10 CFR 50.90

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102-06729-DCM/RKR/CJS
July 25, 2013

U.S. Nuclear Regulatory Commission
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Dear Sirs:

Subject: **Palo Verde Nuclear Generating Station (PVNGS) Units 1, 2, and 3**
Docket Nos. STN 50-528, 50-529, and 50-530
Response to Request for Additional Information Regarding License Amendment Request to Eliminate the Use of the Term CORE ALTERATION in the Technical Specifications

By letter number 102-06486, dated March 8, 2012 [Agencywide Documents Access and Management System (ADAMS) accession number ML12076A045], as supplemented by letter numbers 102-06604 and 102-06655, dated October 11, 2012 and January 31, 2013, respectively (ADAMS accession numbers ML12286A330 and ML13039A013), Arizona Public Service Company (APS) submitted a license amendment request (LAR) for Palo Verde Nuclear Generating Station (PVNGS), Units 1, 2, and 3. The proposed LAR would eliminate the use of the term CORE ALTERATION from the Technical Specifications.

The U.S. Nuclear Regulatory Commission (NRC) staff reviewed the information provided by APS and determined that additional information was needed in order to complete its review. By letter dated June 3, 2013 (ADAMS accession number ML13150A081), the NRC staff provided a request for additional information (RAI) and indicated that APS agreed to respond by July 18, 2013. On July 17, 2013, APS requested and was granted an extension until July 25, 2013, to provide the RAI response. The enclosure to this letter provides the APS response to the NRC RAI.

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

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Should you need further information regarding this response, please contact Robert K. Roehler, Licensing Section Leader, at (623) 393-5241.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on July 25, 2013
(Date)

Sincerely,

TN Weber...
for D.C. Mims

Enclosure: Response to Request for Additional Information Regarding License Amendment Request to Eliminate the Use of the Term CORE ALTERATION in the Technical Specifications

DCM//RKR/CJS/hsc

cc:	A. T. Howell III	NRC Region IV Regional Administrator
	E. J. Leeds	NRC Director Office of Nuclear Reactor Regulation
	J. K. Rankin	NRC NRR Project Manager
	J. A. Reynoso	NRC Senior Resident Inspector (Acting) for PVNGS

ENCLOSURE

Response to Request for Additional Information Regarding License Amendment Request to Eliminate the Use of the Term CORE ALTERATION in the Technical Specifications

Response to Request for Additional Information Regarding License Amendment Request to Eliminate the Use of the Term CORE ALTERATION in the Technical Specifications

Introduction

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NRC Request 1

Chapter 15, Section 15.4.6 of the Updated Final Safety Analysis Report (UFSAR) indicates that during operational Modes 3 through 6, the operator relies upon a high neutron flux alarm from the Boron Dilution Alarm System (BDAS) to identify and terminate a boron dilution event. Pages 3 and 4 of the supplement dated January 31, 2013, indicates that the BDAS relies on the startup channels (source range monitor, SRM) in the excore neutron flux monitoring system (ENFMS) to provide source level neutron flux information. The last paragraph on Page 3 indicates that the ENFMS contains nonsafety-related channels.

The regulations in 10 CFR 50.2 define safety-related structures, systems, and components (SSCs) as those SSCs:

> that are relied upon to remain functional during and following design basis events to assure:
>
> 1) The integrity of the reactor coolant boundary
> 2) The capability to shut down the reactor and maintain it in a safe shutdown condition; or
> 3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the applicable guideline exposures set forth in § 50.34(a)(1) or § 100.11 of this chapter, as applicable.

Please confirm whether the BDAS is a safety-related or nonsafety-related system. If the BDAS is a nonsafety-related system, please justify the

adequacy of use of the BDAS for mitigating the consequences of the boron dilution event, a design-basis-event (DBE).

APS Response

The ENFMS is a nonsafety-related system and it provides input to BDAS. BDAS is also a nonsafety-related system at PVNGS. The justification for the classification as nonsafety-related is provided by first a description of the design history of BDAS, followed by a summary of the relevant licensing history and finally a comparison to the safety-related definition of 10 CFR 50.2.

Design History of the BDAS

American National Standard ANSI N18.2, *Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants*, 1973 was used for the initial design of PVNGS, with regard to accident analysis. An inadvertent boron dilution event is called 'an inadvertent chemical shim dilution' event in the standard and is classified as a Condition II event; meaning, that it is an event that 'may occur during a calendar year for a particular plant.' As such, an inadvertent boron dilution event is not a limiting fault event, for which the dose consequence criteria of 10 CFR Part 100 would apply. The design requirement for such events is described as follows, from Section 2.1.2.3 of the Standard:

> "Condition II incidents shall be accommodated with, at most, a shutdown of the reactor with the plant capable of returning to operation after corrective action. Any release of radioactive materials in effluents to unrestricted areas shall be in conformance with paragraph 20.1 of 10 CFR Part 20, *Standards for Protection Against Radiation.* By itself, a Condition II incident cannot generate a more serious incident of the Condition III or IV type without other incidents occurring independently. A single Condition II incident shall not cause consequential loss of function of any barrier to the escape of radioactive products."

As the dose consequences of an inadvertent dilution event are limited, the design features for the event include manual and automatic reactivity control (see sections 5.2.3 and 5.2.5 of ANSI 18.2). There is no design requirement to provide automatic protection instrumentation for Condition II events, as is the case for Condition IV, faulted events, as described in Section 2.1, *Conditions For Design,* which states, in part:

> "The basic principle applied is that the plant shall be designed such that the most frequent occurrences yield little or no adverse consequence to the public, and such that the improbable extreme situations, having the potential for the greatest adverse consequence

> to the public, shall have a low probability of occurrence. Protection system and engineered safety feature functioning is required, where applicable, in fulfilling this principle."

As indicated in PVNGS UFSAR Section 15.4.6, *Inadvertent Deboration,* the event is classified as an incident of moderate frequency (e.g., not a limiting fault or infrequent incident), as defined in Regulatory Guide 1.70, *Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants, LWR Edition*, Revision 3, November 1978.

When the reactor is critical (Modes 1 and 2) an inadvertent deboration event will result in a slow increase in core power and RCS temperature. This event is slower than other reactivity excursions analyzed (e.g., CEA withdrawals), and the reactor will trip in time to prevent violation of any safety limit.

If the reactor is subcritical with trippable CEAs withdrawn from the core (all rods out in Modes 3 through 5), the high log power trip is active. If the operator is not notified of the event with sufficient time to prevent criticality, the high log trip will actuate and insert the withdrawn CEAs into the core, terminating the event.

If the reactor is subcritical without trippable CEAs (all rods in while in Modes 3 through 5), an event will result in degradation of shutdown margin. Either the BDAS, or chemical surveillance of the RCS boron concentration (when BDAS is not operable), will alert the operator of the event.

If the reactor is in Mode 6 (Refueling), an event will result in degradation of shutdown margin. In Mode 6, the CEAs may be totally removed from the core, thus the all rods out configuration is the most limiting. Either the BDAS, or chemical surveillance of the RCS boron concentration (when BDAS is not operable), will alert the operator of the event.

UFSAR Section 15.4.6.5 concludes:

> "The ID [inadvertent deboration] event will result in acceptable consequences. Sufficient time is available for the operator to detect and to terminate an ID event if it occurs. Fuel integrity is not challenged during this event."

Based upon the design criteria, BDAS and related instrumentation do not protect against limiting faults or infrequent incidents that have the potential for public dose consequences.

Licensing History of the BDAS

BDAS was not part of the initial plant design at the outset of NRC review of the Combustion Engineering (CE) System 80 design. The NRC Safety

Evaluation Report (SER) (NUREG-0852) for the CE System 80 Standard Safety Analysis Report (CESSAR) Section 15.2.4.5, *Inadvertent Boron Dilution,* stated:

> "Section 15.4.6 of the Standard Review Plan requires that at least 15 minutes is available from the time the operator is made aware of an unplanned boron dilution event to the time a loss of shutdown margin occurs during power operation, startup, hot standby, hot shutdown, and cold shutdown. Thirty minutes warning is required during refueling. CE indicated that operating procedures would be utilized to respond to boron dilution events in modes 3 through 6. The staff has requested that control room alarms be available to alert the operating staff to boron dilution events in all modes of operation. If a second alarm is not provided, CE must show that the consequences of the most limiting unmitigated boron dilution event meet the staff criteria and are acceptable. The staff requires that the applicant provide an analysis for all possible boron dilution events in each of the 6 operational modes and confirm that time intervals which meet the SRP criteria from the time of the first alarm to the time when the core would go critical is available. Also, technical specifications should be established to restrict when alarms can be taken out of service. In a letter dated October 29, 1981 CE committed to provide redundant boron alarms for all modes of operation. We will report on the resolution of this issue in a revision to this report."

The NRC SER for the CESSAR, Section 15.2.6, *Conclusions,* stated:

> "CESSAR has presented results for various anticipated operational occurrences (with and without assumed single failures). With the exceptions noted below, the staff finds they meet NRC acceptance criteria with respect to fuel and primary system performance. Therefore, the applicant has provided adequate protection for anticipated operational occurrences (except as noted) and is considered in compliance with GDC 10, 15, and 26."
>
> "In the inadvertent boron dilution event the staff will confirm that CESSAR provides redundant alarms to positively identify boron dilution events in all modes of operation and provides analysis for boron dilution events in each mode of plant operation."

CESSAR Section 7.7.1.10 (Amendment No. 7, March 31, 1982) provided the original description of the BDAS.

In Supplement 1 to the NRC SER for the CESSAR, the NRC staff documented their review of the CE BDAS design as follows:

> "The SER states that CE had provided adequate protection for anticipated operational occurrences with the exception of the inadvertent boron dilution event which was still under review. As indicated in section 15.2.4.5 of this supplement, the staff has now completed its review of the boron dilution event and, therefore, the following evaluation supersedes that presented in section 15.2.6 of the SER."
>
> "CESSAR has presented results for various anticipated operational occurrences (with and without assumed single failures). The staff finds they meet acceptance criteria of the applicable Standard Review Plans with respect to fuel and primary system performance. Therefore, CE has provided adequate protection for anticipated operational occurrences and CESSAR is considered in compliance with GDC 10, 15, and 26."

NUREG-0857, *Safety Evaluation Report Related to the Operation of PVNGS Units 1, 2, and 3* (hereafter referred to as the PVNGS SER), in Section 15.2, *Normal Operation and Anticipated Transients* (of which the inadvertent boron dilution event is one), refers to the staff evaluation in the CESSAR SER for the acceptability of BDAS.

The PVNGS SER, Section 7.4, *Systems Required for Safe Shutdown,* does not include BDAS as a system required for safe shutdown. Instead, BDAS is described in Section 7.7, *Control Systems,* (also called, *Control Systems Not Required for Safety,* in CESSAR Section 7.7). The PVNGS SER refers back to the NRC staff SER for the CESSAR (quoted in part above) for a description of the NRC staff evaluation of these nonsafety-related systems.

BDAS has consistently been classified as a nonsafety-related system in the CE System 80 and PVNGS licensing processes. More information with regard to this licensing history is provided in the response to NRC Request 2.3.

Comparison to the Safety-Related Definition of 10 CFR 50.2

Each of the three (3) sub-parts of the of 10 CFR 50.2 definition of *safety-related structures, systems, and components (SSCs)* are addressed individually.

> (1) Integrity of the reactor coolant boundary – BDAS instrumentation is external to the reactor coolant system and does not interact with the reactor coolant system pressure boundary; therefore, this element of the definition is not applicable to BDAS.
>
> (2) Capability to shut down the reactor and maintain it in a safe shutdown condition – BDAS instrumentation does not interface with the reactor protective system; thus, it does not have the capability to

shutdown the reactor, nor is it credited in establishing initial safe shutdown conditions (control rod insertion and/or shutdown boron concentration) or longer term safe shutdown margin. The system functions to alert operators that an inadvertent boron dilution event may be in progress, but is not credited for routine operational control of safe shutdown margin. This element of the definition is not, therefore, applicable to BDAS.

(3) Capability to prevent or mitigate the consequences of accidents - The BDAS instrumentation has one function and that is to provide operators with an alarm that an inadvertent boron dilution event may be in progress. The purpose of the alarm is to provide operators at least the minimum required time (documented in NUREG-0800, Section 15.4.6) to take manual action to stop the dilution and implement boration procedures. The BDAS instrumentation (in the CE System 80 and PVNGS design) performs no automatic actions, and as such, does not directly mitigate the consequences of a DBE. With respect to preventing a DBE, the system only alarms when an inadvertent boron dilution event may be in progress, so it does not prevent an event from occurring. This element of the definition is not, therefore, applicable to BDAS.

The BDAS and ENFMS systems at PVNGS are highly reliable but are not classified as safety-related. They are not part of the reactor coolant system, nor are they used to shutdown the reactor or directly mitigate the consequences of a DBE. Rather they assist the operations staff in identifying and manually performing actions to address a Condition II, moderate frequency event with limited consequences. The PVNGS current licensing basis credits operator action to mitigate the consequences of an inadvertent boron dilution event. Boron dilution events are comparatively slowly developing events, particularly during refueling operations (Mode 6). Ample time exists to recognize and mitigate the event, as demonstrated in the safety analysis and accepted by the NRC staff. The time available for operator action is described further in the response to NRC Request 2.3, below.

NRC Request 2

Adequacy of BDAS Alarm Setpoint during Core Offload in Mode 6 with one Operable SRM

NRC Request 2.1

Paragraph 3 on page 4 of the supplement dated January 31, 2013, states, in part, that

> Once sufficient fuel assemblies have been removed, such that one SRM (and its associated BDAS channel) has indication of higher neutron flux level than the other SRM; the BDAS channel that is associated with the SRM that has reduced neutron flux indication will automatically reduce its setpoint to a lower value, commensurate with the lower neutron flux reading on the SRM. Thus, from this point in the core offload process until the core is entirely offloaded to the spent fuel pool, the two independent BDAS channel will have different setpoints but remain capable of indicating a boron dilution event.

The above response addresses an automatic reduction in the BDAS alarm setpoint that is limited to core offloaded conditions in Mode 6 with two operable SRMs. It is unclear if the BDAS setpoint will decrease if only one SRM is operable. Please expand the information to address the adequacy of use of one operable SRM (with the other SRM failed, as specified in Condition A of TS 3.9.2) for monitoring the core sub-criticality in Mode 6 while fuel assemblies are being removed from the core in combination with occurrence of a boron dilution event.

APS Response

Request 2.1

As noted in the last sentence of the first paragraph on page 4 of supplement dated January 31, 2013 (quoting UFSAR Section 7.7.1.1.11):

> "There are two redundant and independent channels in the Boron Dilution Alarm System (BDAS) to ensure detection and alarming of the event."

Each of the BDAS channels is independent of each other. So, if one of the SRMs is declared inoperable, the other BDAS channel will continue to function as designed, and its setpoint will automatically decrease as the neutron flux level is decreased.

Technical Specification Action Requirements – Proposed

1) During core offload, if an SRM experiences a failure such that it is inoperable, TS 3.9.2 and 3.3.12 Conditions would be entered. The proposed PVNGS TS would require 2 actions to be completed immediately.

   The first action is TS 3.9.2, Condition A.1, Suspend positive reactivity additions. While this term is not specifically defined, it means that both boron dilutions and changes in reactor coolant system (RCS) temperature that add positive reactivity to the core (hence reduce shutdown margin / decrease core sub-criticality) are to be secured, as clarified by APS RAI response dated October 11, 2012 (APS letter no. 102-06604).

The second action is TS 3.3.12, Condition A.1, Determine the RCS boron concentration. This requirement is based on the boron concentration measurement and the operable BDAS channel providing alternate methods of detection of boron dilution with sufficient time for termination of the event before the reactor achieves criticality (as documented in TS Bases 3.3.12, description of Action A.1).

Therefore, for the remainder of the core offload (assuming the instruments remain inoperable), TS 3.3.12 requires that the RCS boron concentration be monitored at the frequency specified in the Core Operating Limits Report (COLR). As noted in TS Bases A.1, RCS boron concentration is determined by RCS sampling.

2) During core reload, if a SRM experiences a failure such that it is inoperable, TS 3.9.2 and 3.3.12 Conditions would be entered. The proposed PVNGS TS would require 2 actions to be completed immediately.

   The first action is TS 3.9.2, Condition A.1, Suspend positive reactivity additions. While this term is not specifically defined, it means that boron dilutions, placement of additional fuel assemblies into the core and changes in reactor coolant system (RCS) temperature that add positive reactivity to the core (hence reduce shutdown margin / decrease core sub-criticality) are to be secured, as clarified by APS RAI response dated October 11, 2012 (APS letter no. 102-06604).

   The second action is TS 3.3.12, Condition A.1, Determine the RCS boron concentration. This requirement is based on the boron concentration measurement and the operable BDAS channel providing alternate methods of detection of boron dilution with sufficient time for termination of the event before the reactor achieves criticality (as documented in TS Bases 3.3.12, description of Action A.1).

   Therefore, from that point in the core reload (assuming the instruments remain inoperable), TS 3.9.2 would prohibit placing of any more fuel assemblies in the core, since the addition of fuel assemblies would be a positive reactivity addition. In addition, TS 3.3.12 would require that the RCS boron concentration be monitored at the frequency specified in the COLR.

   If repair of the inoperable SRM requires work in proximity to the reactor vessel, core offload would likely be required to limit personnel exposure. The proposed TS would permit core offload, whereas, the existing TS would not and the only viable alternative would be to seek emergency NRC relief in the form of an exigent or emergency TS change.

NRC Request 2.2

Paragraph 4 on page 6 of the supplement dated January 31, 2013, states:

> The proposed change would only allow the addition of soluble poison to the RCS [reactor coolant system] coolant (per TS 3.9.1 and COLR [Core Operating Limits Report] item 3.9.1) that increases the uniform RCS boron concentration. Such a change would not result in a positive change in core reactivity or a reduction in core sub-criticality.

The above response does not consider the reduction in sub-criticality resulting from a boron dilution event, which may be caused by an operator error to inject unborated water into the RCS. Please explain how this is acceptable considering the fact that a boron dilution event is a UFSAR Chapter 15 DBE and would be expected to be considered in support of a TS change to remove the term, CORE ALTERATION, from the TS.

APS Response

Request 2.2

The deletion of the term CORE ALTERATIONS from TS 3.9.1 has no effect on the boron dilution event, which is a moderate frequency event, and not a limiting fault or infrequent incident, as described in the response to NRC Request 1. The boron dilution event could be initiated by an operator error that leads to injection of unborated water into the RCS. As described in the response to NRC Request 1, BDAS instrumentation and RCS sampling (if DBAS is inoperable) are the credited methods to inform the operations staff of such a boron dilution event in Mode 6.

The PVNGS TS, both currently and in the proposed change, contain the statement, 'Suspend positive reactivity additions.' In addition, both versions of the TS also contain the statement, 'Initiate action to restore boron concentration to within limit.' Thus, once compliance with TS 3.9.1 is no longer being met, operators are required to take action to restore the boron concentration (and compliance with TS). TS Bases 3.9.1, *ACTIONS,* description (A.3 in existing TS and proposed A.2) further states:

> "In order to raise the boron concentration as soon as possible, the operator should begin boration with the best source available for unit conditions. Once boration is initiated, it must be continued until the boron concentration is restored."

The proposed change, therefore, does not alter either the likelihood or consequences of an operator error that may initiate a boron dilution event. The existing and proposed TS are both considered sufficient to address an

operator error that may lead to this slow moving, limited consequence, boron dilution event.

NRC Request 2.3

Paragraph 6 on page 6 of the supplement dated January 31, 2013 states:

> Should the remaining operable SRM (when there is reduced numbers of fuel assemblies in the core) experience a failure, such that it is inoperable, then TS Bases 3.3.12, Action B.1, describes the redundant methods that are to be implemented when both independent channels of BDAS are inoperable. The use of redundant methods to monitor the RCS boron concentration provides alternate indications of inadvertent boron dilution. This will allow detection with sufficient time for termination of a boron dilution event before the reactor achieves criticality, consistent with the objectives of the SRP.

The above response discussed the use of the sampling technique to monitor the RCS boron concentration as alternate indications of inadvertent boron dilution. The information does not discuss the frequency of the sampling and time to complete the sampling to assure that the operator can detect and terminate the boron dilution event before the core reaches criticality in 30 minutes, which is specified as an acceptance criterion in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," (SRP) Section 15.4.6, "Inadvertent Decrease in Boron Concentration in the Reactor Coolant System (PWR)," for a boron dilution event during the Mode 6 operation.

Please provide information to show the adequacy of the BDAS alarm setpoint based on input from only one operable SRM for conditions where the fuel assemblies are being removed from the core during the Mode 6 operation. Please (1) discuss identification of the worst case, with fuel assembly removal in a configuration that results in the least effectiveness of the operable SRM to detect neutron fluxes, and the least time available for detection and termination of a boron dilution event; and (2) show that for the identified worst case, the BDAS based on one operable SRM provides adequate, reliable, and un-ambiguous signals for the operator to detect and terminate the boron dilution event before the core reaches criticality in the required 30 minutes in the Mode 6 operation.

Also, please provide information to show that the frequency of the sampling of the RCS boron concentration is adequate and the time of completion of the sampling are sufficient in providing signals for the operator to detect and terminate the boron dilution event before the core reaches criticality in 30 minutes as specified in SRP Section 15.4.6.

APS Response

Request 2.3

The BDAS channels are redundant and independent as described in the response to NRC Request 2.1 above. In addition, as described later in this response, RCS sampling is the appropriate method of detection of an inadvertent deboration event should one or both BDAS channels become inoperable. In that respect, there is no 'worst case' situation for one channel being inoperable, as RCS sampling is effective in detecting the potential inadvertent deboration event. The remaining operable BDAS channel (or RCS sampling on the specified frequency) provides adequate, reliable, and un-ambiguous information for the operator to detect and terminate the boron dilution event.

As noted in UFSAR Chapter 15, Table 15.4.6-2, the RCS volume analyzed in the Mode 6 *Inadvertent Deboration* (ID) event is conservatively set to match the RCS volume in Mode 5 (cold shutdown) in the drained-down configuration (4500 ft^3) which results in the shortest available time for detection and termination of the event. Specifically, the active volume of borated water is assumed to consist of only the volume of the reactor vessel (excluding the upper head region) and the volume of the shutdown cooling system. The volume of borated water that resides in the refuel pool during fuel assembly movement is conservatively not included in the Mode 6 ID event analysis or in the calculations that establish the RCS sampling frequencies in the COLR.

Analyses were performed that determined the required RCS sampling frequency presented in the five tables included in the COLR. These tables specify the frequency of chemical sampling needed to ensure the required notification of an ID event. These tables address specific ranges of initial K_{eff}. The tables have a specific surveillance interval entry for each possible operational mode/number of operating charging pumps combination. The table entries are determined by calculating the minimum time to lose shutdown margin for the particular initial K_{eff}/operating mode/number of operating charging pumps combination and subtracting the time for chemical sampling and the minimum time the operator must have to take corrective action before loss of shutdown margin. The calculation of the frequency of the RCS sampling was performed at the RCS minimum volume to determine the required time limits that are documented in COLR Table 3.3.12-5.

PVNGS TSs 3.9.4 and 3.9.5 both address the Shutdown Cooling (SDC) and coolant circulation requirements when in Mode 6. Both TSs contain the same Note, 'The required SDC loop may be removed from operation for $\leq$ 1 hour per 8-hour period, provided no operations are permitted that would cause reduction of the Reactor Coolant System boron concentration.' Both TS Bases specify that to be considered operable, the flow rate provided by the SDC pump must be greater than or equal to 3780 gpm. These two TSs work in

concert, in Mode 6, to ensure no matter what the RCS level is, that at least one SDC loop is in operation to provide:

1. Removal of decay heat,
2. Mixing of borated coolant, to minimize the possibility of criticality, and
3. Indication of reactor coolant temperature.

During the refueling process the volume of highly borated water is greater than the minimum volume that is analyzed in the limiting Mode 6 ID event. As noted in TS Basis 3.9.1:

> "During refueling, the water volume in the spent fuel pool, the transfer canal, the refueling canal and the reactor vessel form a single mass. As a result, the soluble boron concentration is relatively the same in each of these volumes."

The continuous mixing of these different volumes during the refueling process is accomplished by several processes. The reactor vessel volume is mixed by the forced flow provided by the operating SDC system. The refueling pool is mixed by the movement of the refueling machine mast through the refueling pool in containment. The transfer canal volume of water is mixed by the operation of the fuel transfer machine moving fuel assemblies between the containment and the fuel building. The spent fuel pool, in the fuel building, is mixed by the operation of the spent fuel handling machine and the forced circulation provided by the spent fuel pool cooling system.

In addition, the volume of water in the reactor vessel and the refuel pool is mixed by the thermal driving head induced by the difference in the temperature of water. The water in the reactor vessel that is in contact with the irradiated fuel assemblies is warmer than the bulk water of the refueling pool. Therefore, some natural circulation takes place, as warm water from the reactor vessel rises up from the fuel assemblies in the vessel to the refuel pool, and cool refueling pool water sinks into the reactor vessel.

RCS volume/mass is a first order effect on the calculation of the dilution rate used in the ID event (UFSAR Section 15.4.6.3, Equation 1). As noted previously, the limiting Mode 6 ID event is modeled using RCS minimum volume. If any portion of the connected sources of borated water were credited, the calculation of the amount of time listed in the COLR for the frequency of the RCS sampling would increase.

The RCS sampling frequency tables have changed since initial licensing for two different reasons. The first was the change to the PVNGS TS for the temperature dependent shutdown margin license amendment and the second was implementation of the CE reload improvement process. The reload

improvement process altered the way Chapter 15 Safety Analysis is performed with respect to the design of reload cores.

As mentioned in the response to NRC Request 1, the licensing history for BDAS is described in more detail in this response.

License Amendments (LAs) 23 and 13 for PVNGS Units 1 and 2, respectively, dated October 9, 1987 (ADAMS Accession number ML021690102), changed the PVNGS TS with regard to temperature dependent shutdown margin requirements, including RCS sampling requirements when BDAS is inoperable. The safety evaluation (SE) for the amendments addressed compliance with SRP Section 15.4.6 for boron dilution events. Specifically, the SE stated:

> "In support of the above proposed changes, the licensees have reevaluated the inadvertent boron dilution event, with and without the boron dilution alarm available. Standard Review Plan (SRP) Section 15.4.6 states that for such an event, a minimum of 15 minutes must be available between the time when an alarm announces an unplanned dilution and the time of loss of Shutdown Margin. The results of the licensees' reanalysis indicate that for an unplanned boron dilution from the most limiting conditions, i.e., operating three charging pumps while in Mode 5 with the reactor coolant system drained down, the time to a complete loss of shutdown margin would be 52 minutes. Therefore, there is more than the required 15 minutes available for an alarm to alert the operator before a complete loss of Shutdown Margin would occur."
>
> "In the event that the boration dilution alarm is inoperable, the results of the reanalysis have also shown that by monitoring the reactor coolant system boron concentration at the frequencies shown in revised Tables 3.1-1 through 3.1-5, the operators would have more than the required 15 minutes to take the necessary actions to mitigate the event. In the case of the most limiting event discussed above, the proposed monitoring frequency of 0.5 hours shown in Table 3.1-5 for three charging pumps operating would allow at least 22 minutes for the operators to take appropriate action."

Currently, three charging pump operation in Mode 5, with the shutdown cooling system in operation, is not permitted. This is because the time for sampling and operator action in the current analysis does not support three charging pump operation, but rather limits operation to two or less charging pumps in Mode 5 while on shutdown cooling.

The original PVNGS TS Tables 3.1-1 through 3.1-5, which specified RCS sampling frequencies, were relocated to the COLR for each PVNGS unit by LAs 69, 55 and 42 for PVNGS Units 1, 2, and 3, respectively, dated

December 30, 1992 (ADAMS Accession number ML021700483), which implemented the guidance of Generic Letter 88-16.

In LA 117, issued May 20, 1998 (ADAMS Accession number ML021720060), the BDAS TS requirements were updated to be more consistent with the format and content of the Standard Technical Specifications.

In short, BDAS alarm and alternative RCS sampling (if BDAS is inoperable) is part of the PVNGS licensing basis and complies with SRP Section 15.4.6 criteria for a boron dilution event.

NRC Request 3

The NRC staff's RAI dated January 3, 2013 (ADAMS Accession No. ML12362A292) states, in part, that

> Based on the SRP summarized above, please state whether a dropped source or component (or any other item allowed to be moved by CORE ALTERATIONS) can damage a fuel assembly or break and create a radioactive source term. If so, please provide the analysis that shows that the dose consequences of these scenarios are less limiting than the current fuel handling accident. Provide the assumptions, inputs and results of these analyses.

The APS supplement dated January 31, 2013 states, in part:

> Based on the conservative assumptions outlined above, the consequences of a dropped source or component (or any other item allowed to be moved by core alterations) are bounded by the current fuel handling accident analysis.
>
> Under routine plant operation, there are no sources present, other than used and new fuel assemblies. In the case of a prolonged shutdown, where a startup neutron source may be needed (e.g., all transuranium has decayed and there is not sufficient neutron flux to start-up using used fuel), the time since shutdown will be sufficiently long that the amount of the critical isotope present (i.e., iodine), is negligible. As a result, a drop of a source is bounded by the current fuel handling accident dose consequence analysis. Therefore, no specific analyses have been performed for other non-bounding drop events.
>
> PVNGS procedures control movement of heavy loads consistent with the current licensing basis with regard to NUREG-0612, *Control of Heavy Loads at Nuclear Power Plants.* Non-bounding load drop events

do not meet the criteria of 10 CFR 50.36, *Technical specifications,* subsection (c)(1)(ii) for inclusion in the TS limiting conditions for operation (LCOs). As a result, it is appropriate to remove TS controls for such non-bounding events, as proposed by the LAR elimination of the term CORE ALTERATION.

Currently, the term CORE ALTERATIONS would prohibit certain movement of loads over the reactor vessel if certain mitigating systems are not operable. It is unclear how the items listed as conservatisms in the RAI response would offset the relaxations proposed for mitigating systems (i.e., not requiring operable control room filtration or containment penetrations during CORE ALTERATIONS). The "conservatisms" listed in the RAI response either appear to be allowed operational parameters (i.e., uncertainty in core power, allowed time to offload (72 hours) or would likely not change the calculated dose results significantly (i.e., 4.8 weight percent versus 5.0 weight percent enrichment). Conversely, the proposed changes may have significant effects on the dose consequences. The control room filtration significantly decreases control room doses by filtering a source term and the lack of containment penetration operability can change the location of the release from containment and, therefore, change the atmospheric dispersion factors.

From a qualitative standpoint, it is not apparent how the stated assumptions would offset the increases in dose due to the proposed changes. Therefore, the staff requests a quantitative assessment of the impact of the proposed changes on the PVNGS design basis radiological analyses (i.e., control room habitability, offsite dose).

Please state whether a dropped source or component (or any other item allowed to be moved by CORE ALTERATIONS) can damage a fuel assembly or break and create a radioactive source term. If so, please provide the analysis that shows that the dose consequences of these scenarios are less limiting than the current fuel handling accident. Provide the assumptions, inputs, methodology, and results of these analyses.

Also, please state what is meant by "non-bounding" load drop events. Please explain how a drop of a load that is not a "heavy load" is determined to be bounded by a drop of a fuel assembly.

APS Response

The APS supplement dated January 31, 2013, quoted in NRC Request 3, focused upon offsite dose consequences as compared to control room doses.

The proposed LAR does not alter the existing TS requirements with regard to ensuring that control room doses remain within GDC-19 requirements and that the control room remains habitable [TS LCOs 3.3.9, *Control Room Essential Filtration Actuation Signal (CREFAS),* 3.7.11, *Control Room*

Essential Filtration System (CREFS), 3.7.12, *Control Room Emergency Air Temperature Control System (CREATCS)*].

Specifically, each of the specifications require that the accident mitigation functions of these systems be operable in Modes 1 through 6 and during movement of irradiated fuel. The proposed LAR to eliminate the term CORE ALTERATION does not alter the *Applicability* requirements of these LCOs. As a result, the proposed change does not affect the dose consequences to the control room personnel resulting from postulated design basis accidents. As will be discussed later in this response, the fuel handling accident dose consequence analysis remains bounding for other lighter loads (i.e., less than 2000 pounds). It is acknowledged, as stated in the RAI, that the control room filtration systems significantly decrease control room dose by filtering the potential accident source term.

Containment penetration controls remain in effect (TS LCO 3.9.3, *Containment Penetrations*) for movement of irradiated fuel assemblies in containment. The proposed LAR does not alter this requirement for irradiated fuel movement. As described in the supplemental response and UFSAR Section 15.7.4, no credit is taken for containment purge isolation in limiting potential dose consequences of a fuel handling accident either to the public or to the control room personnel. As a result, the location of the potential fuel handling accident release from containment and atmospheric dispersion factors are not affected by the proposed change. The fuel handling accident dose consequence analysis remains bounding for movement of sources, which will be discussed later in this response.

The proposed LAR elimination of the definition of CORE ALTERATION does not remove controls that protect against inadvertent criticality. Specifically, in preparation for refueling after the reactor head has been removed, the upper guide structure (UGS) lift rig is installed on, and bolted to, the upper guide structure. The CEA extension shafts are then latched to the UGS lift rig working platform. The working platform is raised, withdrawing the CEAs from the reactor core into the UGS. The movement of the CEAs out of the reactor core into the UGS is a CORE ALTERATION under the existing definition. Elimination of the term, however, does not remove the requirement that there be adequate shutdown margin to compensate for the CEA withdrawal (See TS LCO 3.9.1, *Boron Concentration,* greater than or equal to 3000 ppm boron, in accordance with Section 3.9.1 of the COLR).

In addition, elimination of the term CORE ALTERATION does not change the heavy loads controls that are implemented at PVNGS in response to NUREG-0612, *Control of Heavy Loads at Nuclear Power Plants*. Movement of the UGS, with the withdrawn CEAs, is not a CORE ALTERATION under the existing TS definition. This lift has been and will remain controlled as a heavy load, pursuant to the administrative controls that implement NUREG-

0612 at PVNGS, and not by the Technical Specifications. This level of administrative control is consistent with the criteria of 10 CFR 50.36(c)(1)(ii).

The PVNGS UFSAR describes a number of the administrative and physical controls that are implemented to protect against load drops over the reactor vessel during refueling operations. Specifically, Section 9.1.4.1.3, *Cask Handling Crane and Containment Polar Crane*, Design Bases E, states:

> "The containment polar crane shall be equipped with an interlock designed to prevent the trolley from carrying loads over the reactor vessel. The interlock is designed to prevent the trolley from moving within the 15 foot exclusion zone when the reactor vessel contains fuel. This interlock can be bypassed by an administratively controlled key-operated bypass switch to allow for removal and replacement of the Upper Guide Structure and Reactor Vessel Head and for movement of loads located in the area above the reactor vessel."

This Design Basis is further described in UFSAR Section 9.1.4.3.3, *Containment Polar Crane*, as follows:

> "The trolley travel exclusion zone interlock circuit is also administratively controlled by a key operated switch that can be bypassed when fuel is present for removal and replacement of the Reactor Vessel Head and the Upper Guide Structure and for movement of loads located in the area above the reactor vessel."

UFSAR Section 9.1.5.4, *Compliance with Guidelines*, describes the exclusion zones and other positive controls that implement the NUREG-0612 heavy loads controls. These positive administrative controls, in conjunction with increased reliability of the crane and associated lifting devices, make "the probability of a load handling accident extremely small."

The UFSAR requirements for heavy loads controls are implemented, in part, by procedures 72IC-9RX03, *Core Reloading*, and 30DP-9MP12, *Overhead Cranes*. Specifically, Section 3.1.2 of the core reload procedure references the overhead crane procedure for the interlock key and indicates that the key is under the control of the Shift Manager/Control Room Supervisor. Section 4.7.6.1 of the overhead crane procedure states:

> "Loads located in the area above the head can be moved only when fuel has been removed from the Reactor." ...
>
> "The only load to be carried through this interlock, after the head is removed and when fuel is present, is the Upper Guide Structure (UGS), including associated rigging." ...

> "After the UGS is removed, the Reactor Vessel (RV) area interlock will be restored and will not be bypassed until fuel is removed."

In the remote possibility that a source is required to support startup after an extended plant shutdown, such sources are installed in fuel assemblies in the spent fuel pool. The source installation tooling can only be used in the spent fuel pool and there is no tooling available onsite that can be physically used over the reactor vessel (e.g., spent fuel pool tooling is not long enough to be effectively used over the reactor vessel). Sources are light weight (significantly less than a fuel assembly) and are manufactured of source materials that do not become airborne, should the source be damaged in the spent fuel pool or the reactor vessel. As a result, the fuel handling accident dose consequence analysis remains bounding for sources.

In summary, the proposed change to eliminate the term CORE ALTERATION does not alter LCO *Applicability* requirements for accident mitigation systems that limit control room dose. Similarly, controls for movement of irradiated fuel in containment and controls of heavy loads are not altered by the proposed change. Positive TS requirements remain to protect against inadvertent criticality by ensuring ample shutdown margin. Sources are not a significant dose potential and are bounded by the existing fuel handling dose consequence analysis. Therefore, no additional quantitative dose analysis is considered necessary.

NRC Request 4

The APS response dated January 31, 2013, to NRC Request 3 stated, in part, that

> In the PVNGS reactor design, control components (CEAs) are removed from the reactor vessel with the upper guide structure, so a drop of these components is not postulated, and excluded in the definition of CORE ALTERATION.

The term CORE ALTERATION is defined in the PVNGS Technical Specifications as:

> CORE ALTERATION shall be the movement or manipulation of any fuel, sources, or reactivity control components [excluding control element assemblies (CEAs) withdrawn into the upper guide structure], within the reactor vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.

Please explain if the CEAs can be decoupled into a fuel assembly or removed from the upper guide assembly and moved over the reactor core. If so, please justify why these CEA drop scenarios are not considered.

APS Response

UFSAR Section 9.1.4.2.3.3, *Refueling Procedure*, summarizes the general description of refueling activities including the use of the UGS lift rig and how CEA maintenance is performed. Specifically:

> "The upper guide structure (UGS) lift rig is installed on, and bolted to, the upper guide structure. The CEA extension shafts are then latched to the UGS lift rig working platform. The working platform is raised, withdrawing the CEAs from the reactor core into the UGS. As the working platform is raised, refueling pool water level is also increased until the normal water level for fuel movement is achieved. The UGS is then removed from the reactor vessel and transported to the UGS laydown area in the refueling pool."

Later in the same UFSAR Section:

> "If necessary, CEAs may be relocated or replaced within the UGS. This activity may occur simultaneously with fuel movement operations. The CEAs are moved using specialized tooling and the CEA change platform. Expended CEAs are moved to the CEA elevator or other disposal location where the CEA fingers are cut and placed into a transport container. Remnant pieces of the CEA fingers, locking nuts and spiders are then removed from the refueling pool, disassembled and loaded into a disposal container. The transport container, with the cut CEA fingers in it, is transferred to the spent fuel pool using the fuel handling system where they will be stored or disposed of. New CEA fingers, spiders and locking nuts are assembled outside of the refueling pool and placed in the CEA elevator or are assembled in the CEA elevator. The new CEA assembly is then re-installed in the UGS."

The PVNGS design and operational practice is to have each of the CEAs withdrawn from the reactor core into the UGS lift rig and moved as an assembly (Procedure 31MT-9RC33, *Reactor Vessel Upper Guide Structure Removal and Installation*). CEA work is not performed in the reactor vessel, but rather is performed in the UGS laydown area in the refueling pool. Since CEAs are not decoupled into a fuel assembly or removed from the upper guide assembly and moved over the reactor core individually, CEA drop scenarios are not considered.