSSA/SPLB	1	Not Found
File Center 01	1	Not Found
ACRS	1	Not Found
External Recipients:		
NRC PDR	1	Not Found
NOAC	1	Not Found

Total Copies:	14	

Item: ADAMS Document
Library: ML_ADAMS^HQNTAD01
ID: 993410258

Subject:
Response to NRC 06/08/1999 request for additional information regarding amend to tec hspec 3.7.15, "Fuel Storage Pool Boron Concentration," 3.7.17, "Spent Fuel Assembly S torage," & 4.3.1, "Criticality."

Body:
PDR ADOCK 05000528 P

Docket: 05000528, Notes: STANDARDIZED PLANT

AA 2

Docket: 05000529, Notes: Standardized plant.

Docket: 05000530, Notes: Standardized plant.

APS

Palo Verde Nuclear Generating Station

David Mauldin
Vice President
Nuclear Engineering and Support

TEL (623) 393-5553
FAX (623) 393-6077

Mail Station 7605
P.O. Box 52034
Phoenix, AZ 85072-2034

10 CFR 50.90
10 CFR 50.91

102-04376 – CDM/SAB/RKR
November 24, 1999

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Station P1-37
Washington, DC 20555-0001

Reference: Letter 102-04294-JML/SAB/RKR, dated June 8, 1999, from J. M. Levine, APS, to NRC, "Request for Amendment to Technical Specification 3.7.15, Fuel Storage Pool Boron Concentration; 3.7.17, Spent Fuel Assembly Storage; and 4.3.1, Criticality."

Dear Sirs:

Subject: Palo Verde Nuclear Generating Station (PVNGS)
Units 1, 2 and 3
Docket Nos. STN 50-528/529/530
Response to NRC Request for Additional Information

In the referenced letter, Arizona Public Service Company (APS) requested an amendment to Technical Specification 3.7.15, Fuel Storage Pool Boron Concentration; 3.7.17, Spent Fuel Assembly Storage; and 4.3.1, Criticality, for each Palo Verde Nuclear Generating Station (PVNGS) Unit. In a phone call on November 18, 1999, the NRC staff requested additional information required to complete the review of the amendment request. The discussion identified changes to the dilution analysis provided in the referenced letter in enclosure 4, "Palo Verde Nuclear Generating Station Spent Fuel Pool Boron Dilution Analysis (13-NS-C44)." Attachment 1 provides responses to the NRC staff questions. Attachment 2 provides replacement pages for enclosure 4 to the referenced letter.

A001

No commitments are being made to the NRC by this letter.

Should you have any questions, please contact Scott A. Bauer at (623) 393-5978.

Sincerely,

Michael J. Winsor
for D. Mauldin

CDM/SAB/RKR/mah

Attachments

cc: E. W. Merschoff (all w/Attachment)
M. B. Fields
J. H. Moorman
A. V. Godwin

ATTACHMENT 1

Response to NRC Request for Additional Information

Response to NRC Request for Additional Information

1. NRC Question: Page 24 of the referenced letter, enclosure 4, "Palo Verde Nuclear Generating Station Spent Fuel Pool Boron Dilution Analysis (13-NS-C44)," states that the spent fuel pool (SFP) volume is conservatively assumed to be 340,000 gallons. On pages 14, 25, and 27 the volume is given as 320,000 gallons. What is the correct volume?

   APS Response: 340,000 gallons is the nominal SFP volume. 320,000 gallons is the minimum operational volume. The volume of 320,000 gallons was used in the analysis in order to obtain the minimum volume of water needed to dilute the SFP to 900 ppm. The value on page 24 has been revised to 320,000 gallons and the revised page is included in attachment 2.

2. NRC Question: The volumes of non-borated water required to dilute the SFP from 2150 to 900 ppm boron are listed on page 25 for both normal conditions with the pool volume at the minimum level of 320,000 gallons and for post seismic conditions when the pool volume could be as low as 206,000 gallons. Are the listed volumes correct?

   APS Response: The volumes provided on page 25 are the total volumes of water, borated and non-borated that would result if non-borated water was added to the initial volume of water in each case until the 900 ppm boron concentration was reached. The volume of water that had to be added the pool to achieve this dilution is the difference between these values and the initial volume. The volume of water added should have been provided in both cases. The discussion on page 25 has been corrected and the revised page is included in attachment 2. This change also resulted in a change to page 29. The revised page 29 is also included in attachment 2.

3. <u>NRC Question:</u> In Table 4 on page 21, what does the NA mean?

<u>APS Response:</u> The "NA" indicates that there are no scenarios that would result in a discharge of non-borated water into the SFP. These system piping/components are located below the SFP normal water level and/or are isolated by manual valves. A note has been added to Table 4, resulting in a change to page 21. The revised page 21 is included in attachment 2.

4. <u>NRC Question:</u> Are the fire protection water tanks at a higher elevation than the SFPs? If they are, does this result in the fire protection system becoming a dilution source following a seismic event due to gravity draining of the fire protection water tanks?

<u>APS Response:</u> The top of the SFPs in all three units is lower than the maximum level in the fire protection water tanks. Due to site terrain, the Unit 3 SFP is at the lowest elevation. The top of the Unit 3 SFP is approximately 9 feet below the maximum water level in the fire protection water tanks. In order to get dilution flow to the SFP from the fire protection water tanks, a line break would have to occur in the fire protection riser on the 140 foot elevation of the spent fuel building in the space between the floor and 9 feet above the floor level with the worst case break being right at the floor level. This scenario would also depend on there being no other breaks in the fire protection lines between the 140 foot elevation and the fire protection water tanks. PVNGS does not consider this to be a credible scenario. Therefore, this scenario is not part of the PVNGS design basis.

Nonetheless, the volume of water in the top nine feet of the fire protection water tanks is approximately 2.4 E+5 gallons. As discussed in section 7.0 of the dilution analysis it would take 2.8 E+5 gallons of non-borated water to dilute the SFP from 2150 ppm to 900 ppm boron following a seismic event. Therefore, gravity drainage from the fire protection water tanks to the Unit 3 SFP could not deliver sufficient water to dilute the SFP to 900 ppm.

ATTACHMENT 2

Replacement Pages for the Referenced Letter, Enclosure 4, "Palo Verde Nuclear Generating Station Spent Fuel Pool Boron Dilution Analysis (13-NS-C44).

Replace pages 21, and 24 through 29 in reference 1, enclosure 4, "Palo Verde Nuclear Generating Station Spent Fuel Pool Boron Dilution Analysis (13-NS-C44)," with the attached replacement pages 21, and 24 through 29. Section 7.0 "Spent Fuel Pool Dilution Evaluation" is being replaced in its entirety. The changes include the changes discussed in Attachment 1 and editorial changes that do not change the meaning or conclusions.

Table 4 - Summary of Dilution Sources for Spent Fuel Pool

	Normal	Event Related	Pipe Break
Condensate transfer system	100 (gpm)	NA *	NA *
Liquid radwaste recycle monitor tank	150 (gpm)	NA*	NA*
Fire Protection Supply Lines	NA*	45 - 500[1] (gpm) (Fires)	85 (gpm)
Demineralized Water System			
- Decon / Utility station	NA*	50 (gpm) (Operator error)	23 (gpm)
- Spent Fuel Pool Clean up System resin change out	850 Gal / change	NA*	NA*
Domestic water Utility Station	NA*	50 (gpm) (Operator error)	4 (gpm)
Nuclear cooling / Essential cooling	1 (gpm)	85 (gpm) (tube rupture)	NA*
De-boration by pool clean up Ion Exchange	7 – 10 PPM / change	NA*	NA*

* Note: "NA" indicates that there are no scenarios that would result in discharge of non-borated water into the SFP. These system piping / components are located below the SFP normal water level and / or are isolated by manual valves.

6.0 INSTRUMENTATION

6.1 Loss of Offsite Power and Impact on the Spent Fuel Pool Level Instrumentation

Instrumentation is provided which monitors the temperature and the water level in both the refueling and spent fuel pools. Spent fuel pool alarms are annunciated locally and in the control room. Refueling pool computer alarms are annunciated in the control room only. Additional instrumentation, monitored locally, is provided to check inlet and outlet temperatures on the heat exchangers, and to determine the pressure of the cooling pump discharge. The power source for instrumentation is provided by normal 120 VAC (Panel E- NNN–D015). Automatic transfer to back up reliable power source is available. Additionally, in the event that the distribution panel E-NNN-D15 is lost due to shedding of MCC E-NHN-M19 in a loss of all power event, provisions are made to enable operators to load key instruments on emergency diesels (Ref. 55 through 58). Procedures 40AO-9ZZ15 and 40AO-9ZZ12 provide instruction for the implementation of the design and to ensure that loss of annunciators, and degraded electrical power conditions are addressed (Ref. 59 and 60).

[1] 500 gpm is the maximum flow of a fire hose station. 45 gpm, is estimated flow for sprinkler system.

E-NNN-D11 & D12 similar to the BAMP and RWMP instrumentation described previously (Ref. 68 & 70).

Flow indicators SIA-FI-338 & SIB-FI-348 provide Control Room flow indication for the CS pump "A" and "B" Trains respectively (Ref. 75). These instrument loops are designed to Quality Class Q and Seismic Category I requirements (Ref. 75). The Class 1E Instrumentation Distribution Panels, PNA-D25 & PNB-D26 respectively power these flow-monitoring loops (Ref. 62 & 74). These indicators are located on Control Boards B02E and B02D in the Control Room.

The pump discharge pressure is available in the Control Room for the LPSI pump makeup flow path to the SFP. The indicators SIN-PI-306 & PI-307, described above for the CS pumps, provide pressure indication for this alignment also. There is no flow indication either locally or in the Control Room for the LPSI pump makeup flow path to the SFP.

6.3 Loss of Offsite Power and Impact on Other Systems Instrumentation

Instrumentation on all non-quality systems such as nuclear cooling water, liquid radwaste, Domestic water, and Demineralized water could be lost during total loss of power. These systems instrumentation are not critical during mitigation of boron dilution event.

Therefore essential instrumentation on this system could be available. Instrumentation for Condensate Storage and Transfer System and Essential Cooling Water system are available during total loss of power and backup by a 1E power source.

7.0 SPENT FUEL POOL DILUTION EVALUATION

For the purposes of evaluating spent fuel pool dilution times and volumes, the total pool volume available for dilution is conservatively assumed to be 320,000 gallons (Ref. 76), for normal plant operation (operator error), fire event and pipe breaks. The volume of the transfer canal and cask loading pit are 34,000 and 82,000 gallons respectively. During normal plant operation the cask-loading pit is filled with borated water and the transfer canal may be filled. They are both isolated from the main pool by pneumatic sealed gates. The normal configuration of the spent fuel pool is to have all of the gates in place. In this configuration, any dilution of the pool, the transfer canal or cask loading pit is assumed to affect only the body of water that non-borated water is introduced in.

During a seismic event, the total pool volume available for dilution is conservatively assumed to be 206,000 gallons (Ref. 77). During this event, due to the failure of non-quality pool gates and pool clean up system, all bodies of water would reach equilibrium at elevation 133ft. The initial conditions for the event are assumed to be as follows; spent fuel pool water level is at 137 ft, the cask loading pit is only filled to the 133 ft elevation, and the transfer canal is empty prior to the event. In this configuration, any dilution in the spent fuel pool, the transfer canal or cask loading pit is assumed to affect all bodies of water that non-borated water is introduced.

The boron concentration currently maintained in the spent fuel pool is 4,000 – 4,400 ppm for plant Modes 1 through 6. In addition, the boron concentration is maintained at greater than

2,150 ppm when irradiated fuel is stored in the spent fuel pool. Based on newly proposed Technical Specification, LCO 3.7.15, the minimum allowable soluble boron concentration required to maintain the spent fuel boron concentration at $K_{eff} < 0.95$, including uncertainties and burnup, with a 95% probability at a confidence level (95/95) is 900 ppm.

For the purposes of evaluating dilution times and volumes, the initial spent fuel pool boron concentration is assumed to be at the current Technical Specification Limit of 2,150 ppm. The evaluations are based on the spent fuel pool boron concentration being diluted from 2,150 ppm to 900 ppm. To dilute the spent fuel pool volume of 320,000 gallons during normal operation, from 2,150 ppm to 900 ppm, it would conservatively require 4.4E+5 gallons of non-borated water. Following a seismic event, the volume of water in the pool would be approximately 206,000 gallons. To dilute the spent fuel pool from 2,150 ppm to 900 ppm, it would conservatively require 2.84E+5 gallons of non-borated water.

This analysis assumes thorough mixing of all the non-borated water added to the spent fuel pool. It is likely, with cooling flow and convection from the spent fuel decay heat, that thorough mixing would occur. However, if mixing were not adequate, it would be conceivable that a localized pocket of non-borated water could form somewhere in the spent fuel pool. This possibility is addressed by the criticality calculation, which shows that the spent fuel K_{eff} will be less than 1.0 on a 95/95 basis with the spent fuel pool filled with non-borated water. Thus, even if a pocket of non-borated water formed in the spent fuel pool, K_{eff} would not be expected to exceed 1.0 anywhere in the pool.

The time to dilute ($T_{dilution}$) depends on the initial volume of the pool and the postulated rate of dilution. The dilution volumes and times for the dilution scenario discussed in Sections 3.2 and 3.3 are calculated based on the following equation:

$$T_{dilution} = (V/Q) * \text{Ln}\,(C_o/C_{end}) \qquad \text{(Equation 1)}$$

Where:

C_o = the boron concentration of the pool volume at the beginning of the event (2,150 ppm)

C_{end} = the boron endpoint concentration (900 ppm)

Q = dilution rate (gallons of water/minute)

V = volume (gallons) of spent fuel pool.

7.1 Spent Fuel Dilution During Seismic Event (Power Operation / Refueling)

7.1.1 Description of Event

In the hypothetical event of a SSE with loss of offsite power, the spent fuel pool water level could drop due to the failure of non-seismic pool clean up system and gates. The minimum elevation reached during this event is adequate to maintain pool cooling and provide more than 10-ft. of shielding above the fuel assemblies. Post seismic, the spent fuel pool, cask loading pit and transfer canal reach equilibrium. The design leakage due to evaporation is less than 2 gpm.

Liner and boundary valve leakage is limited to less than 5 gpm during plant modes 1 through 5. During modes 5 and 6 the design leakage from the pool boundary is limited to less than 30 gpm when valve PCN–V118 is in the closed position and refueling pool level is less than 133 ft. The normal source of make up during this mode of operation is the gravity feed from the refueling tank as discussed in section 4.1 of this report.

7.1.2 Calculation of Boron Dilution Times and Volumes

Based on the design bases of PVNGS (Ref. 78): non-safety systems are assumed to be not available to mitigate accident conditions and are not considered to operate in a manner which increases the severity of the accident. Therefore, it is assumed all non–seismic systems such as domestic, demineralized water, fire protection, condensate transfer, and liquid radwaste will fail. These systems would not contribute to the event since the motive force of the supply sources (i.e. supply pumps and tanks [2]) would fail as result of seismic event. As described in section 5 of this report, the lines providing water to the utility / decontamination and fire hose stations terminate at elevation 144 ft. There would be a minimum amount of water spilled due to failure of seismic IX piping. The volume of water spilled during the event from non-borated sources is estimated to be less than 15 gallons (100 ft. of 1 ½ inch piping).

The makeup source for this event is the Refueling Water Storage Tank. As stated in section 7.1.1, the gravity feed would be able to makeup for system and boundary leakage in all modes of operation. It should be noted that during modes 5-6 operation when refueling pool level is less than 133 ft., there will be more than 400,000 gal of borated water available to compensate for the design leakage.

7.1.3 Evaluation of Boron Dilution Event

It can be concluded from the above discussion that a seismic event would not result in a boron dilution event. The addition of 15 gallons of non-borated water in 206,000 gallons of borated inventory at a minimum of 2150-ppm is insignificant. Additional safety margin has been added to the design by administrative requirements of the procedure 4XA1-XRK7C. The control room operators would identify and isolate sources of leakage in a timely manner a once valid seismic alarm is received.

7.2 Moderate Energy Line Break

7.2.1 Description of Event

The plant design for protection against piping failures in the fuel building is reviewed to assure that such failures would not cause de-boration of the spent

[2] Note: All supply lines are isolated by manual root valves in the fuel building. The CST and liquid radwaste system are designed to seismic category II, and would retain their pressure boundary integrity up to OBE event.

fuel pool and to assure that the spent fuel pool boron concentration will remain above the minimum required. The review includes moderate energy fluid system piping located within the fuel building. It is assumed for this analysis that a hypothetical random pipe break would occur within the fuel building.

The pipe break is the initiating event, and it does not coincide with any other design bases events. As discussed in section 5 of this report all lines within the fuel building can be classified as moderate energy lines. Per ANS/ANSI 58.2–1980, through–wall crack opening shall be assumed as a circular orifice of cross-section flow area equal to that of rectangle with length of ½ the pipe inside diameter and width of ½ the pipe thickness. Table 4 shows a summary of calculated leakage for different systems. The hypothetical pipe break does not result in loss of equipment power or flooding of SSC. Therefore, no loss of offsite power is required for this event.

7.2.2 Calculation of Boron Dilution Times and Volumes

As described in section 5 and shown on Table 3, the most limiting pipe leakage is due to a failure in the fire protection piping. The fire protection hose station is located at elevation 140 ft. (pool operating deck). It is conservatively assumed that all 85-gpm discharged from the crack would be added to the pool (due to the geometry of the fuel building, a large portion of the leakage will spill on to lower floors). Since the initiating condition would happen during normal operation, the total volume of the spent fuel pool is available for dilution.

Therefore based on equation one:

- C_o = the boron concentration of the pool volume at the beginning of the event = 2150 ppm (assuming smallest pool concentration)
- C_{end} = the boron endpoint concentration = 900 ppm
- Q = dilution rate (gallons of water/minute) = 85 gpm
- V = volume (gallons) of spent fuel pool = 320,000 gals

Using equation one, the time to dilution time is calculated to be approximately 55 hrs. This would provide adequate time for operators to respond to the event. The pool Hi-Hi level alarm is currently set at 138' 4". Assuming the initiation event accurs at the time the pool has minimum possible Technical Specification elevation, i.e. 137' 4", the time for the operator to get an indication in control room is approximate 1.1 hr and the pool concentration at time of alarm would be greater than 2,050 ppm. Isolation of the header is easily achievable since the isolation valve is located outside the fuel building (PIV-014).

7.2.3 Evaluation of Boron Dilution Event

It can be concluded, from the above discussion, that a pipe break event would not result in a boron dilution event which could reduce the margin of safety. The addition of 85 gpm non-borated water to the spent fuel pool is at a flow rate that can be identified and isolated by control room operators in a reasonable time period.

7.3 Normal Operation, Operator Error and Fire

The scenarios evaluated in this section can be categorized into three categories;

1. Normal operational occurrences which include normal replacement of ion exchanges in the pool cooling system, normal evaporation makeup, and normal inter-system leakages (ruptures),

2. Operator error which include misuse of decontamination / utility station and

3. Fire in the fuel building

   a. Detail discussions of each system and normal parameters associated are provided in section 5. During normal operation the pool clean up system ion exchangers need to be replenished. This process would result in de-boration of the fuel pool. Since the normal boron concentration in the spent fuel pool is usually greater then 4,000 ppm. The effect of de-boration due to introduction of 850 gallons of demineralized water or de-boration due to ion-exchanger (7-10 ppm) would be insignificant.

      The inter-system leakage from nuclear cooling / essential cooling system into the pool cooling system adds an insignificant amount of non-borated water into the pool. At a rate of 1 gpm, the leakage would be almost undetectable and could be masked by evaporation from the spent fuel pool. However, control room operators would observe excessive demand on makeup water required for the nuclear cooling / essential cooling water system. In event of inter-system tube rupture, the flow initially would be approximately 85 gpm; however, the closed loop-cooling system would lose pressure rapidly and the control room would have an indication due to Low-Low level alarms produced by surge tank levels. Such event is bounded by the evaluation performed for pipe break (see section 7.2).

   b. It is possible to assume that a single operator using the decontamination station / utility station would mistakenly discharge a portion of hose flow into the spent fuel pool during maintenance activities or decontamination of casks or adjacent areas around the pool. The flow from these stations are limited to approximately 50 gpm. At this flow rate, the discharge from the hose is less than in the case evaluated for a pipe break, therefore, it would be bounded by the evaluation in section 7.2. However, it is highly improbable that the flow would be continuous for a long duration of time. Therefore, the consequences of such an event are much less than a

moderate energy line break event.

c. The probability of fire at the fuel building at elevation 140 ft. is very small due to the lack of combustible loading. In case of a fire at this elevation, the fire team has to utilize the fire hose station HS 100, located at the southeast corner of the operating floor. The maximum flow for this hose station is 500 gpm and assuming that the fire can be terminated within an hour, the total water discharge to elevation 140-ft. would be 30,000 gal. If all flow is discharged directly into the pool, the pool concentration would be reduced from 2,150 to 1,900 ppm (note: normal concentration for spent fuel pool is greater than 4,000 ppm). Because of the limited flow into the spent fuel pool enclosure, and because of the awareness of control room and fire crews, the discharge to the pool would be terminated long before the spent fuel pool boron concentration could be reduced to 900 ppm.

7.4 Conclusions

It is concluded that an unplanned or inadvertent event, which would result in the dilution of the spent fuel pool boron concentration from 2,150 ppm to 900 ppm is not a credible event. This conclusion is based on the following:

1. In order to dilute the spent fuel pool ($K_{eff} > 0.95$), a substantial amount of water is needed. Most sources of water at PVNGS site would be exhausted and it would take continued manual actions on the part of plant personnel to assure that enough water would be available to support such a dilution.

2. Since such a large water volume turnover is required, a spent fuel pool dilution event would be readily detected by plant personnel via alarms, sump flooding in the fuel and / or auxiliary building, or by normal operator rounds through the spent fuel pool area.

3. Evaluations indicate that based on the design flow rates of non-borated water normally available to the spent fuel pool, sufficient time is available to detect and respond to such an event.

It should be noted that this boron dilution evaluation was conducted by evaluating the time and water volumes required for diluting the spent fuel pool from 2,150 ppm to 900 ppm. The 900-ppm end point was utilized to ensure that K_{eff} for the spent fuel would remain less than or equal to 0.95. However, the PVNGS technical requirements manual requires the spent fuel pool to be maintained at a concentration of 4,000 to 4,400 ppm boron. This requirement would provide additional design margin, which is not credited for in this study. In conclusion, the design and administrative procedures in place at PVNGS provide solid design bases for crediting soluble boron in the spent fuel pool and the plant design provides ample margin against an inadvertent dilution event.

DEC 7 1999

11/19/99

Distribution Sheet

Priority: Normal

From: Elaine Walker

Action Recipients:	Copies:	
M Fields	1	Not Found

Internal Recipients:		
OGC/RP	1	Not Found
OE	1	Not Found
NRR/DSSA/SPLB	1	Not Found
NRR/DLPM/LPD3	1	Not Found
J Segala	1	Not Found
FILE CENTER 01	1	Not Found
ACRS	1	Not Found

External Recipients:		
NRC PDR	1	Not Found

Total Copies: 9

Item: ADAMS Document
Library: ML_ADAMS^HQNTAD01
ID: 993340447

Subject:
Application for amend to licenses NPF-41, NPF-51 & NPF-74, proposing changes to TS 5 .5.11.c, "Ventillation Filter Testing Program (VFTP)." Proposed technical specifications, enclosed.

Body:
PDR ADOCK 05000528 P

Docket: 05000528, Notes: STANDARDIZED PLANT

Docket: 05000529, Notes: Standardized plant.

Docket: 05000530, Notes: Standardized plant.

AA2

APS

Palo Verde Nuclear
Generating Station

Gregg R. Overbeck
Senior Vice President
Nuclear

TEL (623) 393-5148
FAX (623) 393-6077

Mail Station 7602
P.O. Box 52034
Phoenix, AZ 85072-2034

102-04373 - GRO/SAB/JAP
November 19, 1999

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Reference: 1. NRC Generic Letter 99-02: Laboratory Testing of Nuclear-Grade Activated Charcoal, dated June 3, 1999.

Dear Sirs:

Subject: Palo Verde Nuclear Generating Station (PVNGS)
Units 1, 2 and 3
Docket Nos. STN 50-528/529/530
Proposed Amendment to Technical Specification 5.5.11 for Laboratory Testing of Nuclear-Grade Activated Charcoal and Response to NRC Generic Letter 99-02

In Reference 1, the NRC determined that testing nuclear-grade activated charcoal to standards other than American Society for Testing and Materials (ASTM) D3803-1989, "Standard Test Method for Nuclear-Grade Activated Carbon," does not provide assurance for complying with the current licensing basis as it relates to the dose limits of General Design Criterion (GDC) 19 of Appendix A to Part 50 of Title 10 of the Code of Federal Regulations (10 CFR 50) and Subpart A of 10 CFR 100. The NRC requested that licensees determine whether their technical specifications (TS) reference ASTM D3803-1989 for charcoal filter laboratory testing. Addressees whose TS do not reference ASTM D3803-1989 were requested to amend their TS to reference ASTM D3803-1989 or propose an alternative test protocol and provide the information discussed in the requested actions.

Consistent with the guidance in GL 99-02, Arizona Public Service Company (APS) requests an amendment to Technical Specification 5.5.11.c, Ventilation Filter Testing Program (VFTP). This proposed amendment will require the testing of the ESF systems charcoal adsorber in accordance with the methodology and tolerances of ASTM D3803-1989, "Standard Test Method for Nuclear-Grade Activated Carbon."

A081

PDR ADOCK 05000528

9933 40447

Provided in Enclosure 1 to this letter are the following sections that support the proposed Technical Specification amendment:

A. Description of the Proposed Technical Specification Amendment
B. Purpose of the Technical Specification
C. Need for the Technical Specification Amendment
D. Safety Analysis for the Proposed Technical Specification Amendment
E. No Significant Hazards Consideration Determination
F. Environmental Impact Determination
G. Marked-up Technical Specification Page
H. Retyped Technical Specification Page

Generic Letter 99-02 also contained several requested actions for licensee response. Enclosure 2 to this letter provides APS' responses to the requested actions.

In accordance with the PVNGS Quality Assurance Program, the Plant Review Board and Offsite Safety Review Committee have reviewed and concurred with this proposed amendment. By copy of this letter, this request is being forwarded to the Arizona Radiation Regulatory Agency (ARRA) pursuant to 10 CFR 50.91(b)(1).

APS has agreed with and has requested no exceptions to the recommendations of GL 99-02. APS requests that the enclosed technical specification amendment request be reviewed and approved by March 3, 2000, with an allowance of 45 days for implementation of the approved amendment.

This letter makes no commitments to the NRC. Please contact Mr. Scott Bauer at (623) 393-5978 if you have any questions or would like additional information regarding this matter.

Sincerely,

Gregg R. Overbeck

GRO/SAB/JAP/mah

Enclosures

cc: E. W. Merschoff
M. B. Fields
J. H. Moorman
A. V. Godwin (ARRA)

STATE OF ARIZONA)
) ss.
COUNTY OF MARICOPA)

I, G. R. Overbeck, represent that I am Senior Vice President – Nuclear, that the foregoing document has been signed by me on behalf of Arizona Public Service Company with full authority to do so, and that to the best of my knowledge and belief, the statements made therein are true and correct.

G. R. Overbeck

Sworn To Before Me This 19 Day Of November, 1999.

Nora E. Meador
Notary Public

"OFFICIAL SEAL"
Nora E. Meador
Notary Public-Arizona
Maricopa County
My Commission Expires 4/6/2003

My Commission Expires

April 6, 2003

ENCLOSURE 1

Proposed Amendment to
PVNGS Unit 1, 2 and 3
Technical Specification 5.5.11.c

A. DESCRIPTION OF THE PROPOSED TECHNICAL SPECIFICATION AMENDMENT

The proposed amendment revises the Palo Verde Nuclear Generating Station (PVNGS) Technical Specification (TS) Section 5.5.11.c, Ventilation Filter Testing Program (VFTP) to incorporate the recommendations of NRC Generic Letter 99-02, Laboratory Testing of Nuclear-Grade Activated Charcoal, dated June 3, 1999. The proposed amendment changes the standard by which the safety-related air-cleaning units used in the engineering safety features (ESF) ventilation systems are tested from American Society for Testing and Materials (ASTM) D3803-1979, "Standard Test Methods for Nuclear-Grade Activated Carbon", to the ASTM D3803-1989 standard.

PVNGS initiated a carbon change-out plan in 1994 based upon testing performed in 1993 on nuclear air treatment system's adsorber media (carbon) using the ASTM D3803-1989 protocol. The change-out of ESF adsorber carbon was completed in 1996. Additionally, PVNGS continued to perform this testing in conjunction with the regulatory required surveillance tests as a trending mechanism for performance and economic information.

This amendment request also revises the references relating to the requirement for obtaining the charcoal adsorber samples per Regulatory Guide 1.52, Revision 2, and ANSI standard N510-1980. The revised reference is Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, as implemented by Section 1.8 of the UFSAR. The ANSI N510-1980 standard is an incorrect reference by which charcoal adsorber samples are obtained. The correct references for obtaining the ESF charcoal adsorber samples are Regulatory Guide 1.52, Revision 2, and ANSI N509-1980. Regulatory Guide 1.52, Revision 2, and ANSI N509-1980 standards are both contained in PVNGS UFSAR, Section 1.8 and PVNGS specific VFTP procedures for obtaining and testing the charcoal adsorber samples.

B. PURPOSE OF THE TECHNICAL SPECIFICATION

The ESF ventilation systems function during emergency conditions to ensure that the offsite radiation exposures and exposures to operating personnel in the control room are within the guideline values of General Design Criterion (GDC) 19 of Appendix A of 10 CFR 50 and Subpart A of 10 CFR 100.

Testing the ESF nuclear-grade activated adsorber carbon (charcoal) is performed to ensure compliance with the current licensing basis as it relates to the dose limits of General Design Criterion (GDC) 19 of Appendix A of 10 CFR 50 and Subpart A of 10 CFR 100.

C. NEED FOR THE TECHNICAL SPECIFICATION AMENDMENT

The NRC has determined that testing nuclear-grade activated charcoal to standards other than ASME D3803-1989, "Standard Test Methods for Nuclear-Grade Activated Carbon," does not provide assurance for complying with the current licensing basis as it relates to the dose limits of General Design Criterion (GDC) 19 of Appendix A to 10 CFR 50 and Subpart A of 10 CFR 100.

The proposed amendment would revise the PVNGS TS Section 5.5.11.c, Ventilation Filter Testing Program for the laboratory testing and acceptance criteria of the ESF charcoal (carbon) adsorbers. The proposed amendment will revise the criteria for the ESF systems being tested from the ASTM D3803-1979 criteria of a temperature of 80°C, +/- 0.5°C and 70% relative humidity with an acceptance criteria of ≤ 1% penetration, to being tested in accordance with the methodology and tolerances of ASTM D3803-1989 criteria of a temperature of 30°C and 70% relative humidity with an acceptance criteria of ≤ 2.5% penetration. By changing this portion of the TS, PVNGS will be in conformance with the recommendations of NRC Generic Letter 99-02.

This amendment request also revises the references relating to the requirement for obtaining the charcoal adsorber samples per Regulatory Guide 1.52, Revision 2, and ANSI standard N510-1980. The revised reference is Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, as implemented by Section 1.8 of the UFSAR. The ANSI N510-1980 standard is an incorrect reference by which charcoal adsorber samples are obtained. This amendment request will remove the incorrect reference (ANSI N510-1980) for obtaining the ESF charcoal adsorber samples. Regulatory Guide 1.52, Revision 2, and ANSI N509-1980 are the correct standards for obtaining charcoal adsorber samples and both are contained in PVNGS UFSAR, Section 1.8 and PVNGS specific VFTP procedures.

D. SAFETY ANALYSIS FOR THE PROPOSED TECHNICAL SPECIFICATION AMENDMENT

The proposed amendment would change PVNGS TS Section 5.5.11.c, Ventilation Filter Testing Program (VFTP), to test to the ASTM D3803-1989 standard instead of the ASTM D3803-1979 standard to conform to the recommendations of NRC Generic Letter 99-02, Laboratory Testing of Nuclear-Grade Activated Charcoal. The proposed amendment is being implemented due to the NRC's position that testing nuclear-grade activated charcoal to standards other than ASTM D3803-1989, "Standard Test Methods for Nuclear-Grade Activated Carbon", does not provide assurance for complying with the current licensing basis as it relates to the dose limits of General Design Criterion (GDC) 19 of Appendix A to 10 CFR 50 and Subpart A of 10 CFR 100. Testing at an elevated temperature, as specified by ASTM D3803-1979 standard, results in an overestimation of the actual iodine-removal capability of the charcoal. Testing with the ASTM D3803-1989 standard gives results that represent a more realistic assessment of the capability of the charcoal.

Adsorber carbon plays a direct role in mitigating the consequences of a radiological event. Safety-related air-cleaning units used in the ESF ventilation systems of nuclear power plants reduce the potential onsite and offsite consequences of a radiological accident by adsorbing radioiodine. The proposed amendment changes the laboratory performance test criteria on adsorber carbon and will yield more accurate results of radioiodine removal efficiency than what is currently required by the TS. Hence, it will better ensure that the adsorber carbon efficiency for PVNGS TS systems used in the mitigation of an accident remains above the assumed carbon decontamination efficiency referenced in Chapter 6 and Chapter 15 of the UFSAR. This proposed amendment implements more stringent test criteria that will produce test results more accurately, better representing the capability of the charcoal adsorbers.

The new laboratory performance test that will be required by the proposed amendment is a more stringent test than the current TS required test. This testing will better characterize the carbon's ability to adsorb radioactive gases, expressed in methyl iodide adsorption efficiency. Therefore, reliability of the charcoal will be increased. Since the test will better identify the efficiency of the carbon, the possibility of operating with carbon with an efficiency below the decontamination efficiency assumed in the accident analysis is much lower.

This proposed amendment does not change, degrade, or prevent actions described or assumed in an accident. It will not alter any assumptions previously made in evaluating radiological consequences or affect any fission product barriers. It does not increase any challenges to safety systems as well. Therefore, this proposed amendment would not increase or have any impact on the consequences of events described and evaluated in Chapter 6 or Chapter 15 of the UFSAR.

E. NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

The Commission has provided standards for determining whether a significant hazards consideration exists as stated in 10 CFR 50.92. A proposed amendment to an operating license for a facility does not involve a significant hazards consideration if operation of the facility in accordance with a proposed amendment would not: 1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or 2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or 3) Involve a significant reduction in the margin of safety.

A discussion of these standards as they relate to this amendment request follows:

Standard 1 – Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

No. The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed change to TS 5.5.11.c, initiates a laboratory performance test of adsorber carbon (charcoal) that yields more accurate results than what is currently required by TS. The proposed change also deletes the specific reference to the ANSI standard by which the adsorber carbon sample is obtained. The proposed changes to test adsorber carbon to a more current and improved ASTM standard and delete the ANSI standard by which the adsorber carbon sample is obtained would not be plant accident initiators as described in Chapter 6 or Chapter 15 of the PVNGS UFSAR. The changes would not involve a significant increase in the probability of an accident previously evaluated.

Carbon adsorption plays a direct role in mitigating the consequences of a radiological event. Safety-related air-cleaning units used in the ESF ventilation systems of nuclear power plants reduce the potential onsite and offsite consequences of a radiological accident by the adsorption of radioiodine. The proposed amendment to change the laboratory performance test for carbon will yield more conservative results than what is currently required by TS. Hence, it will better ensure that the adsorber carbon for TS systems used in the mitigation of an accident remains above the assumed carbon decontamination efficiency referenced in Chapter 6 and Chapter 15 of the UFSAR.

This proposed amendment does not alter, degrade, or prevent actions described or assumed in an accident. It will not alter any assumptions previously made in evaluating radiological consequences or, affect any fission product barriers. It does not increase any challenges to safety systems as well. Therefore, this proposed amendment would not significantly increase the consequences of an accident previously evaluated.

Standard 2 – Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

No. The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed change to TS 5.5.11.c, initiates a laboratory performance test of adsorber carbon that yields more accurate results than what is currently required by TS. The proposed changes to test adsorber carbon to a more current and improved ASTM

standard and delete the specific reference to the ANSI standard by which the adsorber carbon sample is obtained would not be plant accident initiators as described in Chapter 6 or Chapter 15 of the PVNGS UFSAR. The proposed amendment does not change the function of any SSC. TS nuclear air treatment systems function to filter radiological releases during design basis accidents. This change will provide greater assurance that this function is provided. The revised TS required laboratory tests utilize practices now in place, changing only the testing parameters. The changes do not alter, degrade, or prevent actions described or assumed in an accident described in the PVNGS UFSAR from being performed. Therefore, the proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

Standard 3 – Does the proposed change involve a significant reduction in a margin of safety?

No. The proposed change does not involve a significant reduction in a margin of safety.

The margin of safety, as defined in the PVNGS Technical Specifications, is not reduced but is enhanced due to improved testing. This change initiates a laboratory performance test on adsorber carbon that yields more accurate results than what is currently required by TS and deletes the specific reference to the ANSI standard by which the adsorber carbon sample is obtained. The proposed change to test adsorber carbon to a more current and improved ASTM standard will ensure the carbon media's ability to adsorb radioactive gases will remain above that credited in the PVNGS' dose analysis for postulated accidents.

F. ENVIRONMENTAL IMPACT DETERMINATION

The proposed amendment i) involves no significant hazards consideration, ii) does not result in a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, and iii) does not result in a significant increase in individual or cumulative occupational radiation exposure. Therefore, the proposed amendment is categorically excluded from an environmental assessment in accordance with 10 CFR51.22(c)(9).

G. Marked-up Technical Specification Page

5.5 Programs and Manuals (continued)

5.5.11 Ventilation Filter Testing Program (VFTP) (continued)

ESF Ventilation System	Flowrate
Control Room Essential Filtration System (CREFS)	28,600 CFM
Engineered Safety Feature (ESF) Pump Room Exhaust Air Cleanup System (PREACS)	6,000 CFM

b. Demonstrate for each of the ESF systems that an inplace test of the charcoal adsorber shows a penetration and system bypass ≤ 1.0 % when tested in accordance with Regulatory Guide 1.52, Revision 2, and ANSI N510-1980 at the system flowrate specified as follows ± 10%:

ESF Ventilation System	Flowrate
CREFS	28,600 CFM
ESF PREACS	6,000 CFM

INSERT A

c. Demonstrate for each of the ESF systems that a laboratory test of a sample of the charcoal adsorber, when obtained as described in Regulatory Guide 1.52, Revision 2, and ANSI N510-1980 shows the methyl iodide penetration less than the value specified below when tested in accordance with ASTM D3803-1979 at a temperature of 80°C ± 0.5°C and greater than or equal to the relative humidity specified as follows:

ESF Ventilation System	Penetration	RH
CREFS	≤ 1.0%	70%
ESF PREACS	≤ 1.0%	70%

DELETE

(continued)

INSERT A TO PAGE 5.5-18

c. Demonstrate for each of the ESF systems that a charcoal adsorber sample, when obtained in accordance with the application of Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, as described in Section 1.8 of the UFSAR, shows the methyl iodide penetration less than or equal to the value specified below, when tested in accordance with ASTM D3803-1989, at a temperature of 30°C and to the relative humidity specified as follows:

ESF Ventilation System	Penetration	RH
CREFS	≤2.5%	70%
ESF PREACS	≤2.5%	70%

H. Retyped Technical Specification Page

5.5. Programs and Manuals (continued)

5.5.11 Ventilation Filter Testing Program (VFTP) (continued)

ESF Ventilation System	Flowrate
Control Room Essential Filtration System (CREFS)	28,600 CFM
Engineered Safety Feature (ESF) Pump Room Exhaust Air Cleanup System (PREACS)	6,000 CFM

b. Demonstrate for each of the ESF systems that an inplace test of the charcoal adsorber shows a penetration and system bypass ≤ 1.0 % when tested in accordance with Regulatory Guide 1.52, Revision 2, and ANSI N510-1980 at the system flowrate specified as follows ± 10%:

ESF Ventilation System	Flowrate
CREFS	28,600 CFM
ESF PREACS	6,000 CFM

c. Demonstrate for each of the ESF systems that a charcoal adsorber sample, when obtained in accordance with the application of Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, as described in Section 1.8 of the UFSAR, shows the methyl iodide penetration less than or equal to the value specified below, when tested in accordance with ASTM D3803-1989, at a temperature of 30°C and to the relative humidity specified as follows:

ESF Ventilation System	Penetration	RH
CREFS	≤ 2.5%	70%
ESF PREACS	≤ 2.5%	70%

(continued)

<u>ENCLOSURE 2</u>

PVNGS Response to Requested Actions for NRC Generic Letter 99-02, Laboratory Testing of Nuclear-Grade Activated Charcoal

ACTION 1:

NRC Request:
Within 180 days of the date of this generic letter, submit a written response to the NRC describing your current TS requirements for the laboratory testing of charcoal samples for each ESF ventilation system including the specific test protocol, temperature, RH, charcoal bed thickness, total residence time per bed depth, and penetration at which the TS require the test to be performed. If your current TS specifically requires laboratory testing of charcoal samples in accordance with the ASTM D3803-1989 protocol at 30 °C [86 °F], and you have been testing in accordance with this standard, then you only need to address this requested action (i.e. no TS amendment or additional testing is required).

PVNGS Response:
The existing criteria for testing each engineering safety features (ESF) system's (the Control Room Essential Ventilation System [CREVS] and the Fuel Building Essential Ventilation System [PREACS]) adsorber carbon samples at PVNGS is per Technical Specification (TS) 5.5.11, Ventilation Filter Testing Program, Part c, stating:

> Demonstrate for each of the ESF systems that a laboratory test of a sample of the charcoal adsorber, when obtained as described in RG 1.52, revision 2, and ANSI N510-1980 shows the methyl iodide penetration less than the value specified below when tested in accordance with ASTM D3803-1979 at a temperature of 80°C (+/- 0.5°C) and greater than or equal to the relative humidity specified as follows:
>
> | ESF Ventilation System | Penetration | RH |
> |---|---|---|
> | CREFS | $\leq$1.0% | 70% |
> | ESF PREACS | $\leq$1.0% | 70% |

The specified surveillance test utilizes the Method B test protocol listed in ASTM D3803-1979, Table 1, Radioiodine/Methyl Iodide Standard Test Conditions, as follows:

Temperature:	80°C.
Relative Humidity:	70% (credit taken for humidity controlled environment).
Carbon Bed Depth:	2 inches.
Residence Time:	~0.25 second / 40 fpm.
Penetration:	$\leq$ 1.0%

ACTION 2:

NRC Request:
If you choose to adopt the ASTM D3803-1989 protocol, submit a TS amendment request to require testing to this protocol within 180 days of the date of this generic letter. The request should contain the test temperature, RH, and penetration at which the proposed TS will require the test to be performed and the basis for these values. If the system has a face velocity greater than 110 percent of 0.203 m/s [40 ft/min], then the revised TS should specify the face velocity. Also, indicate when the next laboratory test is scheduled to be performed.

PVNGS Response:
PVNGS is requesting an amendment to PVNGS Technical Specification 5.5.11.c for testing each ESF system's (the Control Room Essential Ventilation System [CREVS] and the Fuel Building Essential Ventilation System [PREACS]) adsorber carbon samples in accordance with NRC Generic Letter 99-02, using the test protocol as established in ASTM D3803-1989. Thus, the test's temperature parameter will be 30^{o}C on the basis of the D3803-1989 standard.

PVNGS will apply 70% as the relative humidity test parameter for testing each ESF Control Room Essential Ventilation System's [CREVS] adsorber carbon samples. This is an exception from the ASTM D3803-1989 test protocol, which uses a value of 95% relative humidity. The basis for the test's 70% relative humidity is that it has been determined through design calculation (13-MC-HJ-003, Revision 1, HJ System Heat Load and Equipment Selection Calculation) that air entering the CREVS carbon adsorber will be maintained less than 70% relative humidity. This value has received NRC approval in that it is the specified value for relative humidity in PVNGS' current Technical Specification 5.5.11.c.

PVNGS will apply 70% as the relative humidity test parameter for testing each ESF Fuel Building Essential Ventilation System's [PREACS]) adsorber carbon samples. This is an exception from the ASTM D3803-1989 test protocol, which uses a value of 95% relative humidity. The basis for the test's 70% relative humidity is that it has been determined through design calculation (13-MC-HF-252, Revision 2, Fuel Building /Auxiliary Building Infiltration Calculation & Verification of Adequacy of Existing Unit) that air entering the PREACS carbon adsorber will be treated by the nuclear air treatment system's heater and maintained less than 70% relative humidity. This value has received NRC approval in that it is the specified value for relative humidity in PVNGS' current Technical Specification 5.5.11.c.

The penetration acceptance criterion of ≤2.5% was based upon guidance found in NRC Generic Letter 99-02, Attachment 2. This guidance describes the equation for determining the allowable penetration. Per this guidance, testing to ASTM D3803-1989 allows for a safety factor of 2. A value of 95% was used for the methyl iodide efficiency for charcoal credited in the PVNGS accident analysis based on information found in Table 2 of Regulatory Guide 1.52, Revision 2, March, 1978 and PVNGS UFSAR Chapters 6 and 15.

Therefore, in applying the criteria of Attachment 2 to GL 99-02, PVNGS Technical Specification Section 5.5.11.c will be revised as follows:

Demonstrate for each of the ESF systems that a charcoal adsorber sample, when obtained in accordance with the application of Regulatory Position C.6.b. of Regulatory Guide 1.52, Revision 2, March 1978, as described in Section 1.8 of the UFSAR, shows the methyl iodide penetration less than or equal to the value specified below, when tested in accordance with ASTM D3803-1989 at a temperature of 30°C and to the relative humidity specified as follows:

ESF Ventilation System	Penetration	RH
CREFS	≤2.5%	70%
ESF PREACS	≤2.5%	70%

The following two tables identify the adsorber carbon sampling/testing schedules for the Control Room Essential Ventilation Systems [CREVS] and the Fuel Building Essential Ventilation Systems [PREACS]) at the PVNGS three-unit-site. Additionally, since PVNGS has been routinely performing tests* using the ASTM D3803-1989 protocol, the most recent results are included.

Control Room Essential Ventilation System

Equipment ID NO.	Next Performance	Last Performance	Results (% Efficiency.)*
1M-HJA-F04	11/23/2000	05/21/1999	99.550
1M-HJB-F04	11/08/2000	05/06/1999	99.438
2M-HJA-F04	08/23/2000	02/18/1999	99.412
2M-HJB-F04	06/20/2000	12/16/1998	98.920
3M-HJA-F04	01/18/2001	07/16/1999	99.445
3M-HJB-F04	03/20/2000	09/15/1998	99.180

*Testing was performed using ASTM D3803-1989 (30°C/95%RH) in conjunction with regulatory required tests being performed using D3803-1979, per PVNGS technical specifications. Testing was performed as a trending mechanism for performance, economic evaluation information, and test comparisons. Results from all systems tested comply with the proposed acceptance criteria of ≤ 2.5% penetration.

Fuel Building Essential Ventilation System

Equipment ID NO.	Next Performance	Last Performance	Results (% Efficiency)*
1M-HFA-J01	10/24/2000	04/21/1999	99.614
1M-HFB-J01	10/12/2000	04/05/1999	99.629
2M-HFA-J01	12/18/2000	06/15/1999	98.954
2M-HFB-J01	11/05/2000	05/03/1999	99.520
3M-HFA-J01	12/20/1999	06/16/1998	99.360
3M-HFB-J01	12/02/1999	05/29/1998	99.140

*Testing was performed using ASTM D3803-1989 (30°C/95%RH) in conjunction with regulatory required tests being performed using D3803-1979, per PVNGS technical specifications. Testing was performed as a trending mechanism for performance, economic evaluation information, and test comparisons. Results from all systems tested comply with the proposed acceptance criteria of ≤ 2.5% penetration.

ACTION 3:

NRC Request:
If you are proposing an alternate test protocol, address the attributes discussed below and submit a TS amendment request to require testing to this alternate protocol within 180 days of the date of this generic letter.

PVNGS Response:
N/A. PVNGS is not proposing an alternate test protocol.

ACTION 4:

NRC Request:
At the next required laboratory surveillance test of a charcoal sample that is 60 or more days after the date of this generic letter, test your charcoal samples in accordance with ASTM D3803-1989 or replace all of the charcoal with new charcoal that has been tested in accordance with ASTM D3803-1989. In all cases, the results should meet the acceptance criterion that is derived from applying a safety factor as low as 2 (see the note in Enclosure 2) to the charcoal filter efficiency assumed in your design-basis dose analysis and the charcoal samples should continue to be tested in accordance with ASTM D3803-1989, in lieu of the current TS-required laboratory testing, until the TS amendment is approved by the NRC.

PVNGS Response:
Since PVNGS already performs testing in accordance with the ASTM D3803-1989 protocol for purposes of performance monitoring, actions have been completed to revise existing surveillance testing procedures to conform to ASTM D3803-1989 protocol.

The surveillance test procedures have been revised to test ESF carbon samples in accordance with ASTM D3803-1989, at the temperature of 30°C and 70% relative humidity, stating an acceptance criteria of less than or equal to 2.5% penetration. This testing will be performed in conjunction with the current PVNGS technical specification criteria until this proposed amendment is approved. At the time in which this proposed technical specification amendment is approved, the surveillance testing procedures' acceptance criteria will be revised to eliminate the current technical specification criteria.

ACTION 5:

NRC Request:
Addressees who choose not to do the above actions are requested to notify the NRC in writing of their decision, as soon as a decision is reached but no later than 60 days from the date of this generic letter. The 60 day written response should also discuss (1) addressee plans to pursue a proposed alternative course of action (including the basis for establishing its acceptability), (2) the schedule for submitting that proposal for NRC staff review (that proposal should be submitted to the NRC no later than 180 days from the date of this generic letter), and (3) the basis for continued operability of affected systems and components until such time that the proposed alternative course of action is approved by the NRC.

PVNGS Response:
N/A. PVNGS will comply with NRC Generic Letter 99-02.

10/29/99

Distri~6.txt
Distribution Sheet

Priority: Normal

From: Linda Eusebio

Action Recipients:	Copies:	
NRR/DLPM/LPD4-2	1	Not Found
M Fields	1	Not Found
C Jamerson	1	Not Found
Internal Recipients:		
RidsManager	1	OK
OGC/RP	1	Not Found
NRR/DSSA/SRXB	1	Not Found
NRR/DSSA/SPLB	1	Not Found
File Center 01	1	Not Found
ACRS	1	Not Found
External Recipients:		
NRC PDR	1	Not Found
NOAC	1	Not Found

Total Copies:	11	

Item: ADAMS Document
Library: ML_ADAMS^HQNTAD01
ID: 993140346

Subject:
Response to NRC October 28, 1999 Telcon RAI Re Licensee Proposed Amend to Tech Spec Section 3.8.4, "DC Sources - Operating, " Dated October 8, 1999.

Body:

Docket: 05000528, Notes: STANDARDIZED PLANT

Docket: 05000530, Notes: Standardized plant.

A-2

Palo Verde Nuclear
Generating Station

David Mauldin
Vice President
Nuclear Engineering
and Support

TEL (623) 393-5553
FAX (623) 393-6077

Mail Station 7605
P.O. Box 52034
Phoenix, AZ 85072-2034

102-04366-CDM/SAB/RKR
October 29, 1999

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Station P1-37
Washington, DC 20555-0001

Reference: Letter 102-04356-CDM/SAB/RKR, dated October 8, 1999, from C. D. Mauldin, APS, to NRC, "Proposed Amendment to Technical Specification Section 3.8.4, "DC Sources - Operating" Under Exigent Circumstances."

Dear Sirs:

Subject: Palo Verde Nuclear Generating Station (PVNGS)
Units 1 and 3
Docket Nos. STN 50-528/530
Response to NRC Request for Additional Information

In an October 28, 1999, phone call between NRC and PVNGS staffs, the NRC staff requested additional information to complete their review of the proposed amendment to Technical Specifications submitted in reference 1. Specifically the NRC staff requested additional information regarding the adequacy of the battery charging method following discharge testing.

The battery vendor performed a series of tests in 1996 to determine the optimum method for charging batteries following performance (deep discharge) tests. The vendor tested seven groups of batteries using different charging methodologies following deep discharges. The testing identified an optimum charging methodology that maximized battery capacity recovery following a deep discharge. The battery group that was tested using the optimum method was deep discharged and recharged nine consecutive cycles varying the duration of the recharge. The testing showed that battery capacity remained above 100% in all but the last two tests. In all cases the capacity was well above the 90 percent minimum required by Technical Specifications.

A001

ML993140346

PVNGS adopted the optimum charging method recommended by the battery vendor based on this testing and uses it following both service and performance tests. Based on the vendor testing, the charging method used, and operating experience with the Unit 1 batteries; PVNGS is confident that the Unit 1 batteries are capable of supplying over 100 percent of their rated capacity following the recharge from the battery service test.

In reference 1, Arizona Public Service Company (APS) requested the proposed Technical Specification amendment be processed under exigent circumstances in accordance with 10 CFR 50.91(a)(6). APS still maintains the position that this amendment merits NRC approval prior to startup from the current refueling outage. Without exigent approval, APS will be forced to consider extending the current outage or planning for an additional outage in early December 1999 to perform the required Technical Specification surveillance.

No commitments are being made to the NRC by this letter.

Should you have any questions, please contact Scott A. Bauer at (623) 393-5978.

Sincerely,

David Mauldin

CDM/SAB/RKR/kg

cc: E. W. Merschoff
M. B. Fields
J. H. Moorman
A. V. Godwin

NOV 0 5 1998