# PVNGS

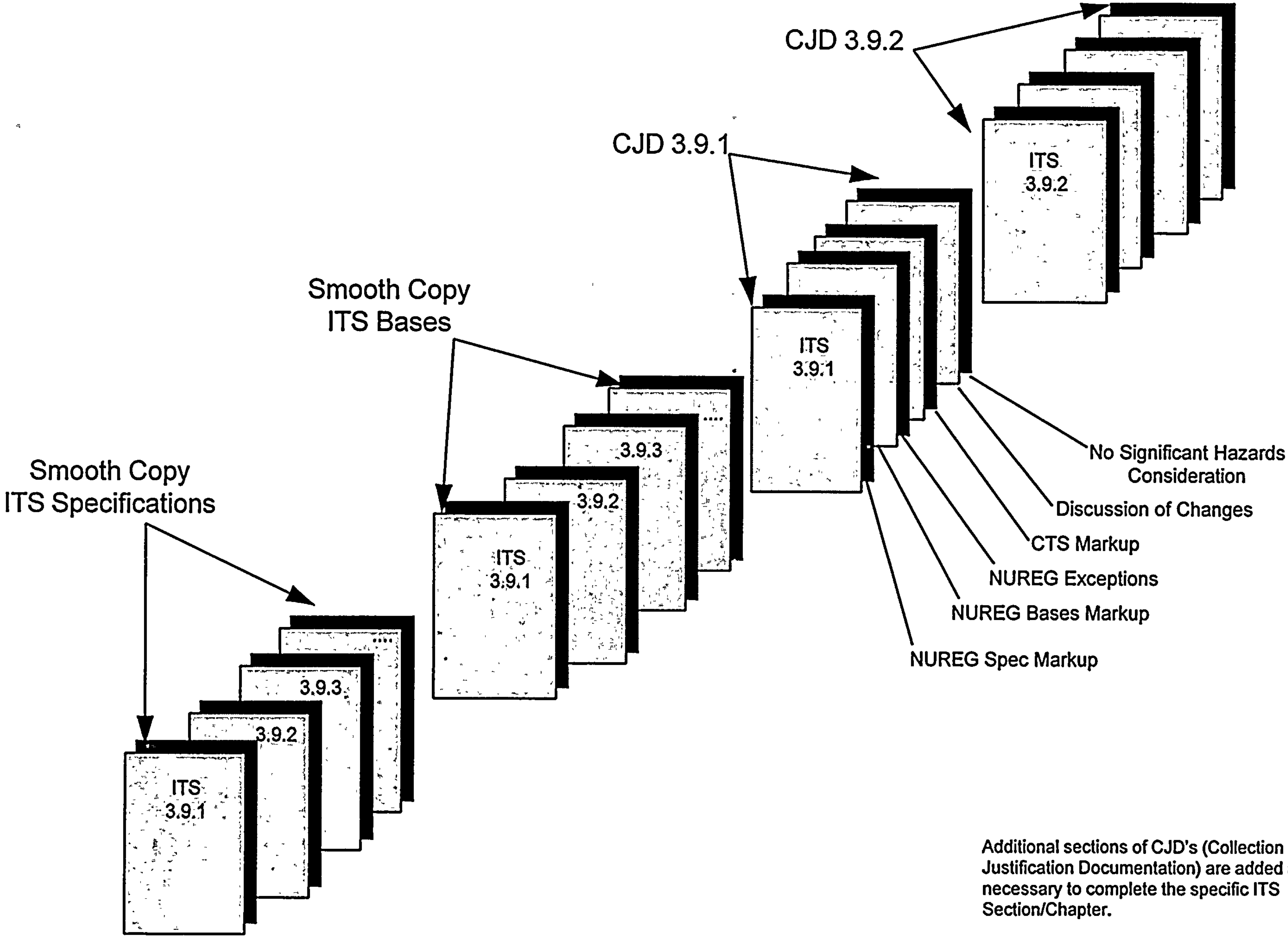
## Palo Verde Nuclear Generating Station
## Units 1, 2, and 3

ITS 3.9

# Improved Technical Specifications

# PVNGS ITS
# SECTION 3.9 - REFUELING OPERATIONS

# ITS REVIEW PACKAGE CONTENTS
## Volume 17




Additional sections of CJD's (Collection of Justification Documentation) are added as necessary to complete the specific ITS Section/Chapter.

SMOOTH COPY
ITS SECTION 3.9

# 3.9 REFUELING OPERATIONS

## 3.9.1 Boron Concentration

LCO 3.9.1 Boron concentrations of the Reactor Coolant System and the refueling canal shall be maintained within the limit specified in the COLR.

APPLICABILITY: MODE 6.

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. Boron concentration not within limit. | A.1 Suspend CORE ALTERATIONS. | Immediately |
| | AND | |
| | A.2 Suspend positive reactivity additions. | Immediately |
| | AND | |
| | A.3 Initiate action to restore boron concentration to within limit. | Immediately |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | FREQUENCY |
|---|---|
| SR 3.9.1.1 Verify boron concentration is within the limit specified in the COLR. | 72 hours |

## 3.9 REFUELING OPERATIONS

### 3.9.2 Nuclear Instrumentation

LCO 3.9.2 Two startup range monitors (SRMs) shall be OPERABLE.

APPLICABILITY: MODE 6.

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. One required SRM inoperable. | A.1 Suspend CORE ALTERATIONS.<br><br>AND<br><br>A.2 Suspend positive reactivity additions. | Immediately<br><br><br><br>Immediately |
| B. Two required SRMs inoperable. | B.1 Initiate action to restore one SRM to OPERABLE status.<br><br>AND<br><br>B.2 Perform SR 3.9.1.1. | Immediately<br><br><br><br>4 hours<br><br>AND<br><br>Once per 12 hours thereafter |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.9.2.1 | Perform CHANNEL CHECK. | 12 hours |
| SR 3.9.2.2 | -------------------NOTE-------------------<br>Neutron detectors are excluded from CHANNEL CALIBRATION.<br>--------------------------------------------<br>Perform CHANNEL CALIBRATION. | 18 months |

## 3.9 REFUELING OPERATIONS

### 3.9.3 Containment Penetrations

LCO 3.9.3 The containment penetrations shall be in the following status:

a. The equipment hatch closed and held in place by four bolts;

b. One door in each air lock closed; and

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere either:

   1. closed by a manual or automatic isolation valve, blind flange, or equivalent, or

   2. capable of being closed by an OPERABLE Containment Purge and Exhaust Isolation System.

APPLICABILITY: During CORE ALTERATIONS,
During movement of irradiated fuel assemblies within containment.

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. One or more containment penetrations not in required status. | A.1 Suspend CORE ALTERATIONS.<br><br>AND<br><br>A.2 Suspend movement of irradiated fuel assemblies within containment. | Immediately<br><br><br><br>Immediately |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.9.3.1 | Verify each required containment penetration is in the required status. | 7 days |
| SR 3.9.3.2 | Verify each required containment purge and exhaust valve actuates to the isolation position on an actual or simulated actuation signal. | 18 months |

## 3.9 REFUELING OPERATIONS

### 3.9.4 Shutdown Cooling (SDC) and Coolant Circulation—High Water Level

LCO 3.9.4 One SDC loop shall be OPERABLE and in operation.

---NOTE---
The required SDC loop may be removed from operation for ≤ 1 hour per 8 hour period, provided no operations are permitted that would cause reduction of the Reactor Coolant System boron concentration.

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APPLICABILITY: MODE 6 with the water level ≥ 23 ft above the top of reactor vessel flange.

ACTIONS

| CONDITION | REQUIRED ACTION | | COMPLETION TIME |
|---|---|---|---|
| A. SDC loop requirements not met. | A.1 | Suspend operations involving a reduction in reactor coolant boron concentration. | Immediately |
| | <u>AND</u> | | |
| | A.2 | Suspend loading irradiated fuel assemblies in the core. | Immediately |
| | <u>AND</u> | | |
| | A.3 | Initiate action to satisfy SDC loop requirements. | Immediately |
| | <u>AND</u> | | |
| | | | (continued) |

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. (continued) | A.4 Close all containment penetrations providing direct access from containment atmosphere to outside atmosphere. | 4 hours |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | FREQUENCY |
|---|---|
| SR 3.9.4.1 Verify one SDC loop is operable and in operation. | 12 hours |

## 3.9 REFUELING OPERATIONS

### 3.9.5 Shutdown Cooling (SDC) and Coolant Circulation—Low Water Level

LCO 3.9.5 Two SDC loops shall be OPERABLE, and one SDC loop shall be in operation.

-----------------------NOTE-----------------------
The required SDC loop may be removed from operation for ≤ 1 hour per 8 hour period, provided no operations are permitted that would cause reduction of the Reactor Coolant System boron concentration.
--------------------------------------------------

APPLICABILITY: MODE 6 with the water level < 23 ft above the top of reactor vessel flange.

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. One SDC loop inoperable. | A.1 Initiate action to restore SDC loop to OPERABLE status. | Immediately |
| | <u>OR</u> | |
| | A.2 Initiate action to establish ≥ 23 ft of water above the top of reactor vessel flange. | Immediately |

ACTIONS (continued)

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| B. No SDC loop OPERABLE or in operation. | B.1 Suspend operations involving a reduction in reactor coolant boron concentration. | Immediately |
| | AND | |
| | B.2 Initiate action to restore one SDC loop to OPERABLE status and to operation. | Immediately |
| | AND | |
| | B.3 Close all containment penetrations providing direct access from containment atmosphere to outside atmosphere. | 4 hours |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.9.5.1 | Verify required SDC loops are OPERABLE and one SDC loop is in operation. | 12 hours |
| SR 3.9.5.2 | Verify correct breaker alignment and indicated power available to the required SDC pump that is not in operation. | 7 days |

## 3.9 REFUELING OPERATIONS

### 3.9.6 Refueling Water Level-Fuel Assemblies

LCO 3.9.6 Refueling water level shall be maintained ≥ 23 ft above the top of the reactor vessel flange.

APPLICABILITY: During movement of fuel assemblies within containment when either the fuel assemblies being moved or the fuel assemblies seated within the reactor vessel are irradiated.

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. Refueling water level not within limit. | A.1 Suspend movement of fuel assemblies within containment. | Immediately |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.9.6.1 | Verify refueling water level is ≥ 23 ft above the top of reactor vessel flange. | 24 hours |

## 3.9 REFUELING OPERATIONS

### 3.9.7 Refueling Water Level-CEAs

LCO 3.9.7 Refueling water level shall be maintained ≥ 23 ft above the top of irradiated fuel assemblies seated within the reactor vessel.

APPLICABILITY: During movement of CEAs within the reactor vessel, when the fuel assemblies seated within the reactor vessel are irradiated.

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. Refueling water level not within limit. | A.1 Suspend movement of CEAs within the reactor vessel. | Immediately |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.9.7.1 | Verify refueling water level is ≥ 23 ft above the top of irradiated fuel assemblies seated within the reactor vessel. | 24 hours |

SMOOTH COPY
ITS SECTION 3.9 - BASES

# B 3.9 REFUELING OPERATIONS

## B 3.9.1 Boron Concentration

BASES

BACKGROUND

The limit on the boron concentrations of the Reactor Coolant System (RCS) and the refueling canal, during refueling ensures that the reactor remains subcritical during MODE 6. Refueling boron concentration is the soluble boron concentration in the coolant in each of these volumes having direct access to the reactor core during refueling.

The soluble boron concentration offsets the core reactivity and is measured by chemical analysis of a representative sample of the coolant in each of the volumes. The refueling boron concentration limit is specified in the COLR. Unit procedures ensure the specified boron concentration in order to maintain an overall core reactivity of $k_{eff} \leq 0.95$ during fuel handling, with control element assemblies (CEAs) and fuel assemblies assumed to be in the most adverse configuration (least negative reactivity) allowed by unit procedures.

GDC 26 of 10 CFR 50, Appendix A, requires that two independent reactivity control systems of different design principles be provided (Ref. 1). One of these systems must be capable of holding the reactor core subcritical under cold conditions. The Chemical and Volume Control System (CVCS) is the system capable of maintaining the reactor subcritical in cold conditions by maintaining the boron concentration.

The reactor is brought to shutdown conditions before beginning operations to open the reactor vessel for refueling. After the RCS is cooled and depressurized and the vessel head is unbolted and the head is slowly removed. The refueling canal is flooded with borated water from the refueling water tank into the open reactor vessel by gravity feeding or by the use of the Shutdown Cooling (SDC) System pumps.

(continued)

BASES

BACKGROUND (continued)

The pumping action of the SDC System in the RCS and the natural circulation due to thermal driving heads in the reactor vessel and the refueling canal mix the water to obtain a uniform concentration. The SDC System is in operation during refueling (see LCO 3.9.4, "Shutdown Cooling (SDC) and Coolant Circulation—High Water Level," and LCO 3.9.5, "Shutdown Cooling (SDC) and Coolant Circulation—Low Water Level") to provide forced circulation in the RCS and assist in maintaining the boron concentrations in the RCS and the refueling canal above the COLR limit.

APPLICABLE SAFETY ANALYSES

During refueling operations, the reactivity condition of the core is consistent with the initial conditions assumed for the boron dilution accident in the accident analysis and is conservative for MODE 6. The boron concentration limit specified in the COLR is based on the core reactivity at the beginning of each fuel cycle (the end of refueling) and includes an uncertainty allowance.

The required boron concentration and the unit refueling procedures that demonstrate the correct fuel loading plan (including full core mapping) ensure the $k_{eff}$ of the core will remain ≤ 0.95 during the refueling operation. Hence, at least a 5% Δk/k margin of safety is established during refueling.

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 limiting boron dilution accident analyzed occurs in MODE 5 (Ref. 2). A detailed discussion of this event is provided in B 3.1.2, "SHUTDOWN MARGIN—Reactor Trip Breakers Closed."

The RCS boron concentration satisfies Criterion 2 of 10 CFR 50.36 (c)(2)(ii).

(continued)

BASES

LCO

The LCO requires that a minimum boron concentration be maintained in the RCS and the refueling canal to ensure a uniform boron concentration is maintained for reactivity control in the volumes having direct access to the reactor vessel while in MODE 6. The boron concentration limit specified in the COLR ensures a core $k_{eff}$ of $\leq 0.95$ is maintained during fuel handling operations. Violation of the LCO could lead to an inadvertent criticality during MODE 6.

APPLICABILITY

This LCO is applicable in MODE 6 to ensure that the fuel in the reactor vessel will remain subcritical. The required boron concentration ensures a $k_{eff} \leq 0.95$. Above MODE 6, LCO 3.1.1, "SHUTDOWN MARGIN (SDM)—Reactor Trip Breakers Open," and LCO 3.1.2, "SHUTDOWN MARGIN—Reactor Trip Breakers Closed," ensure that an adequate amount of negative reactivity is available to shut down the reactor and to maintain it subcritical.

ACTIONS

A.1 and A.2

Continuation of CORE ALTERATIONS or positive reactivity additions (including actions to reduce boron concentration) is contingent upon maintaining the unit in compliance with the LCO. If the boron concentration of any coolant volume in the RCS or the refueling canal is less than its limit, all operations involving CORE ALTERATIONS or positive reactivity additions must be suspended immediately.

Suspension of CORE ALTERATIONS and positive reactivity additions shall not preclude moving a component to a safe position.

A.3

In addition to immediately suspending CORE ALTERATIONS or positive reactivity additions, boration to restore the concentration must be initiated immediately.

(continued)

BASES

ACTIONS

A.3 (continued)

In determining the required combination of boration flow rate and concentration, there is no unique design basis event that must be satisfied. The only requirement is to restore the boron concentration to its required value as soon as possible at greater than or equal to 26 gpm of a solution containing greater than 4000 ppm boron. 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 restoration time depends on the amount of boron that must be injected to reach the required concentration.

SURVEILLANCE REQUIREMENTS

SR 3.9.1.1

This SR ensures the coolant boron concentration in the RCS and the refueling canal is within the COLR limits. The boron concentration of the coolant in each volume is determined periodically by chemical analysis.

A minimum Frequency of once every 72 hours is therefore a reasonable amount of time to verify the boron concentration of representative samples. The Frequency is based on operating experience, which has shown 72 hours to be adequate.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 26.

2. UFSAR, Section 9.1.2.

# B 3.9 REFUELING OPERATIONS

## B 3.9.2 Nuclear Instrumentation

BASES

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BACKGROUND

The Startup Channel Neutron Flux Monitors or Startup Range Monitors (SRMs) and the associated Boron Dilution Alarm System (BDAS) are used during core alterations or movement of irradiated fuel assemblies in containment to monitor the core reactivity condition. The installed SRMs are part of the Excore Nuclear Instrumentation System. These detectors are located external to the reactor vessel and detect neutrons leaking from the core. The use of portable detectors is permitted, provided the LCO requirements are met.

The installed SRMs are BF3 detectors operating in the proportional region of the gas filled detector characteristic curve. The detectors monitor the neutron flux in counts per second. The instrument range covers five decades of neutron flux (1E+5 cps) with a 5% instrument accuracy. The detectors also provide continuous visual indication in the control room and an audible indication in the control room and containment. An audible BDAS alarm alerts operators to a possible dilution accident. The excore startup channels are designed in accordance with the criteria presented in Reference 1.

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APPLICABLE SAFETY ANALYSES

Two OPERABLE SRMs and the associated BDAS are required to provide a signal to alert the operator to unexpected changes in core reactivity such as by a boron dilution accident or an improperly loaded fuel assembly. The safety analysis of the uncontrolled boron dilution accident is described in Reference 2. The analysis of the uncontrolled boron dilution accident shows that normally available SHUTDOWN MARGIN would be reduced, but there is sufficient time for the operator to take corrective actions.

The SRMs satisfy Criterion 3 of 10 CFR 50.36 (c)(2)(ii).

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LCO

This LCO requires two SRMs OPERABLE to ensure that redundant monitoring capability is available to detect changes in core reactivity.

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(continued)

BASES

APPLICABILITY

In MODE 6, the SRMs must be OPERABLE to determine changes in core reactivity. There is no other direct means available to check core reactivity levels.

In MODES 3, 4, and 5, the installed source range detectors and circuitry are required to be OPERABLE by LCO 3.3.12, "Boron Dilution Alarm System (BDAS).

ACTIONS

A.1 and A.2

With only one SRM OPERABLE, redundancy has been lost. Since these instruments are the only direct means of monitoring core reactivity conditions, CORE ALTERATIONS and positive reactivity additions must be suspended immediately. Performance of Required Action A.1 shall not preclude completion of movement of a component to a safe position.

B.1

With no SRM OPERABLE, action to restore a monitor to OPERABLE status shall be initiated immediately. Once initiated, action shall be continued until an SRM is restored to OPERABLE status.

B.2

With no SRM OPERABLE, there is no direct means of detecting changes in core reactivity. However, since CORE ALTERATIONS and positive reactivity additions are not to be made, the core reactivity condition is stabilized until the SRMs are OPERABLE. This stabilized condition is determined by performing SR 3.9.1.1 to verify that the required boron concentration exists.

The Completion Time of 4 hours is sufficient to obtain and analyze a reactor coolant sample for boron concentration. The Frequency of once per 12 hours ensures that unplanned changes in boron concentration would be identified. The 12 hour Frequency is reasonable, considering the low probability of a change in core reactivity during this period.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.9.2.1

SR 3.9.2.1 is the performance of a CHANNEL CHECK, which is a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that the two indication channels should be consistent with core conditions. Changes in fuel loading and core geometry can result in significant differences between source range channels, but each channel should be consistent with its local conditions.

The Frequency is based on operating experience that demonstrates the rarity of channel failure. Since the probability of two random failures in redundant channels in any 12 hour period is extremely low, a CHANNEL CHECK minimizes the chance of loss of function due to failure of redundant channels.

SR 3.9.2.2

SR 3.9.2.2 is the performance of a CHANNEL CALIBRATION every 18 months. This SR is modified by a Note stating that neutron detectors are excluded from the CHANNEL CALIBRATION. The detectors are of simple construction, and any failures in the detectors will be apparent as change in channel output. The Surveillance verifies that the channel responds to a measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drift between successive calibrations to ensure that the channel remains operational. This SR is an extension of SR 3.3.12 for the Boron Dilution Alarm System CHANNEL CALIBRATION listed here because of its Applicability in these MODES. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage. Operating experience has shown these components usually pass the Surveillance when performed on the 18 month Frequency.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 13, GDC 26, GDC 28, and GDC 29.

2. UFSAR, Section 15.4.6.

# B 3.9 REFUELING OPERATIONS

## B 3.9.3 Containment Penetrations

BASES

BACKGROUND

During CORE ALTERATIONS or movement of fuel assemblies within containment with irradiated fuel in containment, a release of fission product radioactivity within the containment will be restricted from escaping to the environment when the LCO requirements are met. In MODES 1, 2, 3, and 4, this is accomplished by maintaining containment OPERABLE as described in LCO 3.6.1, "Containment." In MODE 6, the potential for containment pressurization as a result of an accident is not likely; therefore, requirements to isolate the containment from the outside atmosphere can be less stringent. The LCO requirements are referred to as "containment closure" rather than "containment OPERABILITY." Containment closure means that all potential escape paths are closed or capable of being closed. Since there is no potential for containment pressurization, the Appendix J leakage criteria and tests are not required.

The containment serves to contain fission product radioactivity that may be released from the reactor core following an accident, such that offsite radiation exposures are maintained well within the requirements of 10 CFR 100. Additionally, the containment structure provides radiation shielding from the fission products that may be present in the containment atmosphere following accident conditions.

The containment equipment hatch, which is part of the containment pressure boundary, provides a means for moving large equipment and components into and out of containment. During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, the equipment hatch must be held in place by at least four bolts. Good engineering practice dictates that the bolts required by this LCO be approximately equally spaced.

The containment air locks, which are also part of the containment pressure boundary, provide a means for personnel access during MODES 1, 2, 3, and 4 operation in accordance with LCO 3.6.2, "Containment Air Locks." Each air lock has doors at both ends. The doors are normally interlocked to prevent simultaneous opening when containment OPERABILITY is required. During periods of shutdown when containment

(continued)

BASES

BACKGROUND (continued)

closure is not required, the door interlock mechanism may be disabled, allowing both doors of an air lock to remain open for extended periods when frequent containment entry is necessary. During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, containment closure is required; therefore, the door interlock mechanism may remain disabled, but one air lock door must always remain closed.

The requirements on containment penetration closure ensure that a release of fission product radioactivity within containment will be restricted from escaping to the environment. The closure restrictions are sufficient to restrict fission product radioactivity release from containment due to a fuel handling accident during refueling.

The Containment Purge and Exhaust System includes two subsystems. The refueling purge subsystem includes a 42 inch supply penetration and a 42 inch exhaust penetration. The second subsystem, power access purge subsystem, includes an 8 inch supply penetration and an 8 inch exhaust penetration. During MODES 1, 2, 3, and 4, the two valves in each of the refueling purge supply and exhaust penetrations are secured in the closed position. The two valves in each of the two power access purge penetrations can be opened intermittently, but are closed automatically by the Engineered Safety Features Actuation System (ESFAS). Neither of the subsystems is subject to a Specification in MODE 5.

In MODE 6, large air exchanges are necessary to conduct refueling operations. The refueling purge system is used for this purpose and the valves are closed by the ESFAS in accordance with LCO 3.3.8, "Containment Purge Isolation Actuation Signal (CPIAS)."

The Power Access Purge System remains operational in MODE 6 and the valves are also closed by the ESFAS.

The other containment penetrations that provide direct access from containment atmosphere to outside atmosphere must be isolated on at least one side. Isolation may be achieved by an OPERABLE automatic isolation valve, or by a manual isolation valve, blind flange, or equivalent.

(continued)

BASES

| | |
|---|---|
| BACKGROUND (continued) | Equivalent isolation methods must be approved and may include use of devices designed to allow eddy current testing and sludge lancing of the steam generators. Devices which present a substantial restriction to the release of containment atmosphere may be considered equivalent. |
| APPLICABLE SAFETY ANALYSES | During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, the most severe radiological consequences result from a fuel handling accident. The fuel handling accident is a postulated event that involves damage to irradiated fuel (Ref. 2). Fuel handling accidents, analyzed in Reference 2, include dropping a single irradiated fuel assembly and handling tool or a heavy object onto other irradiated fuel assemblies. The requirements of LCO 3.9.6, "Refueling Water Level-Fuel Assemblies," LCO 3.9.7, "Refueling Water Level-CEAs," and the minimum decay time of 100 hours prior to CORE ALTERATIONS ensure that the release of fission product radioactivity, subsequent to a fuel handling accident, results in doses that are well within the guideline values specified in 10 CFR 100. The acceptance limits for offsite radiation exposure are contained in Standard Review Plan Section 15.7.4, Rev. 1 (Ref. 3), which defines "well within" 10 CFR 100 to be 25% or less of the 10 CFR 100 values.<br><br>Containment penetrations satisfy Criterion 3 of 10 CFR 50.36 (c)(2)(ii). |
| LCO | This LCO limits the consequences of a fuel handling accident in containment by limiting the potential escape paths for fission product radioactivity released within containment. The LCO requires any penetration providing direct access from the containment atmosphere to the outside atmosphere to be closed except for the OPERABLE containment purge supply and exhaust penetrations. For the OPERABLE containment purge supply and exhaust penetrations, this LCO ensures that these penetrations are isolable by a valve in the Containment Purge Isolation System. The OPERABILITY requirements for this LCO ensure that the automatic purge valve closure times specified in the UFSAR can be achieved and therefore meet the assumptions used in the safety analysis to ensure releases through the valves are terminated, such that the radiological doses are within the acceptance limit. |

(continued)

BASES

APPLICABILITY

The containment penetration requirements are applicable during CORE ALTERATIONS or movement of irradiated fuel assemblies within containment because this is when there is a potential for a fuel handling accident. In MODES 1, 2, 3, and 4, containment penetration requirements are addressed by LCO 3.6.1, "Containment." In MODES 5 and 6, when CORE ALTERATIONS or movement of irradiated fuel assemblies within containment are not being conducted, the potential for a fuel handling accident does not exist. Therefore, under these conditions no requirements are placed on containment penetration status.

ACTIONS

A.1 and A.2

With the containment equipment hatch, air locks, or any containment penetration that provides direct access from the containment atmosphere to the outside atmosphere not in the required status, including the Containment Purge Isolation System not capable of automatic actuation when the purge valves are open, the unit must be placed in a condition in which the isolation function is not needed. This is accomplished by immediately suspending CORE ALTERATIONS and movement of irradiated fuel assemblies within containment. Performance of these actions shall not preclude completion of movement of a component to a safe position.

SURVEILLANCE REQUIREMENTS

SR 3.9.3.1

This Surveillance demonstrates that each of the containment penetrations required to be in its closed position is in that position. The Surveillance on the open purge and exhaust valves will demonstrate that the valves are not blocked from closing. Also, the Surveillance will demonstrate that each valve operator has motive power, which will ensure each valve is capable of being closed by an OPERABLE automatic containment purge isolation signal.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.9.3.1 (continued)

The Surveillance is performed every 7 days during CORE ALTERATIONS or movement of irradiated fuel assemblies within the containment. The Surveillance interval is selected to be commensurate with the normal duration of time to complete fuel handling operations. A surveillance before the start of refueling operations will provide two or three surveillance verifications during the applicable period for this LCO. As such, this Surveillance ensures that a postulated fuel handling accident that releases fission product radioactivity within the containment will not result in a release of fission product radioactivity to the environment.

SR 3.9.3.2

This Surveillance demonstrates that each containment purge valve actuates to its isolation position on manual initiation or on an actual or simulated high radiation signal. The 18 month Frequency maintains consistency with other similar ESFAS instrumentation and valve testing requirements. The CPIAS is tested in accordance with LCO 3.3.8, "Containment Purge Isolation Actuation Signal (CPIAS)." SR 3.6.3.5 demonstrates that the isolation time of each valve is in accordance with the Inservice Testing Program requirements. These surveillances performed during MODE 6 will ensure that the valves are capable of closing after a postulated fuel handling accident to limit a release of fission product radioactivity from the containment.

REFERENCES

1. GPU Nuclear Safety Evaluation SE-0002000-001, Rev. 0, May 20, 1988.
2. UFSAR, Section 15.7.4.
3. NUREG-0800, Section 15.7.4, Rev. 1, July 1981.

(continued)

# B 3.9 REFUELING OPERATIONS

## B 3.9.4 Shutdown Cooling (SDC) and Coolant Circulation—High Water Level

BASES

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BACKGROUND

The purposes of the SDC System in MODE 6 are to remove decay heat and sensible heat from the Reactor Coolant System (RCS), as required by GDC 34, to provide mixing of borated coolant, to provide sufficient coolant circulation to minimize the effects of a boron dilution accident, and to prevent boron stratification (Ref. 1). Heat is removed from the RCS by circulating reactor coolant through the SDC heat exchanger(s), where the heat is transferred to the Essential Cooling Water System via the SDC heat exchanger(s). The coolant is then returned to the RCS via the RCS cold leg(s). Operation of the SDC System for normal cooldown or decay heat removal is manually accomplished from the control room. The heat removal rate is adjusted by controlling the flow of reactor coolant through the SDC heat exchanger(s) and bypassing the heat exchanger(s). Mixing of the reactor coolant is maintained by this continuous circulation of reactor coolant through the SDC System.

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APPLICABLE SAFETY ANALYSES

If the reactor coolant temperature is not maintained below 200°F, boiling of the reactor coolant could result. This could lead to inadequate cooling of the reactor fuel due to a resulting loss of coolant in the reactor vessel. Additionally, boiling of the reactor coolant could lead to a reduction in boron concentration in the coolant due to the boron plating out on components near the areas of the boiling activity, and because of the possible addition of water to the reactor vessel with a lower boron concentration than is required to keep the reactor subcritical. The loss of reactor coolant and the reduction of boron concentration in the reactor coolant would eventually challenge the integrity of the fuel cladding, which is a fission product barrier. One train of the SDC System is required to be operational in MODE 6, with the water level ≥ 23 ft above the top of the reactor vessel flange, to prevent this challenge. The LCO does permit de-energizing of the SDC pump for short durations under the condition that the boron concentration is not diluted. This conditional de-energizing of the SDC pump does not result in a challenge to the fission product barrier. SDC and Coolant Circulation—High Water Level satisfies Criterion 2 of 10 CFR 50.36 (c)(2)(ii).

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(continued)

BASES

LCO

Only one SDC loop is required for decay heat removal in MODE 6, with water level ≥ 23 ft above the top of the reactor vessel flange. Only one SDC loop is required because the volume of water above the reactor vessel flange provides backup decay heat removal capability. At least one SDC loop must be in operation to provide:

a. Removal of decay heat;

b. Mixing of borated coolant to minimize the possibility of a criticality; and

c. Indication of reactor coolant temperature.

An OPERABLE SDC loop includes an SDC pump, a heat exchanger, valves, piping, instruments, and controls to ensure an OPERABLE flow path and to determine the low end temperature. The flow path starts in one of the RCS hot legs and is returned to the RCS cold legs.

The LCO is modified by a Note that allows the required operating SDC loop to be removed from service for up to 1 hour in each 8 hour period, provided no operations are permitted that would cause a reduction of the RCS boron concentration. Boron concentration reduction is prohibited because uniform concentration distribution cannot be ensured without forced circulation. This permits operations such as core mapping or alterations in the vicinity of the reactor vessel hot leg nozzles, and RCS to SDC isolation valve testing. During this 1 hour period, decay heat is removed by natural convection to the large mass of water in the refueling cavity.

APPLICABILITY

One SDC loop must be in operation in MODE 6, with the water level ≥ 23 ft above the top of the reactor vessel flange, to provide decay heat removal. The 23 ft level was selected because it corresponds to the 23 ft requirement established for fuel movement in LCO 3.9.6, "Refueling Water Level - Fuel Assemblies."

(continued)

BASES

APPLICABILITY (continued)

Requirements for the SDC System in other MODES are covered by LCOs in Section 3.4, Reactor Coolant System (RCS), and Section 3.5, Emergency Core Cooling Systems (ECCS). SDC loop requirements in MODE 6, with the water level < 23 ft above the top of the reactor vessel flange, are located in LCO 3.9.5, "Shutdown Cooling (SDC) and Coolant Circulation—Low Water Level."

ACTIONS

SDC loop requirements are met by having one SDC loop OPERABLE and in operation, except as permitted in the Note to the LCO.

A.1

If SDC loop requirements are not met, there will be no forced circulation to provide mixing to establish uniform boron concentrations. Reduced boron concentrations can occur through the addition of water with a lower boron concentration than that contained in the RCS. Therefore, actions that reduce boron concentration shall be suspended immediately.

A.2

If SDC loop requirements are not met, actions shall be taken immediately to suspend loading irradiated fuel assemblies in the core. With no forced circulation cooling, decay heat removal from the core occurs by natural convection to the heat sink provided by the water above the core. A minimum refueling water level of 23 ft above the reactor vessel flange provides an adequate available heat sink. Suspending any operation that would increase the decay heat load, such as loading an irradiated fuel assembly, is a prudent action under this condition.

A.3

If SDC loop requirements are not met, actions shall be initiated and continued in order to satisfy SDC loop requirements.

(continued)

BASES

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ACTIONS (continued)

A.4

If SDC loop requirements are not met, all containment penetrations to the outside atmosphere must be closed to prevent fission products, if released by a loss of decay heat event, from escaping the containment building. The 4 hour Completion Time allows fixing most SDC problems without incurring the additional action of violating the containment atmosphere.

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SURVEILLANCE REQUIREMENTS

SR 3.9.4.1

This Surveillance demonstrates that the SDC loop is in operation and circulating reactor coolant at a flowrate of greater than or equal to 3780 gpm. The flow rate is determined by the flow rate necessary to provide sufficient decay heat removal capability and to prevent thermal and boron stratification in the core. The Frequency of 12 hours is sufficient, considering the flow, temperature, pump control, and alarm indications available to the operator in the control room for monitoring the SDC System.

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REFERENCES

1. UFSAR, Section 5.4.7.

# B 3.9 REFUELING OPERATIONS

## B 3.9.5 Shutdown Cooling (SDC) and Coolant Circulation—Low Water Level

BASES

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BACKGROUND

The purposes of the SDC System in MODE 6 are to remove decay heat and sensible heat from the Reactor Coolant System (RCS), as required by GDC 34, to provide mixing of borated coolant, to provide sufficient coolant circulation to minimize the effects of a boron dilution accident, and to prevent boron stratification (Ref. 1). Heat is removed from the RCS by circulating reactor coolant through the SDC heat exchanger(s), where the heat is transferred to the Essential Cooling Water System via the SDC heat exchanger(s). The coolant is then returned to the RCS via the RCS cold leg(s). Operation of the SDC System for normal cooldown or decay heat removal is manually accomplished from the control room. The heat removal rate is adjusted by controlling the flow of reactor coolant through the SDC heat exchanger(s) and bypassing the heat exchanger(s). Mixing of the reactor coolant is maintained by this continuous circulation of reactor coolant through the SDC System.

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APPLICABLE SAFETY ANALYSES

If the reactor coolant temperature is not maintained below 200°F, boiling of the reactor coolant could result. This could lead to inadequate cooling of the reactor fuel due to the resulting loss of coolant in the reactor vessel. Additionally, boiling of the reactor coolant could lead to a reduction in boron concentration in the coolant due to the boron plating out on components near the areas of the boiling activity, and because of the possible addition of water to the reactor vessel with a lower boron concentration than is required to keep the reactor subcritical. The loss of reactor coolant and the reduction of boron concentration in the reactor coolant would eventually challenge the integrity of the fuel cladding, which is a fission product barrier. Two trains of the SDC System are required to be OPERABLE, and one train is required to be in operation in MODE 6, with the water level < 23 ft above the top of the reactor vessel flange, to prevent this challenge.

SDC and Coolant Circulation—Low Water Level satisfies Criterion 2 of 10 CFR 50.36 (c)(2)(ii).

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(continued)

BASES

LCO

In MODE 6, with the water level < 23 ft above the top of the reactor vessel flange, both SDC loops must be OPERABLE. Additionally, one loop of the SDC System must be in operation in order to provide:

a. Removal of decay heat;

b. Mixing of borated coolant to minimize the possibility of a criticality; and

c. Indication of reactor coolant temperature.

An OPERABLE SDC loop consists of an SDC pump, a heat exchanger, valves, piping, instruments, and controls to ensure an OPERABLE flow path and to determine the low end temperature. The flow path starts in one of the RCS hot legs and is returned to the RCS cold legs.

Both SDC pumps may be aligned to the Refueling Water Tank (RWT) to support filling the refueling cavity or for performance of required testing.

The LCO is modified by a Note that allows a required operating SDC loop to be removed from service for up to 1 hour in each 8 hour period, provided no operations are permitted that would cause a reduction of the RCS boron concentration. Boron concentration reduction is prohibited because uniform concentration distribution cannot be ensured without forced circulation. This permits operations such as core mapping or alterations in the vicinity of the reactor vessel hot leg nozzles, and RCS to SDC isolation valve testing. During this 1 hour period, decay heat is removed by natural convection to the large mass of water in the refueling cavity.

APPLICABILITY

Two SDC loops are required to be OPERABLE, and one SDC loop must be in operation in MODE 6, with the water level < 23 ft above the top of the reactor vessel flange, to provide decay heat removal. Requirements for the SDC System in other MODES are covered by LCOs in Section 3.4, Reactor Coolant System. MODE 6 requirements, with a water level ≥ 23 ft above the reactor vessel flange, are covered in LCO 3.9.4, "Shutdown Cooling and Coolant Circulation—High Water Level."

(continued)

BASES

ACTIONS

A.1 and A.2

If one SDC loop is inoperable, action shall be immediately initiated and continued until the SDC loop is restored to OPERABLE status and to operation, or until ≥ 23 ft of water level is established above the reactor vessel flange. When the water level is established at ≥ 23 ft above the reactor vessel flange, the Applicability will change to that of LCO 3.9.4, "Shutdown Cooling and Coolant Circulation—High Water Level," and only one SDC loop is required to be OPERABLE and in operation. An immediate Completion Time is necessary for an operator to initiate corrective actions.

B.1

If no SDC loop is in operation or no SDC loops are OPERABLE, there will be no forced circulation to provide mixing to establish uniform boron concentrations. Reduced boron concentrations can occur by the addition of water with lower boron concentration than that contained in the RCS. Therefore, actions that reduce boron concentration shall be suspended immediately.

B.2

If no SDC loop is in operation or no SDC loops are OPERABLE, action shall be initiated immediately and continued without interruption to restore one SDC loop to OPERABLE status and operation. Since the unit is in Conditions A and B concurrently, the restoration of two OPERABLE SDC loops and one operating SDC loop should be accomplished expeditiously.

B.3

If no SDC loop is in operation or no SDC loops are OPERABLE, all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere must be closed within 4 hours. With the SDC loop requirements not met, the potential exists for the coolant to boil and release radioactive gas to the containment atmosphere. Closing containment penetrations that are open to the outside atmosphere ensures that dose limits are not exceeded.

The Completion Time of 4 hours is reasonable, based on the low probability of the coolant boiling in that time.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.9.5.1

This Surveillance demonstrates that one SDC loop is operating and circulating reactor coolant at a flowrate of greater than or equal to 3780 gpm. The flow rate is determined by the flow rate necessary to provide sufficient decay heat removal capability and to prevent thermal and boron stratification in the core. In addition, this Surveillance demonstrates that the other SDC loop is OPERABLE.

In addition, during operation of the SDC loop with the water level in the vicinity of the reactor vessel nozzles, the SDC loop flow rate determination must also consider the SDC pump suction requirements. The Frequency of 12 hours is sufficient, considering the flow, temperature, pump control, and alarm indications available to the operator to monitor the SDC System in the control room.

Verification that the required loops are OPERABLE and in operation ensures that loops can be placed in operation as needed, to maintain decay heat and retain forced circulation. The Frequency of 12 hours is considered reasonable, since other administrative controls are available and have proven to be acceptable by operating experience.

SR 3.9.5.2

Verification that the required pump that is not in operation is OPERABLE ensures that an additional SDC pump can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available to the required pump. The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.

REFERENCES

1. UFSAR, Section 5.4.7.

# B 3.9 REFUELING OPERATIONS

## B 3.9.6 Refueling Water Level-Fuel Assemblies

BASES

| | |
|---|---|
| BACKGROUND | The movement of fuel assemblies within containment requires a minimum water level of 23 ft above the top of the reactor vessel flange when either the fuel assemblies being moved or the fuel assemblies seated within the reactor vessel are irradiated. During refueling this maintains sufficient water level in the refueling canal, the fuel transfer canal, the refueling cavity, and the spent fuel pool. Sufficient water is necessary to retain iodine fission product activity in the water in the event of a fuel handling accident (Refs. 1 and 2). Sufficient iodine activity would be retained to limit offsite doses from the accident to < 33% of 10 CFR 100 limits, which meets the intent of the guidance of Reference 3. |
| APPLICABLE SAFETY ANALYSES | During movement of fuel assemblies, the water level in the refueling canal and refueling cavity is an initial condition design parameter in the analysis of the fuel handling accident in containment postulated by Regulatory Guide 1.25 (Ref. 1). A minimum water level of 23 ft (Regulatory Position C.1.c of Ref. 1) allows a decontamination factor of 100 (Regulatory Position C.1.g of Ref. 1) to be used in the accident analysis for iodine. This relates to the assumption that 99% of the total iodine released from the pellet to cladding gap of all the dropped fuel assembly rods is retained by the refueling cavity water. The fuel pellet to cladding gap is assumed to contain 10% of the total fuel rod iodine inventory (Ref. 1).<br><br>The fuel handling accident analysis inside containment is described in Reference 2. With a minimum water level of 23 ft and a minimum decay time of 100 hours prior to fuel handling, the analysis and test programs demonstrate that the iodine release due to a postulated fuel handling accident is adequately captured by the water and offsite doses are maintained within allowable limits (Ref. 4).<br><br>Refueling water level satisfies Criterion 2 of 10 CFR 50.36 (c)(2)(ii). |

(continued)

BASES

LCO

A minimum refueling water level of 23 ft above the reactor vessel flange is required to ensure that the radiological consequences of a postulated fuel handling accident inside containment are within acceptable limits as provided by the guidance of Reference 3.

APPLICABILITY

LCO 3.9.6 is applicable when moving fuel assemblies within containment when either the fuel assemblies being moved or the fuel assemblies seated in the reactor vessel are irradiated. The LCO minimizes the possibility of a fuel handling accident in containment that is beyond the assumptions of the safety analysis. If irradiated fuel is not present in containment, there can be no significant radioactivity release as a result of a postulated fuel handling accident. Requirements for fuel handling accidents in the spent fuel pool are covered by LCO 3.7.14, "Fuel Storage Pool Water Level."

ACTIONS

A.1

With a water level of < 23 ft above the top of the reactor vessel flange, all operations involving movement of fuel assemblies shall be suspended immediately to ensure that a fuel handling accident cannot occur.

The suspension of fuel movement shall not preclude completion of movement of a component to a safe position.

SURVEILLANCE REQUIREMENTS

SR 3.9.6.1

Verification of a minimum water level of 23 ft above the top of the reactor vessel flange ensures that the design basis for the postulated fuel handling accident analysis during refueling operations is met. Water at the required level above the top of the reactor vessel flange limits the consequences of damaged fuel rods that are postulated to result from a fuel handling accident inside containment (Ref. 2).

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.9.6.1 (continued)

The Frequency of 24 hours is based on engineering judgment and is considered adequate in view of the large volume of water and the normal procedural controls of valve positions, and HVAC operations which make significant unplanned level changes unlikely.

REFERENCES

1. Regulatory Guide 1.25, March 23, 1972.
2. UFSAR, Section 15.7.4.
3. NUREG-0800, Section 15.7.4.
4. 10 CFR 100.10.

# B 3.9 REFUELING OPERATIONS

## B 3.9.7 Refueling Water Level - CEAs

BASES

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BACKGROUND

The movement of CEAs within the reactor vessel, when irradiated fuel assemblies are seated in the reactor vessel requires a minimum water level of 23 ft above the top of the irradiated fuel. During refueling this maintains sufficient water level in the refueling canal, the fuel transfer canal, the refueling cavity, and the spent fuel pool. Sufficient water is necessary to retain iodine fission product activity in the water in the event of a fuel handling accident (Refs. 1 and 2). Sufficient iodine activity would be retained to limit offsite doses from the accident to < 33% of 10 CFR 100 limits, which meets the intent of the guidance of Reference 3.

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APPLICABLE SAFETY ANALYSES

During movement of CEA's the water level in the refueling canal and refueling cavity is an initial condition design parameter in the analysis of the fuel handling accident in containment postulated by Regulatory Guide 1.25 (Ref. 1). A minimum water level of 23 ft (Regulatory Position C.1.c of Ref. 1) allows a decontamination factor of 100 (Regulatory Position C.1.g of Ref. 1) to be used in the accident analysis for iodine. This relates to the assumption that 99% of the total iodine released from the pellet to cladding gap of all the dropped fuel assembly rods is retained by the refueling cavity water. The fuel pellet to cladding gap is assumed to contain 10% of the total fuel rod iodine inventory (Ref. 1).

The fuel handling accident analysis inside containment is described in Reference 2. With a minimum water level of 23 ft and a minimum decay time of 100 hours prior to fuel handling, the analysis and test programs demonstrate that the iodine release due to a postulated fuel handling accident is adequately captured by the water and offsite doses are maintained within allowable limits (Ref. 4).

Refueling water level satisfies Criterion 2 of 10 CFR 50.36 (c)(2)(ii).

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(continued)

BASES

LCO

A minimum refueling water level of 23 ft above irradiated assemblies seated within the reactor vessel is required to ensure that the radiological consequences of a postulated fuel handling accident inside containment are within acceptable limits as provided by the guidance of Reference 3.

APPLICABILITY

LCO 3.9.7 is applicable during movement of CEAs within the reactor vessel when irradiated fuel assemblies are seated within the reactor vessel. The LCO minimizes the possibility of a fuel handling accident in containment that is beyond the assumptions of the safety analysis. If irradiated fuel is not present in containment, there can be no significant radioactivity release as a result of a postulated fuel handling accident. Requirements for fuel handling accidents in the spent fuel pool are covered by LCO 3.7.14, "Fuel Storage Pool Water Level."

ACTIONS

A.1

With a water level of < 23 ft above the top of irradiated fuel assemblies seated within the reactor vessel, all operations involving movement of CEAs within the reactor vessel shall be suspended immediately to ensure that a fuel handling accident cannot occur.

The suspension of movement of CEAs shall not preclude completion of movement of a component to a safe position.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.9.7.1

Verification of a minimum water level of 23 ft above the top of irradiated fuel assemblies seated within the reactor vessel ensures that the design basis for the postulated fuel handling accident analysis during refueling operations is met. Water at the required level above the top of the irradiated fuel limits the consequences of damaged fuel rods that are postulated to result from a fuel handling accident inside containment (Ref. 2).

The Frequency of 24 hours is based on engineering judgment and is considered adequate in view of the large volume of water and the normal procedural controls of valve positions and HVAC operations, which make significant unplanned level changes unlikely.

REFERENCES

1. Regulatory Guide 1.25, March 23, 1972.
2. UFSAR, Section 15.7.4.
3. NUREG-0800, Section 15.7.4.
4. 10 CFR 100.10.

CE STS
NUREG-1432 REV. 1
SPECIFICATION 3.9.1
MARK UP

<CTS>
<DOC>

## 3.9 REFUELING OPERATIONS

### 3.9.1 Boron Concentration

<3.9.1> LCO 3.9.1 Boron concentrations of the Reactor Coolant System~~,~~ {and} the refueling canal~~, and the refueling cavity~~ shall be maintained within the limit specified in the COLR. 2

APPLICABILITY: MODE 6.

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| <3.9.1ACT> A. Boron concentration not within limit. | A.1 Suspend CORE ALTERATIONS. | Immediately |
| | **AND** | |
| | A.2 Suspend positive reactivity additions. | Immediately |
| | **AND** | |
| | A.3 Initiate action to restore boron concentration to within limit. | Immediately |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | FREQUENCY |
|---|---|
| <4.9.1.2> SR 3.9.1.1 Verify boron concentration is within the limit specified in the COLR. | 72 hours |

CE STS
NUREG-1432 REV. 1
SPECIFICATION 3.9.1
BASES MARK UP

## B 3.9 REFUELING OPERATIONS

### B 3.9.1 Boron Concentration

BASES

BACKGROUND

and3

The limit on the boron concentrations of the Reactor Coolant System (RCS), the refueling canal~~, and refueling cavity~~ (2) during refueling ensures that the reactor remains subcritical during MODE 6. Refueling boron concentration is the soluble boron concentration in the coolant in each of these volumes having direct access to the reactor core during refueling.

The soluble boron concentration offsets the core reactivity and is measured by chemical analysis of a representative sample of the coolant in each of the volumes. The refueling boron concentration limit is specified in the COLR. Unit procedures ensure the specified boron concentration in order to maintain an overall core reactivity of $k_{eff} \leq 0.95$ during fuel handling, with control element assemblies (CEAs) and fuel assemblies assumed to be in the most adverse configuration (least negative reactivity) allowed by unit procedures.

GDC 26 of 10 CFR 50, Appendix A, requires that two independent reactivity control systems of different design principles be provided (Ref. 1). One of these systems must be capable of holding the reactor core subcritical under cold conditions. The Chemical and Volume Control System (CVCS) is the system capable of maintaining the reactor subcritical in cold conditions by maintaining the boron concentration.

and

The reactor is brought to shutdown conditions before beginning operations to open the reactor vessel for refueling. After the RCS is cooled and depressurized and the vessel head is unbolted, the head is slowly removed ~~to~~ (2) ~~form the refueling cavity~~. The refueling canal ~~and the~~ is ~~refueling cavity are~~ then flooded with borated water from the refueling water tank into the open reactor vessel by gravity feeding or by the use of the Shutdown Cooling (SDC) System pumps.

The pumping action of the SDC System in the RCS and the natural circulation due to thermal driving heads in the (2) reactor vessel and the refueling ~~cavity~~ mix ~~the added~~
Canal

(continued)

BASES

BACKGROUND (continued)

to obtain a Uniform Concentration

(2)

~~concentrated boric acid with~~ the water ~~in the refueling canal~~. The SDC System is in operation during refueling (see LCO 3.9.4, "Shutdown Cooling (SDC) and Coolant Circulation—High Water Level," and LCO 3.9.5, "Shutdown Cooling (SDC) and Coolant Circulation—Low Water Level") to provide forced circulation in the RCS and assist in maintaining the boron concentrations in the RCS, and the refueling canal ~~and the refueling cavity~~ above the COLR limit.

APPLICABLE SAFETY ANALYSES

During refueling operations, the reactivity condition of the core is consistent with the initial conditions assumed for the boron dilution accident in the accident analysis and is conservative for MODE 6. The boron concentration limit specified in the COLR is based on the core reactivity at the beginning of each fuel cycle (the end of refueling) and includes an uncertainty allowance.

The required boron concentration and the unit refueling procedures that demonstrate the correct fuel loading plan (including full core mapping) ensure the $k_{eff}$ of the core will remain ≤ 0.95 during the refueling operation. Hence, at least a 5% Δk/k margin of safety is established during refueling.

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

(2)

The limiting boron dilution accident analyzed occurs in MODE 5 (Ref. 2). A detailed discussion of this event is provided in B 3.1.2, "SHUTDOWN MARGIN—~~$T_{avg}$ ≤ 200°F~~." Reactor Trip Breakers Open

The RCS boron concentration satisfies Criterion 2 of ~~the NRC Policy Statement~~. 10CFR 50.36(c)(2)(ii) (1)

LCO

The LCO requires that a and minimum boron concentration be maintained in the RCS, and the refueling canal ~~and refueling cavity~~ while in MODE 6. The boron concentration limit specified in the COLR ensures a core $k_{eff}$ of ≤ 0.95 is (2) (3)

to ensure a uniform boron concentration is maintained for reactivity control in the volumes having direct access to the reactor vessel

(continued)

BASES

LCO (continued)

maintained during fuel handling operations. Violation of the LCO could lead to an inadvertent criticality during MODE 6.

APPLICABILITY

(2)

This LCO is applicable in MODE 6 to ensure that the fuel in the reactor vessel will remain subcritical. The required boron concentration ensures a $k_{eff} \leq 0.95$. Above MODE 6, LCO 3.1.1, "SHUTDOWN MARGIN (SDM) ~~$T_{avg} > 200°F$~~ Reactor Trip Breakers Open," and LCO 3.1.2, "SHUTDOWN MARGIN ~~$T_{avg} \leq 200°F$~~ Reactor Trip Breakers Closed," ensure that an adequate amount of negative reactivity is available to shut down the reactor and to maintain it subcritical.

ACTIONS

A.1 and A.2

Continuation of CORE ALTERATIONS or positive reactivity additions (including actions to reduce boron concentration) is contingent upon maintaining the unit in compliance with the LCO. If the boron concentration of any coolant volume in the RCS or the refueling canal ~~or the refueling cavity~~ is less than its limit, all operations involving CORE ALTERATIONS or positive reactivity additions must be suspended immediately. (2)

Suspension of CORE ALTERATIONS and positive reactivity additions shall not preclude moving a component to a safe position.

A.3

In addition to immediately suspending CORE ALTERATIONS or positive reactivity additions, boration to restore the concentration must be initiated immediately.

In determining the required combination of boration flow rate and concentration, there is no unique design basis event that must be satisfied. The only requirement is to restore the boron concentration to its required value as soon as possible at greater than or equal to 26 gpm of a solution containing greater than 4000 ppm boron. (2) In order to raise the boron concentration as soon as possible, the operator should begin boration with the best source available for unit conditions.

(continued)

BASES

ACTIONS

A.3 (continued)

Once boration is initiated, it must be continued until the boron concentration is restored. The restoration time depends on the amount of boron that must be injected to reach the required concentration.

SURVEILLANCE REQUIREMENTS

SR 3.9.1.1

This SR ensures the coolant boron concentration in the RCS, and the refueling canal ~~and the refueling cavity~~ is within the COLR limits. The boron concentration of the coolant in each volume is determined periodically by chemical analysis. 2

A minimum Frequency of once every 72 hours is therefore a reasonable amount of time to verify the boron concentration of representative samples. The Frequency is based on operating experience, which has shown 72 hours to be adequate.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 26.
2. FSAR, Section ~~[ ]~~ 9.1.2 1 .

# NUREG-1432 EXCEPTIONS
# SPECIFICATION 3.9.1

1. Grammar and/or editorial changes have been made to enhance clarity. No technical or intent changes to the Specification are made by this change.

2. The plant specific titles, nomenclature, number parameter/value, reference, system description, system design, operating practices or analysis description was used (additions, deletions, and/or changes are included). Plant specific parameters/values were directly transferred from the CTS to the ITS, or from the plant design basis to the ITS. The Bases have been revised to be consistent with the LCO/Surveillance.

3. NUREG 1432, Specification 3.9.1, "Boron Concentration" requires the boron concentration be taken of the Reactor Coolant System and the refueling canal, and the refueling cavity. The terms "refueling canal" and "refueling cavity" are bracketed terms in the NUREG. The PVNGS CTS does not use the term "refueling cavity" in Specification 3.9.1, "Boron Concentration" since it is not applicable to the PVNGS design.

   The PVNGS ITS LCO for Specification 3.9.1 requires the boron concentration of the Reactor Coolant System and the refueling canal shall be within the limit specified in the COLR. This ensures a uniform boron concentration is maintained for reactivity control in the water volumes having direct access to the reactor vessel during Mode 6.

PVNGS CTS
SPECIFICATION 3.9.1
MARK UP

A.1

3.9 ~~3/4.9~~ REFUELING OPERATIONS

3.9.1 ~~3/4.9.1~~ BORON CONCENTRATION

~~LIMITING CONDITION FOR OPERATION~~

A.2

LCO3.9.1 ~~3.9.1~~ ~~With the reactor vessel head closure bolts less than fully tensioned or with the head removed, the~~ boron concentration of ~~all filled portions of~~ the Reactor Coolant System and the refueling canal shall be maintained ~~uniform and~~ within the limit specified in the Core Operating Limits Report (COLR). A.4

APPLICABILITY: MODE 6~~*~~ A.2

ACTION: and A.1

ACT A. With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS ~~or~~ and positive reactivity changes and initiate and continue boration ~~at greater than or equal to 26 gpm of a solution containing > 4000 ppm boron or its equivalent~~ until the boron concentration is within limits. LA.1

SURVEILLANCE REQUIREMENTS

~~4.9.1.1 The boron concentration shall be determined to be within the limit specified in the COLR prior to:~~ LA.2

~~a. Removing or unbolting the reactor vessel head, and~~

~~b. Withdrawal of any full-length CEA in excess of 3 feet from its fully inserted position within the reactor pressure vessel.~~

SR3.9.1.1 ~~4.9.1.2~~ The boron concentration ~~of the Reactor Coolant System and the refueling canal~~ shall be ~~determined by chemical analysis at least~~ verified within the limits specified in the COLR once per 72 hours. A.5

A.2

~~*The reactor shall be maintained in MODE 6 whenever fuel is in the reactor vessel with the reactor vessel head closure bolts less than fully tensioned or with the head removed.~~

DISCUSSION OF CHANGES
SPECIFICATION 3.9.1

ADMINISTRATIVE CHANGES

A.1 All reformatting and renumbering is in accordance with the Combustion Engineering Plant (CEOG) Standard Technical Specifications NUREG-1432, Rev. 1 (NUREG-1432). As a result, the Palo Verde Nuclear Generating Station (PVNGS) Improved Technical Specifications (ITS) should be more readable, and therefore understandable, by plant operators as well as other users. During the reformatting and renumbering of the ITS, no technical changes (either actual or interpretational) to the Current Technical Specification (CTS) were made unless they were identified and justified.

Editorial rewording (either adding or deleting) is made consistent with NUREG-1432. During NUREG-1432 development, certain wording preferences or English language conventions were adopted which resulted in no technical changes (either actual or interpretational) to the CTS.

Additional information has also been added to more fully describe each subsection. This wording is consistent with NUREG-1432. Since the design is already approved by the NRC, adding more detail does not result in a technical change.

A.2 CTS 3/4.9.1 is Applicable in Mode 6 yet states that the conditions specified apply with the reactor vessel head bolts less than fully tensioned or with the head removed. ITS Definition Table 1.1-1 defines Mode 6 as Refueling with one or more reactor vessel head closure bolts less than fully tightened; therefore, ITS 3.9.1 does not restate the requirements that the reactor vessel head bolts be less than fully tensioned or the head removed. Omitting these redundant statements does not impact safety and is consistent with NUREG-1432.

A.3 Not Used

ADMINISTRATIVE CHANGES (continued)

A.4 CTS 3.9.1 states in part, "... the boron concentration of all filled portions of the Reactor Coolant System and the refueling canal shall be maintained uniform ... ." There is no requirement for uniform boron concentration contained in ITS LCO 3.9.1. Due to the volume of water and the fact that shutdown cooling operation and refueling activities maintain water circulation, as long as the boron concentration is maintained within limits, it is safe to assume that the boron concentration will be uniform in the RCS and refueling canal. It is not necessary to require uniform concentration as part of the LCO since administrative requirements and other TS requirements provide for adequate mixing. This change does not impact safety and is consistent with NUREG-1432.

A.5 CTS 4.9.1.2 states, "The boron concentration of the Reactor Coolant System and the refueling canal shall be determined by chemical analysis at least once per 72 hours." ITS SR 3.9.1.1 Requires verification that boron concentration is within the limit specified in the COLR once per 72 hours. Instead of requiring that the boron concentration be determined by chemical analysis, the ITS requires that the results of the Surveillance be verified to be within the limit specified in the COLR. This change is administrative since the intent of both CTS 4.9.1.2 and ITS SR 3.9.1.1 is to test the boron concentration and verify it is within limits. Deleting unnecessary details and adding clarification does not impact safety. This change is consistent with NUREG-1432.

TECHNICAL CHANGES - MORE RESTRICTIVE

None

TECHNICAL CHANGES - RELOCATIONS

LA.1 CTS 3.9.1 states in part, "... continue boration at greater than or equal to 26 gpm of a solution containing > 4000 ppm boron or its equivalent ..." ITS LCO 3.9.1 Required Action A.3 states, "Initiate action to restore boron concentration to within limit." This requirement is not required to determine the OPERABILITY of a system, component or structure and therefore is being relocated to the ITS Bases.

The Action specified by CTS 3.9.1 merely requires that boration be performed by feeding at the minimum charging flow rate allowed by CTS 3.1.2.2 with a borated water source meeting the minimum boron concentration limit required by CTS 3.1.2.5. Both of these Specifications are relocated to the TRM per the split report. In determining the required combination of boration flow rate and concentration, there is no unique design basis event that must be satisfied. The only requirement is to restore the boron concentration to its required value as soon as possible. In order to raise the boron concentration as soon as possible, the operator should begin boration with the best source available for unit conditions.

Any change to the requirements in the Bases will be governed by the provisions of the ITS 5.5.14, "Technical Specification Bases Control Program" and 10 CFR 50.59. This provides an equivalent level of regulatory control and is an administrative change with no impact on the margin of safety. This requirement does not need to be in the ITS to provide adequate protection to the public health and safety. Therefore, relocation of this requirement is acceptable and is consistent with NUREG-1432.

TECHNICAL CHANGES - RELOCATIONS (continued)

LA.2 CTS 4.9.1.1 requires that the boron concentration be determined to be within the limits specified in the COLR prior to removing unbolting or removing the reactor pressure vessel head, and prior to withdrawal of any full-length CEA in excess of 3 feet from its fully inserted position within the reactor pressure vessel. These details are not included in ITS 3.9.1. This requirement is not required to determine the OPERABILITY of a system, component or structure and therefore is being relocated to the Technical Requirements Manual (TRM).

The limitations on boron concentration during refueling operations is intended to assure that the reactor will remain subcritical during Mode 6 including Core Alterations. The requirements of CTS 4.9.1.1 are intended to assure that the boron concentration is within the specified limits prior to evolutions which could cause an increase in reactivity. The 72 hour Frequency specified in CTS 4.9.1.2 and ITS SR 3.9.1.1 provide adequate assurance that boron concentration will be within limits any time the unit is in Mode 6.

Any change to the requirements in the TRM will be governed by the provisions of 10 CFR 50.59. This provides an equivalent level of control and is an administrative change with no impact on the margin of safety. This requirement does not need to be in the ITS to provide adequate protection to the public health and safety. Therefore, relocation of this requirement is acceptable and is consistent with NUREG-1432.

TECHNICAL CHANGES - LESS RESTRICTIVE

None

# NO SIGNIFICANT HAZARDS CONSIDERATION
# SPECIFICATION 3.9.1

ADMINISTRATIVE CHANGES
(ITS 3.9.1 Discussion of Changes Labeled A.1, A.2, A.4, and A.5)

Arizona Public Service Company, Palo Verde Nuclear Generating Station (PVNGS), Units 1, 2, and 3, is converting to the ITS as outlined in NUREG-1432, "Standard Technical Specifications, Combustion Engineering Plants." The proposed changes involve the reformatting, renumbering, rewording of the Technical Specifications (TS) and Bases with no change in intent, and the incorporation of current operating practices consistent with NUREG-1432. These changes, since they do not involve technical changes to the Current TS (CTS), are administrative. Below are the No Significant Hazards Consideration (NSHC) for the conversion of this Section/Chapter to NUREG-1432.

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 a proposed amendment, would not 1) involve a significant increase in the probability or consequences of an accident previously evaluated; 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?

> The proposed changes involve reformatting, renumbering, and rewording of the CTS and Bases along with incorporation of PVNGS current operating practices and other changes to the CTS as discussed in the specific Discussion of Changes listed above in order to be consistent with NUREG-1432. The reformatting, renumbering, and rewording along with the other changes listed above, involves no technical changes to the CTS. Specifically, there will be no change in the requirements imposed on PVNGS due to these changes. During development of NUREG-1432, certain wording preferences or English language conventions were adopted. The proposed changes to this Section/Chapter are administrative in nature and do not impact initiators of any analyzed events. They also do not impact the assumed mitigation of accidents or transient events. Therefore, these changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

ADMINISTRATIVE CHANGES
(ITS 3.9.1 Discussion of Changes Labeled (A.1, A.2, A.4, and A.5) (continued)

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

The proposed changes involve reformatting, renumbering, and rewording of the CTS, along with the incorporation of PVNGS current operating practices and other changes, as discussed, in order to be consistent with NUREG-1432. The proposed changes do not involve a physical alteration of the plant (no new or different type of equipment will be installed) or change the methods governing normal plant operation. The proposed changes will not impose any new or different requirements or eliminate any existing requirements. Therefore, these changes do 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?

The proposed changes involve reformatting, renumbering, and rewording of the CTS, along with the incorporation of PVNGS current operating practices and other changes, as discussed, in order to be consistent with NUREG-1432. The proposed changes are administrative in nature and will not involve any technical changes. The proposed changes will not reduce a margin of safety because they have no impact on any safety analysis assumptions. Also, because these changes are administrative in nature, no question of safety is involved. Therefore, these changes do not involve a significant reduction in a margin of safety.

TECHNICAL CHANGES - RELOCATIONS
(ITS 3.9.1 Discussion of Changes Labeled LA.1 and LA.2)

Arizona Public Service Company, Palo Verde Nuclear Generating Station (PVNGS), Units 1, 2, and 3 is converting to the ITS as outlined in NUREG-1432. The proposed changes, since detail is being removed from the CTS to a Licensee Controlled Document, are less restrictive. The descriptions of these changes are in the Discussion of Changes listed above.

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 a proposed amendment, would not 1) involve a significant increase in the probability or consequences of an accident previously evaluated; 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?

> The proposed changes relocate requirements from the CTS to a Licensee Controlled Document. These changes do not result in any hardware changes or changes to plant operating practices. The details being relocated are not assumed to be an initiator of any analyzed event. The Licensee Controlled Document containing the relocated requirements will be maintained using the provisions of 10 CFR 50.59 or other specified control processes and is subject to the change control process in the Administrative Controls Section of the ITS. Since any changes to a Licensee Controlled Document will be evaluated, no increase in the probability or consequences of an accident previously evaluated will be allowed. Therefore, these changes will not involve a significant increase in the probability or consequences of an accident previously evaluated.

TECHNICAL CHANGES - RELOCATIONS
(ITS 3.9.1 Discussion of Changes Labeled LA.1 and LA.2) (continued)

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

The proposed changes relocate requirements from the CTS to a Licensee Controlled Document. These changes will not alter the plant configuration (no new or different type of equipment will be installed) or change the methods governing normal plant operation. These changes will not impose different requirements and adequate control of information will still be maintained. These changes will not alter assumptions made in the safety analysis or licensing basis. Therefore, these changes will 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?

The proposed changes relocate requirements from the CTS to a Licensee Controlled Document. These changes will not reduce a margin of safety since they have no impact on any safety analysis assumptions. In addition, the requirements to be transposed from the CTS to the Licensee Controlled Document are the same as the CTS. Since any future changes to this Licensee Controlled Document will be evaluated per the requirements of 10 CFR 50.59, or other specified control processes, no reduction (significant or insignificant) in a margin of safety will be allowed. Therefore, these changes will not involve a significant reduction in a margin of safety.

The NRC review provides a certain margin of safety, and although this review will no longer be performed prior to submittal, the NRC still inspects the 10 CFR 50.59 process. The proposed changes are consistent with NUREG-1432, which was approved by the NRC Staff. The change controls for proposed relocated details and requirements provide an acceptable level of regulatory authority. Revising the CTS to reflect the approved level of detail per NUREG-1432 reinforces the conclusion that there is not a significant reduction in the margin of safety. Therefore, revising the CTS to reflect the NRC accepted level of detail and requirements ensures no reduction in a margin of safety.

CE STS
NUREG-1432 REV. 1
SPECIFICATION 3.9.2
MARK UP

<DOC>
<CTS>

## 3.9 REFUELING OPERATIONS

### 3.9.2 Nuclear Instrumentation

<3.9.2> LCO 3.9.2 Two source range monitors (SRMs) shall be OPERABLE.

startup 2

APPLICABILITY: MODE 6.

ACTIONS

| | CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|---|
| <3.9.2 ACT a.> | A. One [required] SRM inoperable. | A.1 Suspend CORE ALTERATIONS. | Immediately |
| | | AND | |
| | | A.2 Suspend positive reactivity additions. | Immediately |
| <3.9.2 ACT b.> <DOC M.1> | B. Two [required] SRMs inoperable. | B.1 Initiate action to restore one SRM to OPERABLE status. | Immediately |
| | | AND | |
| <DOC M.2> | | B.2 Perform SR 3.9.1.1. | 4 hours<br>AND<br>Once per 12 hours thereafter |

SURVEILLANCE REQUIREMENTS

| | SURVEILLANCE | FREQUENCY |
|---|---|---|
| <4.9.2.a> | SR 3.9.2.1 Perform CHANNEL CHECK. | 12 hours |
| <DOC A.2> <DOC M.3> | SR 3.9.2.2 -------------------NOTE------------------- Neutron detectors are excluded from CHANNEL CALIBRATION. ------------------------------------------ Perform CHANNEL CALIBRATION. | 18 months |

CE STS
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SPECIFICATION 3.9.2
BASES MARK UP

startup channel neutron flux monitors or

B 3.9 REFUELING OPERATIONS

B 3.9.2 Nuclear Instrumentation

and the associated Boron Dilution Alarm System (BDAS) (2)

BASES

| | |
|---|---|
| BACKGROUND | (2) startup The ~~source~~ range monitors (SRMs) are used during ~~refueling operations~~ CORE ALTERATIONS or movement of irradiated fuel assemblies in Containment (2) to monitor the core reactivity condition. The installed SRMs are part of the Excore Nuclear Instrumentation System ~~(NIS)~~. These detectors are located external to the reactor vessel and detect neutrons leaking from the core. The use of portable detectors is permitted, provided the LCO requirements are met. |
| | The installed SRMs are BF3 detectors operating in the proportional region of the gas filled detector characteristic curve. The detectors monitor the neutron flux in counts per second. The instrument range covers five decades of neutron flux (1E+5 cps) with a ~~[5%]~~ instrument accuracy. The detectors also provide continuous visual indication in the control room and (2) and audible indication in the control room and containment. an audible alarm ~~to~~ alerts (BDAS) operators to a possible dilution accident. The ~~NIS is~~ excore startup channels are designed in accordance with the criteria presented in Reference 1. |
| | ~~If used, portable detectors should be functionally equivalent to the NIS SRMs.~~ (2) |
| APPLICABLE SAFETY ANALYSES | Two OPERABLE SRMs and the associated BDAS (2) are required to provide a signal to alert the operator to unexpected changes in core reactivity such as by a boron dilution accident or an improperly loaded fuel assembly. The safety analysis of the uncontrolled boron dilution accident is described in Reference 2. The analysis of the uncontrolled boron dilution accident shows that normally available SHUTDOWN MARGIN would be reduced, but there is sufficient time for the operator to take corrective actions. |
| | The SRMs satisfy Criterion 3 of ~~the NRC Policy Statement.~~ 10 CFR 50.36(c)(2)(ii) (1) |
| LCO | This LCO requires two SRMs OPERABLE to ensure that redundant monitoring capability is available to detect changes in core reactivity. |

(continued)

BASES (continued)

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APPLICABILITY

In MODE 6, the SRMs must be OPERABLE to determine changes in core reactivity. There is no other direct means available to check core reactivity levels.

In MODES 2, 3, 4, and 5, the installed source range detectors and circuitry are required to be OPERABLE by LCO 3.3.2, 12 "~~RPS Instrumentation Shutdown~~ Boron Dilution Alarm System (BDAS)." 2

---

ACTIONS

A.1 and A.2

With only one SRM OPERABLE, redundancy has been lost. Since these instruments are the only direct means of monitoring core reactivity conditions, CORE ALTERATIONS and positive reactivity additions must be suspended immediately. Performance of Required Action A.1 shall not preclude completion of movement of a component to a safe position.

B.1

With no SRM OPERABLE, action to restore a monitor to OPERABLE status shall be initiated immediately. Once initiated, action shall be continued until an SRM is restored to OPERABLE status.

B.2

With no SRM OPERABLE, there is no direct means of detecting changes in core reactivity. However, since CORE ALTERATIONS and positive reactivity additions are not to be made, the core reactivity condition is stabilized until the SRMs are OPERABLE. This stabilized condition is determined by performing SR 3.9.1.1 to verify that the required boron concentration exists.

The Completion Time of 4 hours is sufficient to obtain and analyze a reactor coolant sample for boron concentration. The Frequency of once per 12 hours ensures that unplanned changes in boron concentration would be identified. The 12 hour Frequency is reasonable, considering the low probability of a change in core reactivity during this period.

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(continued)

BASES (continued)

2

SURVEILLANCE REQUIREMENTS

The frequency is based on operating experience that demonstrates the rarity of channel failure. Since the probability of two random failures in redundant channels in any 12 hour period is extremely low, a CHANNEL CHECK minimizes the chance of loss of function due to failure of redundant channels.

The detectors are of simple construction, and any failures in the detectors will be apparent as change in channel output.

SR 3.9.2.1

SR 3.9.2.1 is the performance of a CHANNEL CHECK, which is a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that the two indication channels should be consistent with core conditions. Changes in fuel loading and core geometry can result in significant differences between source range channels, but each channel should be consistent with its local conditions.

~~The Frequency of 12 hours is consistent with the CHANNEL CHECK Frequency specified similarly for the same instruments in LCO 3.3.1, "Reactor Protection System."~~

SR 3.9.2.2

SR 3.9.2.2 is the performance of a CHANNEL CALIBRATION every 18 months. This SR is modified by a Note stating that neutron detectors are excluded from the CHANNEL CALIBRATION. ~~The CHANNEL CALIBRATION for the source range neutron flux monitors consists of obtaining the detector plateau or preamp discriminator curves, evaluating those curves, and comparing the curves to the manufacturer's data.~~ The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage. Operating experience has shown these components usually pass the Surveillance when performed on the 18 month Frequency.

The Surveillance verifies that the channel responds to a measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drift between successive calibrations to ensure that the channel remains operational. This SR is an extension of SR 3.3.12 for the Boron Dilution Alarm System CHANNEL CALIBRATION listed here because of its Applicability in these MODES.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 13, GDC 26, GDC 28, and GDC 29.
2. UFSAR ~~FSAR~~, Section ~~[ ]~~ 15.4.6 (1)

# NUREG-1432 EXCEPTIONS
# SPECIFICATION 3.9.2

1. Grammar and/or editorial changes have been made to enhance clarity. No technical or intent changes to the Specification are made by this change.

2. The plant specific titles, nomenclature, number parameter/value, reference, system description, system design, operating practices or analysis description was used (additions, deletions, and/or changes are included). Plant specific parameters/values were directly transferred from the CTS to the ITS, or from the plant design basis to the ITS. The Bases have been revised to be consistent with the LCO/Surveillance.

PVNGS CTS
SPECIFICATION 3.9.2
MARK UP

Specification 3.9.2

A.1

3.9 REFUELING OPERATIONS

3.9.2 ~~3/4.9.2 INTRUMENTATION~~ Nuclear Instrumentation

~~LIMITED CONDITION FOR OPERATION~~

LCO3.9.2 ~~3.9.2 As a minimum,~~ two startup range ~~channel neutron flux~~ monitors shall be OPERABLE ~~and operating, each with continuous visual indication in the control room and one with audible indication in the containment and control room.~~ LA.1

APPLICABILITY: MODE 6.

ACTION:

ACT A. a. With one of the above required monitors inoperable ~~or not operating,~~ L.1 immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

ACT B. b. With both of the above required monitors inoperable ~~or not operating,~~ L.1 Immediately initiate action to restore one SRM to OPERABLE status, and M.1 determine the boron concentration of the Reactor Coolant System and the refueling canal Within 4 hours and M.2 at least once per 12 hours thereafter.

SURVEILLANCE REQUIREMENTS

~~4.9.2 Each startup channel neutron flux monitor shall be demonstrated OPERABLE by performance of:~~

SR3.9.2.1 ~~a.~~ Perform A CHANNEL CHECK at least once per 12 hours,

~~b. A CHANNEL FUNCTIONAL TEST within 8 hours prior to the initial start of CORE ALTERATIONS, and~~ L.2

~~c. A CHANNEL FUNTIONAL TEST at least once per 7 days.~~

A.2 NOTE: Neutron detectors are excluded from CHANNEL CALIBRATION

SR3.9.2.2 Perform a CHANNEL CALIBRATION once per 18 months. M.3

# DISCUSSION OF CHANGES
# SPECIFICATION 3.9.2

ADMINISTRATIVE CHANGES

A.1 All reformatting and renumbering is in accordance with the Combustion Engineering Plant (CEOG) Standard Technical Specifications NUREG-1432, Rev. 1 (NUREG-1432). As a result, the Palo Verde Nuclear Generating Station (PVNGS) Improved Technical Specifications (ITS) should be more readable, and therefore understandable, by plant operators as well as other users. During the reformatting and renumbering of the ITS, no technical changes (either actual or interpretational) to the Current Technical Specification (CTS) were made unless they were identified and justified.

Editorial rewording (either adding or deleting) is made consistent with NUREG-1432. During NUREG-1432 development, certain wording preferences or English language conventions were adopted which resulted in no technical changes (either actual or interpretational) to the CTS.

Additional information has also been added to more fully describe each subsection. This wording is consistent with NUREG-1432. Since the design is already approved by the NRC, adding more detail does not result in a technical change.

A.2 CTS 4.9.2.b and 4.9.2.c require CHANNEL FUNCTIONAL TESTS for the SRMs. ITS SR 3.9.2.2 requires a CHANNEL CALIBRATION for the SRMs but is modified by a Note which excludes the neutron detectors from the CHANNEL CALIBRATION. The neutron detectors are also not included in the CHANNEL FUNCTIONAL TESTS required by CTS 4.9.2.b and CTS 4.9.2.c. Adding a Note clarifying that the neutron detectors are excluded from the CHANNEL CALIBRATION does not affect the application of this Specification. This change does not impact safety. This change is consistent with NUREG-1432.

TECHNICAL CHANGES - MORE RESTRICTIVE

M.1 CTS 3.9.2 Action b states in part, "With both of the required monitors inoperable ... determine the boron concentration of the Reactor Coolant System ... ." ITS LCO 3.9.2 contains the additional requirement that action be initiated immediately to restore one SRM to OPERABLE status. With one SRM inoperable, both CTS 3.9.2 and ITS LCO 3.9.2 Actions require suspending Core Alterations and positive reactivity changes. With two required SRMs inoperable , both CTS 3.9.2 and ITS LCO 3.9.2 require monitoring of boron concentration. These Actions are sufficient to place the unit in a safe and stable condition and assure it is maintained there. Adding the requirement to immediately initiate action to return one SRM to OPERABLE status is prudent since they are the only direct means of detecting changes in core reactivity. The addition of this requirement constitutes a more restrictive change to PVNGS plant operation. This change is consistent with NUREG-1432.

M.2 CTS 3.9.2 Action b states in part, "With both of the above required monitors inoperable ... determine the boron concentration of the Reactor Coolant System at least once per 12 hours." ITS LCO 3.9.2 Required Action B.2 requires that with two required SRMs inoperable, the boron concentration of the Reactor Coolant System and the refueling canal be verified within the limit specified in the COLR within 4 hours and once per 12 hours thereafter. CTS 3.9.2 Action b requires only the RCS boron concentration to be verified while ITS LCO 3.9.2 Required Action B.2 requires all volumes specified in the LCO be verified. The addition of the refueling canal is prudent since it is connected to the RCS when flooded, thus monitoring the boron concentration assures system stability. Addition of the requirement to initially verify boron concentration within 4 hours of Condition entry provides operators with assurance that the unit is initially in a safe and stable condition upon loss of reactivity indication. The addition of this requirement constitutes a more restrictive change to PVNGS plant operation. This change is consistent with NUREG-1432.

M.3 CTS 4.9.2.b and CTS 4.9.2.c require performing CHANNEL FUNCTIONAL TESTS. ITS SR3.9.2.2 requires a CHANNEL CALIBRATION of the SRMs. The CHANNEL CALIBRATION is a more rigorous Surveillance and by definition includes a CHANNEL FUNCTIONAL TEST. Performing a more rigorous Surveillance provides assurance that all functions of the SRMs will be capable of performing as required. PVNGS current operating practice is to perform a Channel Calibration once per 18 months as part of maintenance. The change in frequency is discussed in DOC L2. This change does not impact safety. This change is consistent with NUREG-1432.

TECHNICAL CHANGES - RELOCATIONS

LA.1 CTS 3.9.2 states in part, "... two startup channel neutron flux monitors shall be OPERABLE and operating, each with continuous visual indication in the control room and one with audible indication in the containment and control room." ITS LCO 3.9.2 states, "Two Startup Range Monitors (SRMs) shall be OPERABLE." The SRMs are required during refueling operations to alert the operators of unexpected changes in core reactivity. Two channels are required Operable to ensure that redundant monitoring capability is available. The monitoring systems are required to be in operation to be considered Operable. The details of system operation contained in the CTS LCO are not required to determine the Operability of a system, component or structure and therefore are being relocated to the Bases.

Any change to the requirements in the Bases will be governed by the Bases Control Program. This provides an equivalent level of control and is an administrative change with no impact on the margin of safety. This requirement does not need to be in the ITS to provide adequate protection to the public health and safety. Therefore, relocation of this requirement to a Licensee Controlled Document is acceptable and is consistent with NUREG-1432.

TECHNICAL CHANGES - LESS RESTRICTIVE

L.1 CTS 3.9.2 Action a and Action b provide the Required Actions necessary with either one or two SRMs inoperable or not operating. ITS LCO 3.9.2 Action A and Action B provide the Actions necessary with either one or two SRMs inoperable. The SRMs are required to be in operation to be considered Operable; therefore, it is not necessary to state the condition of "not operating" since if the monitor is not operating, it is inoperable. Deleting "not operating" from the Conditions does not alter the application of this Specification. This change does not impact safety and is consistent with NUREG-1432.

TECHNICAL CHANGES - LESS RESTRICTIVE (continued)

L.2 CTS 4.9.2.b and CTS 4.9.2.c require a CHANNEL FUNCTIONAL TEST within 8 hours prior to the initial start of Core Alterations, and at least once per 7 days. ITS 3.9.2 does not require CHANNEL FUNCTIONAL TESTS for the SRMs. The SRMs provide the operator with information to monitor the core reactivity condition. The SRMs provide visual and audible indication only, they do not provide control or alarm functions. The CHANNEL CHECK performed per CTS 4.9.2.a and ITS SR 3.9.2 1 provide assurance that the necessary indication is available when needed. A CHANNEL FUNCTIONAL TEST is required for other Specifications to demonstrate the operability of alarm, interlock, and control functions that are required for those functions and cannot be demonstrated by the performance of CHANNEL CHECKS. Since the monitoring capabilities of the SRMs are adequately verified by performance of CHANNEL CHECKS, removal of the requirement to perform CHANNEL FUNCTIONAL TESTS does not impact safety. Performing a more rigorous CHANNEL CALIBRATION (which by definition includes a CHANNEL FUNCTIONAL TEST) once per 18 months provides adequate assurance that the system will perform as designed. PVNGS current operating practice is to perform the CHANNEL CALIBRATION once per 18 months.This change is consistent with NUREG-1432.

# NO SIGNIFICANT HAZARDS CONSIDERATION
# SPECIFICATION 3.9.2

ADMINISTRATIVE CHANGES
(ITS 3.9.2 Discussion of Changes Labeled A.1 and A.2)

Arizona Public Service Company, Palo Verde Nuclear Generating Station (PVNGS), Units 1, 2, and 3, is converting to the ITS as outlined in NUREG-1432, "Standard Technical Specifications, Combustion Engineering Plants." The proposed changes involve the reformatting, renumbering, rewording of the Technical Specifications (TS) and Bases with no change in intent, and the incorporation of current operating practices consistent with NUREG-1432. These changes, since they do not involve technical changes to the Current TS (CTS), are administrative. Below are the No Significant Hazards Consideration (NSHC) for the conversion of this Section/Chapter to NUREG-1432.

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 a proposed amendment, would not 1) involve a significant increase in the probability or consequences of an accident previously evaluated; 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?

> The proposed changes involve reformatting, renumbering, and rewording of the CTS and Bases along with incorporation of PVNGS current operating practices and other changes to the CTS as discussed in the specific Discussion of Changes listed above in order to be consistent with NUREG-1432. The reformatting, renumbering, and rewording along with the other changes listed above, involves no technical changes to the CTS. Specifically, there will be no change in the requirements imposed on PVNGS due to these changes. During development of NUREG-1432, certain wording preferences or English language conventions were adopted. The proposed changes to this Section/Chapter are administrative in nature and do not impact initiators of any analyzed events. They also do not impact the assumed mitigation of accidents or transient events. Therefore, these changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

ADMINISTRATIVE CHANGES
(ITS 3.9.2 Discussion of Changes Labeled (A.1 and A.2) (continued)

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

The proposed changes involve reformatting, renumbering, and rewording of the CTS, along with the incorporation of PVNGS current operating practices and other changes, as discussed, in order to be consistent with NUREG-1432. The proposed changes do not involve a physical alteration of the plant (no new or different type of equipment will be installed) or change the methods governing normal plant operation. The proposed changes will not impose any new or different requirements or eliminate any existing requirements. Therefore, these changes do 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?

The proposed changes involve reformatting, renumbering, and rewording of the CTS, along with the incorporation of PVNGS current operating practices and other changes, as discussed, in order to be consistent with NUREG-1432. The proposed changes are administrative in nature and will not involve any technical changes. The proposed changes will not reduce a margin of safety because they have no impact on any safety analysis assumptions. Also, because these changes are administrative in nature, no question of safety is involved. Therefore, these changes do not involve a significant reduction in a margin of safety.

TECHNICAL CHANGES - MORE RESTRICTIVE
(ITS 3.9.2 Discussion of Changes Labeled M.1, M.2 and M.3)

Arizona Public Service Company, Palo Verde Nuclear Generating Station (PVNGS), Units 1, 2, and 3 is converting to the ITS as outlined in NUREG-1432. This particular NSHC is for the changes labeled "Technical Changes - More Restrictive" described in the specific Discussion of Changes listed above. The proposed changes incorporate more restrictive changes into the CTS by either making current requirements more stringent or adding new requirements which currently do not exist.

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 a proposed amendment, would not 1) involve a significant increase in the probability or consequences of an accident previously evaluated; 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?

> The proposed changes provide more stringent requirements than previously existed in the CTS. The more stringent requirements will not result in operation that will increase the probability of initiating an analyzed event. If anything, the new requirements may decrease the probability or consequences of an analyzed event by incorporating the more restrictive changes discussed in the specific Discussion of Changes listed above. These changes will not alter assumptions relative to mitigation of an accident or transient event. The more restrictive requirements will not alter the operation and will continue to ensure process variables, structures, systems, or components are maintained consistent with safety analyses and licensing basis. These changes have been reviewed to ensure that no previously evaluated accident has been adversely affected. Therefore, these changes will not involve a significant increase in the probability or consequences of an accident evaluated.

TECHNICAL CHANGES - MORE RESTRICTIVE
(ITS 3.9.2 Discussion of Changes Labeled M.1, M.2 and M.3)

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

Making existing requirements more restrictive and adding more restrictive requirements to the CTS will not alter the plant configuration (no new or different type of equipment will be installed) or change the methods governing normal plant operation. These changes do impose different requirements. However, they are consistent with the assumptions made in the safety analyses, licensing basis, and NUREG-1432. Therefore, these changes will 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?

The proposed changes provide more stringent requirements than previously existed in the CTS. An evaluation of these changes concluded that adding these more restrictive requirements either increases or has no impact on the margin of safety. The changes provide additional restrictions which may enhance plant safety. These changes maintain requirements of the safety analysis, licensing basis, and NUREG-1432. As such, no question of safety is involved. Therefore, these changes will not involve a significant reduction in a margin of safety.

TECHNICAL CHANGES - RELOCATIONS
(ITS 3.9.2 Discussion of Changes Labeled LA.1)

Arizona Public Service Company, Palo Verde Nuclear Generating Station (PVNGS), Units 1, 2, and 3 is converting to the ITS as outlined in NUREG-1432. The proposed changes, since detail is being removed from the CTS to a Licensee Controlled Document, are less restrictive. The descriptions of these changes are in the Discussion of Changes listed above.

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 a proposed amendment, would not 1) involve a significant increase in the probability or consequences of an accident previously evaluated; 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?

> The proposed changes relocate requirements from the CTS to a Licensee Controlled Document. These changes do not result in any hardware changes or changes to plant operating practices. The details being relocated are not assumed to be an initiator of any analyzed event. The Licensee Controlled Document containing the relocated requirements will be maintained using the provisions of 10 CFR 50.59 or other specified control processes and is subject to the change control process in the Administrative Controls Section of the ITS. Since any changes to a Licensee Controlled Document will be evaluated, no increase in the probability or consequences of an accident previously evaluated will be allowed. Therefore, these changes will not involve a significant increase in the probability or consequences of an accident previously evaluated.

TECHNICAL CHANGES - RELOCATIONS
(ITS 3.9.2 Discussion of Changes Labeled LA.1) (continued)

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

The proposed changes relocate requirements from the CTS to a Licensee Controlled Document. These changes will not alter the plant configuration (no new or different type of equipment will be installed) or change the methods governing normal plant operation. These changes will not impose different requirements and adequate control of information will still be maintained. These changes will not alter assumptions made in the safety analysis or licensing basis. Therefore, these changes will 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?

The proposed changes relocate requirements from the CTS to a Licensee Controlled Document. These changes will not reduce a margin of safety since they have no impact on any safety analysis assumptions. In addition, the requirements to be transposed from the CTS to the Licensee Controlled Document are the same as the CTS. Since any future changes to this Licensee Controlled Document will be evaluated per the requirements of 10 CFR 50.59, or other specified control processes, no reduction (significant or insignificant) in a margin of safety will be allowed. Therefore, these changes will not involve a significant reduction in a margin of safety.

The NRC review provides a certain margin of safety, and although this review will no longer be performed prior to submittal, the NRC still inspects the 10 CFR 50.59 process. The proposed changes are consistent with NUREG-1432, which was approved by the NRC Staff. The change controls for proposed relocated details and requirements provide an acceptable level of regulatory authority. Revising the CTS to reflect the approved level of detail per NUREG-1432 reinforces the conclusion that there is not a significant reduction in the margin of safety. Therefore, revising the CTS to reflect the NRC accepted level of detail and requirements ensures no reduction in a margin of safety.

TECHNICAL CHANGES - LESS RESTRICTIVE
(ITS 3.9.2 Discussion of Changes Labeled L.1)

Arizona Public Service Company, Palo Verde Nuclear Generating Station (PVNGS), Units 1, 2, and 3 is converting to the ITS as outlined in NUREG-1432. The proposed change involves making the CTS less restrictive. Below is the description of this less restrictive change and the NSHC for the conversion to NUREG 1432.

L.1 CTS 3.9.2 Action a and Action b provide the Required Actions necessary with either one or two SRMs inoperable or not operating. ITS LCO 3.9.2 Action A and Action B provide the Actions necessary with either one or two SRMs inoperable. The SRMs are required to be in operation to be considered Operable; therefore, it is not necessary to state the condition of "not operating" since if the monitor is not operating, it is inoperable. Deleting "not operating" from the Conditions does not alter the application of this Specification. This change does not impact safety and is consistent with NUREG-1432.

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 a proposed amendment, would not 1) involve a significant increase in the probability or consequences of an accident previously evaluated; 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:

TECHNICAL CHANGES - LESS RESTRICTIVE
(ITS 3.9.2 Discussion of Changes Labeled L.1) (continued)

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

The proposed change modifies the Conditions of the Action Statements by removing the words, "or not operating." The Startup Range Monitors (SRMs) do not receive any type of auto-start and therefore must be operating to be considered OPERABLE. The details of system operation and the specific requirements for system OPERABILITY are provided elsewhere. Since the SRMs are required to be operating to be considered Operable, it is not necessary to restate this requirement in the Conditions and Required Actions.

This change is consistent with NUREG-1432. This change does not result in any hardware changes or changes to plant operating practices nor does it affect plant operation. Therefore, this change will not involve a significant increase in the probability or 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?

The proposed change modifies the Conditions of the Action Statements by removing the words, "or not operating." The Startup Range Monitors (SRMs) do not receive any type of auto-start and therefore must be operating to be considered OPERABLE. The details of system operation and the specific requirements for system OPERABILITY are provided elsewhere. Since the SRMs are required to be operating to be considered Operable, it is not necessary to restate this requirement in the Conditions and Required Actions.

This change is consistent with NUREG-1432. This change will not alter the plant configuration (no new or different type of equipment will be installed) or change the methods of governing normal plant operation. This change will not alter assumptions made in the safety analysis or licensing basis. Therefore, this change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

TECHNICAL CHANGES - LESS RESTRICTIVE
(ITS 3.9.2 Discussion of Changes Labeled L.1) (continued)

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

The proposed change modifies the Conditions of the Action Statements by removing the words, "or not operating." The Startup Range Monitors (SRMs) do not receive any type of auto-start and therefore must be operating to be considered OPERABLE. The details of system operation and the specific requirements for system OPERABILITY are provided elsewhere. Since the SRMs are required to be operating to be considered Operable, it is not necessary to restate this requirement in the Conditions and Required Actions.

This change will not reduce a margin of safety since it has no impact on safety analysis assumptions. This change is consistent with NUREG-1432, which was approved by the NRC Staff. Therefore, this change does not result in a reduction in a margin of safety.

TECHNICAL CHANGES - LESS RESTRICTIVE
(ITS 3.9.2 Discussion of Changes Labeled L.2)

Arizona Public Service Company, Palo Verde Nuclear Generating Station (PVNGS), Units 1, 2, and 3 is converting to the ITS as outlined in NUREG-1432. The proposed change involves making the CTS less restrictive. Below is the description of this less restrictive change and the NSHC for conversion to NUREG-1432.

L.2 CTS 4.9.2.b and CTS 4.9.2.c require a CHANNEL FUNCTIONAL TEST within 8 hours prior to the initial start of Core Alterations, and at least once per 7 days. ITS 3.9.2 does not require CHANNEL FUNCTIONAL TESTS for the SRMs. The SRMs provide the operator with information to monitor the core reactivity condition. The SRMs provide visual and audible indication only, they do not provide control or alarm functions. The CHANNEL CHECK performed per CTS 4.9.2.a and ITS SR 3.9.2 1 provide assurance that the necessary indication is available when needed. A CHANNEL FUNCTIONAL TEST is required for other Specifications to demonstrate the operability of alarm, interlock, and control functions that are required for those functions and cannot be demonstrated by the performance of CHANNEL CHECKS. Since the monitoring capabilities of the SRMs are adequately verified by performance of CHANNEL CHECKS, removal of the requirement to perform CHANNEL FUNCTIONAL TESTS does not impact safety. Performing a more rigorous CHANNEL CALIBRATION (which by definition includes a CHANNEL FUNCTIONAL TEST) once per 18 months provides adequate assurance that the system will perform as designed. PVNGS current operating practice is to perform the CHANNEL CALIBRATION once per 18 months.This change is consistent with NUREG-1432.

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 a proposed amendment, would not 1) involve a significant increase in the probability or consequences of an accident previously evaluated; 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:

TECHNICAL CHANGES - LESS RESTRICTIVE
(ITS 3.9.2 Discussion of Changes Labeled L.2) (continued)

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

The proposed change removes the requirement to perform Channel Functional tests within 8 hours prior to the initial start of Core Alterations and every 7 days. The SRMs provide visual and audible indication only, they do not provide any alarm or control functions. The Channel Checks which are required to be performed once per 12 hours during Mode 6 are capable of adequately verifying the monitoring capabilities of the SRMs. A Channel Calibration which is performed on an 18 month frequency has also been added to the ITS. This SR is more rigorous than the Channel Functional Test.

This change is consistent with NUREG-1432. This change does not result in any hardware changes or changes to plant operating practices nor does it affect plant operation. Therefore this change will not involve a significant increase in the probability or 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?

The proposed change removes the requirement to perform Channel Functional tests within 8 hours prior to the initial start of Core Alterations and every 7 days. The SRMs provide visual and audible indication only, they do not provide any alarm or control functions. The Channel Checks which are required to be performed once per 12 hours during Mode 6 are capable of adequately verifying the monitoring capabilities of the SRMs. A Channel Calibration which is performed on an 18 month frequency has also been added to the ITS. This SR is more rigorous than the Channel Functional Test.

This change is consistent with NUREG-1432. This change will not alter the plant configuration (no new or different type of equipment will be installed) or change the methods of governing normal plant operation. This change will not alter assumptions made in the safety analysis or licensing basis. Therefore, this change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

TECHNICAL CHANGES - LESS RESTRICTIVE
(ITS 3.9.2 Discussion of Changes Labeled L.2) (continued)

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

The proposed change removes the requirement to perform Channel Functional tests within 8 hours prior to the initial start of Core Alterations and every 7 days. The SRMs provide visual and audible indication only, they do not provide any alarm or control functions. The Channel Checks which are required to be performed once per 12 hours during Mode 6 are capable of adequately verifying the monitoring capabilities of the SRMs. A Channel Calibration which is performed on an 18 month frequency has also been added to the ITS. This SR is more rigorous than the Channel Functional Test.

This change will not reduce a margin of safety since it has no impact on safety analysis assumptions. This change is consistent with NUREG-1432, which was approved by the NRC Staff. Therefore, this change does not result in a reduction in a margin of safety.

CE STS
NUREG-1432 REV. 1
SPECIFICATION 3.9.3
MARK UP

<CTS>
<DOC> 3.9 REFUELING OPERATIONS

3.9.3 Containment Penetrations

<3.9.4> LCO 3.9.3 The containment penetrations shall be in the following status:

a. The equipment hatch closed and held in place by [four] bolts;

b. One door in each air lock closed; and

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere either:

1. closed by a manual or automatic isolation valve, blind flange, or equivalent, or

2. capable of being closed by an OPERABLE Containment Purge and Exhaust Isolation System.

APPLICABILITY: During CORE ALTERATIONS,
During movement of irradiated fuel assemblies within containment.

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| <3.9.4ACT> A. One or more containment penetrations not in required status. | A.1 Suspend CORE ALTERATIONS. | Immediately |
| | AND | |
| | A.2 Suspend movement of irradiated fuel assemblies within containment. | Immediately |

SURVEILLANCE REQUIREMENTS

| | | SURVEILLANCE | FREQUENCY |
|---|---|---|---|
| <4.9.4> | SR 3.9.3.1 | Verify each required containment penetration is in the required status. | 7 days |
| <4.9.9> | SR 3.9.3.2 | Verify each required containment purge and exhaust valve actuates to the isolation position on an actual or simulated actuation signal. | *18* months |

CE STS
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SPECIFICATION 3.9.3
BASES MARK UP

## B 3.9 REFUELING OPERATIONS

## B 3.9.3 Containment Penetrations

BASES

**BACKGROUND**

During CORE ALTERATIONS or movement of fuel assemblies within containment with irradiated fuel in containment, a release of fission product radioactivity within the containment will be restricted from escaping to the environment when the LCO requirements are met. In MODES 1, 2, 3, and 4, this is accomplished by maintaining containment OPERABLE as described in LCO 3.6.1, "Containment." In MODE 6, the potential for containment pressurization as a result of an accident is not likely; therefore, requirements to isolate the containment from the outside atmosphere can be less stringent. The LCO requirements are referred to as "containment closure" rather than "containment OPERABILITY." Containment closure means that all potential escape paths are closed or capable of being closed. Since there is no potential for containment pressurization, the Appendix J leakage criteria and tests are not required.

The containment serves to contain fission product radioactivity that may be released from the reactor core following an accident, such that offsite radiation exposures are maintained well within the requirements of 10 CFR 100. Additionally, the containment structure provides radiation shielding from the fission products that may be present in the containment atmosphere following accident conditions.

The containment equipment hatch, which is part of the containment pressure boundary, provides a means for moving large equipment and components into and out of containment. During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, the equipment hatch must be held in place by at least four bolts. Good engineering practice dictates that the bolts required by this LCO be approximately equally spaced.

The containment air locks, which are also part of the containment pressure boundary, provide a means for personnel access during MODES 1, 2, 3, and 4 operation in accordance with LCO 3.6.2, "Containment Air Locks." Each air lock has 2 doors ~~a door~~ at both ends. The doors are normally interlocked to prevent simultaneous opening when containment OPERABILITY is required. During periods of shutdown when containment

(continued)

BASES

BACKGROUND (continued)

closure is not required, the door interlock mechanism may be disabled, allowing both doors of an air lock to remain open for extended periods when frequent containment entry is necessary. During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, containment closure is required; therefore, the door interlock mechanism may remain disabled, but one air lock door must always remain closed.

The requirements on containment penetration closure ensure that a release of fission product radioactivity within containment will be restricted from escaping to the environment. The closure restrictions are sufficient to restrict fission product radioactivity release from containment due to a fuel handling accident during refueling.

refueling purge 1

2 The Containment Purge and Exhaust System includes two subsystems. The normal subsystem includes a 42 inch ~~purge~~ supply penetration and a 42 inch exhaust penetration. The second subsystem, ~~a minipurge system~~ power access purge subsystem, includes an 8 inch ~~purge~~ supply 2 penetration and an 8 inch exhaust penetration. During MODES 1, 2, 3, and 4, the two valves in each of the ~~normal~~ refueling purge and exhaust penetrations are secured in the closed position. supply 2 The two valves in each of the two ~~minipurge~~ power access purge 2 penetrations can be opened intermittently, but are closed automatically by the Engineered Safety Features Actuation System (ESFAS). Neither of the subsystems is subject to a Specification in MODE 5.

the refueling 2

In MODE 6, large air exchanges are necessary to conduct refueling operations. The normal 42 inch purge system is used for this purpose and all valves are closed by the ESFAS in accordance with ~~LCO 3.3.2, "Reactor Protective System (RPS) – Shutdown."~~ LCO 3.3.8, "Containment Purge Isolation Actuation Signal (CPIAS)."

power access purge 2 the

[The ~~minipurge~~ system remains operational in MODE 6 and ~~all four~~ the valves are also closed by the ESFAS.

or

2 ~~The minipurge system is not used in MODE 6. All four [8] inch valves are secured in the closed position.~~]

The other containment penetrations that provide direct access from containment atmosphere to outside atmosphere

(continued)

BASES

BACKGROUND (continued)

must be isolated on at least one side. Isolation may be achieved by an OPERABLE automatic isolation valve, or by a manual isolation valve, blind flange, or equivalent. Equivalent isolation methods must be approved and may include use of ~~a material that can provide a temporary, atmospheric pressure ventilation barrier for the other containment penetrations during fuel movements (Ref. 1).~~

(2) devices designed to allow eddy current testing and sludge lancing of the steam generators. Devices which present a substantial restriction to the release of containment atmosphere may be considered equivalent.

APPLICABLE SAFETY ANALYSES

During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, the most severe radiological consequences result from a fuel handling accident. The fuel handling accident is a postulated event that involves damage to irradiated fuel (Ref. 2). Fuel handling accidents, analyzed in Reference ~~2~~ 3, include dropping a single irradiated fuel assembly and handling tool or a heavy object onto other irradiated fuel assemblies. The requirements of LCO 3.9.6, "Refueling Water Level," and the minimum decay time of ~~[72]~~ 100 hours prior to CORE ALTERATIONS ensure that the release of fission product radioactivity, subsequent to a fuel handling accident, results in doses that are well within the guideline values specified in 10 CFR 100. The acceptance limits for offsite radiation exposure are contained in Standard Review Plan Section 15.7.4, Rev. 1 (Ref. ~~2~~ 3), which defines "well within" 10 CFR 100 to be 25% or less of the 10 CFR 100 values.

– Fuel Assemblies" LCO 3.9.7, "Refueling Water Level – CEAs (2)

Containment penetrations satisfy Criterion 3 of ~~the NRC Policy Statement~~ 10CFR50.36(c)(2)(ii).

LCO

This LCO limits the consequences of a fuel handling accident in containment by limiting the potential escape paths for fission product radioactivity released within containment. The LCO requires any penetration providing direct access from the containment atmosphere to the outside atmosphere to be closed except for the OPERABLE containment purge ~~and~~ supply (2) exhaust penetrations. For the OPERABLE containment purge supply (2) and exhaust penetrations, this LCO ensures that these penetrations are isolable by a valve in (2) the Containment Purge ~~and Exhaust~~ Isolation System. The OPERABILITY requirements for this LCO ensure that the automatic purge ~~and exhaust~~ (2) valve closure times specified in the ~~FSAR~~ UFSAR (1) can be achieved and therefore meet the assumptions used in the safety analysis

(continued)

BASES

| | |
|---|---|
| LCO (continued) | to ensure releases through the valves are terminated, such that the radiological doses are within the acceptance limit. |
| APPLICABILITY | The containment penetration requirements are applicable during CORE ALTERATIONS or movement of irradiated fuel assemblies within containment because this is when there is a potential for a fuel handling accident. In MODES 1, 2, 3, and 4, containment penetration requirements are addressed by LCO 3.6.1, "Containment." In MODES 5 and 6, when CORE ALTERATIONS or movement of irradiated fuel assemblies within containment are not being conducted, the potential for a fuel handling accident does not exist. Therefore, under these conditions no requirements are placed on containment penetration status. |
| ACTIONS | A.1 and A.2<br><br>With the containment equipment hatch, air locks, or any containment penetration that provides direct access from the containment atmosphere to the outside atmosphere not in the required status, including the Containment Purge and Exhaust Isolation System not capable of automatic actuation when the (2) purge and exhaust valves are open, the unit must be placed in a condition in which the isolation function is not needed. This is accomplished by immediately suspending CORE ALTERATIONS and movement of irradiated fuel assemblies within containment. Performance of these actions shall not preclude completion of movement of a component to a safe position. |
| SURVEILLANCE REQUIREMENTS | SR 3.9.3.1<br><br>This Surveillance demonstrates that each of the containment penetrations required to be in its closed position is in that position. The Surveillance on the open purge and exhaust valves will demonstrate that the valves are not blocked from closing. Also, the Surveillance will demonstrate that each valve operator has motive power, which will ensure each valve is capable of being closed by an |

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.9.3.1 (continued)

2

OPERABLE automatic containment purge and exhaust isolation signal.

The Surveillance is performed every 7 days during CORE ALTERATIONS or movement of irradiated fuel assemblies within the containment. The Surveillance interval is selected to be commensurate with the normal duration of time to complete fuel handling operations. A surveillance before the start of refueling operations will provide two or three surveillance verifications during the applicable period for this LCO. As such, this Surveillance ensures that a postulated fuel handling accident that releases fission product radioactivity within the containment will not result in a release of fission product radioactivity to the environment.

SR 3.9.3.2

2

This Surveillance demonstrates that each containment purge and exhaust valve actuates to its isolation position on manual initiation or on an actual or simulated high radiation signal. The 18 month Frequency maintains consistency with other similar ESFAS instrumentation and valve testing requirements. In LCO 3.3.4 [(Digital) or 3.3.3 (Analog)], "Miscellaneous Actuations," the Containment Purge Isolation Signal System requires a CHANNEL CHECK every 7 days and a CHANNEL FUNCTIONAL TEST every 31 days to ensure the channel OPERABILITY during refueling operations. Every 18 months a CHANNEL CALIBRATION is performed. The system actuation response time is demonstrated every 18 months, during refueling, on a STAGGERED TEST BASIS. SR 3.6.3.5 demonstrates that the isolation time of each valve is in accordance with the Inservice Testing Program requirements. These surveillances performed during MODE 6 will ensure that the valves are capable of closing after a postulated fuel handling accident to limit a release of fission product radioactivity from the containment.

The CPIAS is tested in accordance with LCO 3.3.8, "Containment Purge Isolation Actuation Signal (CPIAS)."

(continued)

BASES (continued)

REFERENCES

1. GPU Nuclear Safety Evaluation SE-0002000-001, Rev. 0, May 20, 1988.
2. UFSAR ~~FSAR~~, Section ~~**~~ 15.7.4
3. NUREG-0800, Section 15.7.4, Rev. 1, July 1981.

# NUREG-1432 EXCEPTIONS
# SPECIFICATION 3.9.3

1. Grammar and/or editorial changes have been made to enhance clarity. No technical or intent changes to the Specification are made by this change.

2. The plant specific titles, nomenclature, number parameter/value, reference, system description, system design, operating practices or analysis description was used (additions, deletions, and/or changes are included). Plant specific parameters/values were directly transferred from the CTS to the ITS, or from the plant design basis to the ITS. The Bases have been revised to be consistent with the LCO/Surveillance.

PVNGS CTS
SPECIFICATION 3.9.3
MARK UP

Specification 3.9.3

A.1

3.9 REFUELING OPERATIONS

3.9.3 ~~3/4.9.4~~ CONTAINMENT ~~BUILDING~~ PENETRATIONS

~~LIMITING CONDITION FOR OPERATION~~

LCO3.9.3 ~~3.9.4~~ The containment ~~building~~ penetrations shall be in the following status:

a. The equipment hatch ~~door~~ closed and held in place by ~~a minimum~~ of four bolts,

b. ~~A minimum of~~ one door in each airlock ~~is~~ closed, and

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere ~~shall be~~ either:

1. Closed by an A.4 automatic isolation valve, blind flange, or manual A.4 isolation valve, or equivalent L.3 or

2. ~~Be~~ capable of being closed by an OPERABLE ~~automatic~~ containment purge ~~valve.~~ and exhaust isolation system. A.5

APPLICABILITY: During CORE ALTERATIONS or movement of irradiated fuel within the containment.

ACT A. ACTION: one or more containment penetration not in required status

With ~~the requirements of the above specification not satisfied~~, immediately suspend all operations involving CORE ALTERATIONS or movement of irradiated fuel in the containment building.

SURVEILLANCE REQUIREMENTS

A.2 verified to be in the required status

SR3.9.3.1 ~~4.9.4~~ Each ~~of the above~~ required containment ~~building~~ penetrations shall be determined to be either in its closed/isolated condition or capable of being closed by an OPERABLE automatic containment purge valve within 72 hours prior to the start of and at least once per 7 days during CORE ALTERATIONS or movement of irradiated fuel in the containment building by: A.2 LA.1

~~a. Verifying the penetrations are in their closed/isolated condition, or~~ A.2

~~b. Testing the containment purge valves per the applicable portions of Specification 4.9.9.~~ A.3

Specification 3.9.3

A.1

3.9 REFUELING OPERATIONS

3.9.3 ~~3/4.9.9~~ CONTAINMENT ~~PURGE VALVE ISOLATION SYSTEM~~ Penetrations

~~LIMITING CONDITION FOR OPERATION~~

~~3.9.9 The containment purge valve isolation system shall be OPERABLE.~~

~~APPLICABILITY: During CORE ALTERATIONS or movement of irradiated fuel within the containment.~~

~~ACTION:~~

~~With the containment purge valve isolation system inoperable, close each of the containment purge penetrations providing direct access from the containment atmosphere to the outside atmosphere.~~

A.3

SURVEILLANCE REQUIREMENTS

SR 3.9.3.2 ~~4.9.9~~ ~~The containment purge valve isolation system shall be demonstrated OPERABLE within 72 hours prior to the start of and at least once per 7 days during CORE ALTERATIONS by~~ verify~~ing~~ that containment purge valve isolation occurs ~~on manual initiation and on~~ on an actual or simulated CPIAS once per 18 months L.1 L.2

DISCUSSION OF CHANGES
SPECIFICATION 3.9.3

ADMINISTRATIVE CHANGES

A.1 All reformatting and renumbering is in accordance with the Combustion Engineering Plant (CEOG) Standard Technical Specifications NUREG-1432, Rev. 1 (NUREG-1432). As a result, the Palo Verde Nuclear Generating Station (PVNGS) Improved Technical Specifications (ITS) should be more readable, and therefore understandable, by plant operators as well as other users. During the reformatting and renumbering of the ITS, no technical changes (either actual or interpretational) to the Current Technical Specification (CTS) were made unless they were identified and justified.

Editorial rewording (either adding or deleting) is made consistent with NUREG-1432. During NUREG-1432 development, certain wording preferences or English language conventions were adopted which resulted in no technical changes (either actual or interpretational) to the CTS.

Additional information has also been added to more fully describe each subsection. This wording is consistent with NUREG-1432. Since the design is already approved by the NRC, adding more detail does not result in a technical change.

A.2 CTS 4.9.4 states in part, "Each of the above required containment building penetrations shall be determined to be either in its closed/isolated condition or capable of being closed by an OPERABLE automatic containment purge valve ... by: a. verifying the penetrations are in their closed/isolated condition ... ." ITS SR 3.9.3.1 states, "Verify each required containment penetration is in the required status." The required status referred to in ITS SR 3.9.3.1 is delineated in ITS LCO 3.9.3 and is also contained in CTS 3.9.4; however, these requirements are restated in CTS 4.9.4 for clarity. Replacing the actual requirements by the statement "... is in the required status..." does not alter the requirements of the LCO or SR. This change does not impact safety and is consistent with NUREG-1432.

ADMINISTRATIVE CHANGES

A.3 CTS 4.9.4.b references CTS 4.9.9 for containment purge valve SRs during Core Alterations or movement of irradiated fuel within the containment building. The requirements of CTS 3/4.9.9 for containment purge valves have been combined with CTS 3/4.9.4 for containment penetrations in ITS 3.9.3. Since these requirements are contained in a single ITS Specification, there is no need for a cross reference or separate LCO. Combining CTS 3/4.9.4 and 3/4.9.9 is acceptable since complying with the Actions of CTS 3.9.9 results in meeting the LCO of ITS LCO 3.9.3. The Actions specified in ITS LCO 3.9.3 are adequate to place the unit in a safe condition with any containment penetration including purge penetrations inoperable. This change does not impact safety. This change is consistent with NUREG-1432.

A.4 CTS 3.9.4.c.1 requires piping penetrations (other than purge valve penetrations) which provide direct access from the containment atmosphere to outside atmosphere be closed by, "... an isolation valve, blind flange, or manual valve ... ." ITS LCO 3.9.3.c.1 requires these penetrations to be closed by, "... a manual or automatic isolation valve, blind flange, or equivalent ... ." The intent of these LCOs is to isolate the penetration using a method which has previously been determined to be acceptable. This can be either an automatic or manual isolation valve. Clarifying the nomenclature for these devices does not alter the requirements of the LCO. This change does not impact safety and is consistent with NUREG-1432.

A.5 CTS 3.9.4.c.2 requires that containment purge penetrations be capable of being closed by an OPERABLE automatic containment purge valve. CTS 3.9.9 requires the containment purge and exhaust isolation system to be OPERABLE. ITS LCO 3.9.3.c.2 requires containment purge penetrations to be capable of being closed by an OPERABLE Containment Purge and Exhaust Isolation System. The two CTS Specifications have been combined into one ITS Specification. The Action associated with CTS 3.9.9 requires that with the containment purge valve isolation system inoperable, each containment purge penetration be closed, which is consistent with the requirements of ITS LCO 3.9.3.c.1 & CTS 3.9.4.c.1. Since there is no time limit associated with CTS 3.9.9 Action, there is an acceptable level of safety provided by compliance and it is acceptable to move this requirement to the LCO of ITS LCO 3.9.3. Combining the requirements of CTS 3/4.9.4 and CTS 3/4.9.9 into one ITS Specification does not impact safety. This change is consistent with NUREG-1432.

TECHNICAL CHANGES - MORE RESTRICTIVE

None

TECHNICAL CHANGES - RELOCATIONS

LA.1 CTS 4.9.4 states in part, "Each of the above required containment building penetrations shall be determined to be either in its closed/isolated condition or capable of being closed by an OPERABLE automatic containment purge valve within 72 hours prior to the start of and at least once per 7 days during CORE ALTERATIONS or movement of irradiated fuel in the containment building... ." ITS SR 3.9.3.1 requires verification that each required containment penetration is in the required status once per 7 days but does not require this Surveillance to be current prior to entering the condition specified in the Applicability. ITS SR 3.0.4 also does not require the Surveillances to be current prior to entering the condition specified in the Applicability since it is only applicable in Modes 1, 2, 3, and 4. The requirement for the penetrations to be verified in the required status prior to entering the Applicability will be relocated to the Technical Requirements Manual (TRM). Relocation of this information to the TRM is acceptable since the Actions sufficiently define the remedial measures to be taken.

Any change to the requirements in the TRM will be governed by the provisions of 10 CFR 50.59. This provides an equivalent level of control and is an administrative change with no impact on the margin of safety. This requirement does not need to be in the ITS to provide adequate protection to the public health and safety. Therefore, relocation of this requirement to a Licensee Controlled Document is acceptable and is consistent with NUREG-1432.

TECHNICAL CHANGES - LESS RESTRICTIVE

L.1 CTS 4.9.9 requires the containment purge valves to be surveilled to assure that isolation occurs on manual initiation and on CPIAS. ITS SR 3.9.3.2 requires that automatic actuation be verified upon receipt of an actual or simulated actuation signal. CTS 4.9.9 requires that the Surveillance be performed using both a manual actuation and a CPIAS test signal. ITS SR 3.9.3.2 does not require the containment purge isolation valves to be tested using manual actuation. Testing of manual actuation is not required since the LCO implies that automatic actuation is required if the valves are open. ITS SR 3.9.3.2 also allows the flexibility of using an actual actuation of the Containment Purge and Exhaust Isolation System to be credited for this Surveillance. Allowing credit to be taken for an actual system actuation is acceptable since the components are not capable of discriminating between an actual and test signal. Allowing credit to be taken for an actual ESFAS actuation is consistent with PVNGS current operating practice. This change has no impact on safety. This change is consistent with NUREG-1432.

L.2 CTS 4.9.9 requires that containment purge valve automatic isolation be demonstrated within 72 hours prior to the start of and at least once per 7 days during Core Alterations or movement of irradiated fuel assemblies in containment. ITS SR 3.9.3.2 requires that the same Surveillance be performed once per 18 months. Performing this Surveillance once per 18 months is acceptable since the potential for pressurizing containment during a refueling accident are greatly reduced compared to the potential in Modes 1, 2, 3, and 4. Automatic actuation of containment purge valves and other automatic containment isolation valves is required to be verified once per 18 months for containment isolation in MODES 1, 2, 3, and 4. Increasing the interval for the refueling Surveillance of the containment purge valves is consistent with the isolation requirements. This change is consistent with NUREG-1432.

TECHNICAL CHANGES - LESS RESTRICTIVE (continued)

L.3 CTS 3.9.4.c.1 requires piping penetrations (other than purge valve penetrations) which provide direct access from the containment atmosphere to outside atmosphere to be closed by, "... an isolation valve, blind flange, or manual valve ... ." ITS LCO 3.9.3.c.1 requires these penetrations to be closed by, "... a manual or automatic isolation valve, blind flange, or equivalent ... ." The intent of the LCO is to assure that radioactive material released within the containment as a result of a fuel handling accident is restricted from release to the outside atmosphere. Using devices equivalent to the other specifically listed devices for containment closure is acceptable due to the lack of potential to pressurize containment under these conditions. This change does not impact safety and is consistent with NUREG-1432.

# NO SIGNIFICANT HAZARDS CONSIDERATION
# SPECIFICATION 3.9.3

ADMINISTRATIVE CHANGES
(ITS 3.9.3 Discussion of Changes Labeled A.1, A.2, A.3, A.4, and A.5)

Arizona Public Service Company, Palo Verde Nuclear Generating Station (PVNGS), Units 1, 2, and 3, is converting to the ITS as outlined in NUREG-1432, "Standard Technical Specifications, Combustion Engineering Plants." The proposed changes involve the reformatting, renumbering, rewording of the Technical Specifications (TS) and Bases with no change in intent, and the incorporation of current operating practices consistent with NUREG-1432. These changes, since they do not involve technical changes to the Current TS (CTS), are administrative. Below are the No Significant Hazards Consideration (NSHC) for the conversion of this Section/Chapter to NUREG-1432.

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 a proposed amendment, would not 1) involve a significant increase in the probability or consequences of an accident previously evaluated; 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?

> The proposed changes involve reformatting, renumbering, and rewording of the CTS and Bases along with incorporation of PVNGS current operating practices and other changes to the CTS as discussed in the specific Discussion of Changes listed above in order to be consistent with NUREG-1432. The reformatting, renumbering, and rewording along with the other changes listed above, involves no technical changes to the CTS. Specifically, there will be no change in the requirements imposed on PVNGS due to these changes. During development of NUREG-1432, certain wording preferences or English language conventions were adopted. The proposed changes to this Section/Chapter are administrative in nature and do not impact initiators of any analyzed events. They also do not impact the assumed mitigation of accidents or transient events. Therefore, these changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

ADMINISTRATIVE CHANGES
(ITS 3.9.3 Discussion of Changes Labeled (A.1, A.2, A.3, A.4, and A.5) (continued)

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

The proposed changes involve reformatting, renumbering, and rewording of the CTS, along with the incorporation of PVNGS current operating practices and other changes, as discussed, in order to be consistent with NUREG-1432. The proposed changes do not involve a physical alteration of the plant (no new or different type of equipment will be installed) or change the methods governing normal plant operation. The proposed changes will not impose any new or different requirements or eliminate any existing requirements. Therefore, these changes do 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?

The proposed changes involve reformatting, renumbering, and rewording of the CTS, along with the incorporation of PVNGS current operating practices and other changes, as discussed, in order to be consistent with NUREG-1432. The proposed changes are administrative in nature and will not involve any technical changes. The proposed changes will not reduce a margin of safety because they have no impact on any safety analysis assumptions. Also, because these changes are administrative in nature, no question of safety is involved. Therefore, these changes do not involve a significant reduction in a margin of safety.

TECHNICAL CHANGES - RELOCATIONS
(ITS 3.9.3 Discussion of Changes Labeled LA.1)

Arizona Public Service Company, Palo Verde Nuclear Generating Station (PVNGS), Units 1, 2, and 3 is converting to the ITS as outlined in NUREG-1432. The proposed changes, since detail is being removed from the CTS to a Licensee Controlled Document, are less restrictive. The descriptions of these changes are in the Discussion of Changes listed above.

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 a proposed amendment, would not 1) involve a significant increase in the probability or consequences of an accident previously evaluated; 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?

> The proposed changes relocate requirements from the CTS to a Licensee Controlled Document. These changes do not result in any hardware changes or changes to plant operating practices. The details being relocated are not assumed to be an initiator of any analyzed event. The Licensee Controlled Document containing the relocated requirements will be maintained using the provisions of 10 CFR 50.59 or other specified control processes and is subject to the change control process in the Administrative Controls Section of the ITS. Since any changes to a Licensee Controlled Document will be evaluated, no increase in the probability or consequences of an accident previously evaluated will be allowed. Therefore, these changes will not involve a significant increase in the probability or consequences of an accident previously evaluated.

TECHNICAL CHANGES - RELOCATIONS
(ITS 3.9.3 Discussion of Changes Labeled LA.1) (continued)

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

The proposed changes relocate requirements from the CTS to a Licensee Controlled Document. These changes will not alter the plant configuration (no new or different type of equipment will be installed) or change the methods governing normal plant operation. These changes will not impose different requirements and adequate control of information will still be maintained. These changes will not alter assumptions made in the safety analysis or licensing basis. Therefore, these changes will 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?

The proposed changes relocate requirements from the CTS to a Licensee Controlled Document. These changes will not reduce a margin of safety since they have no impact on any safety analysis assumptions. In addition, the requirements to be transposed from the CTS to the Licensee Controlled Document are the same as the CTS. Since any future changes to this Licensee Controlled Document will be evaluated per the requirements of 10 CFR 50.59, or other specified control processes, no reduction (significant or insignificant) in a margin of safety will be allowed. Therefore, these changes will not involve a significant reduction in a margin of safety.

The NRC review provides a certain margin of safety, and although this review will no longer be performed prior to submittal, the NRC still inspects the 10 CFR 50.59 process. The proposed changes are consistent with NUREG-1432, which was approved by the NRC Staff. The change controls for proposed relocated details and requirements provide an acceptable level of regulatory authority. Revising the CTS to reflect the approved level of detail per NUREG-1432 reinforces the conclusion that there is not a significant reduction in the margin of safety. Therefore, revising the CTS to reflect the NRC accepted level of detail and requirements ensures no reduction in a margin of safety.

TECHNICAL CHANGES - LESS RESTRICTIVE
(ITS 3.9.3 Discussion of Changes Labeled L.1)

Arizona Public Service Company, Palo Verde Nuclear Generating Station (PVNGS), Units 1, 2, and 3 is converting to the ITS as outlined in NUREG-1432. The proposed change involves making the CTS less restrictive. Below is the description of this less restrictive change and the NSHC for the conversion to NUREG 1432.

L.1 CTS 4.9.9 requires the containment purge valves to be surveilled to assure that isolation occurs on manual initiation and on CPIAS. ITS SR 3.9.3.2 requires that automatic actuation be verified upon receipt of an actual or simulated actuation signal. CTS 4.9.9 requires that the Surveillance be performed using both a manual actuation and a CPIAS test signal. ITS SR 3.9.3.2 does not require the containment purge isolation valves to be tested using manual actuation. Testing of manual actuation is not required since the LCO implies that automatic actuation is required if the valves are open. ITS SR 3.9.3.2 also allows the flexibility of using an actual actuation of the Containment Purge and Exhaust Isolation System to be credited for this Surveillance. Allowing credit to be taken for an actual system actuation is acceptable since the components are not capable of discriminating between an actual and test signal. Allowing credit to be taken for an actual ESFAS actuation is consistent with PVNGS current operating practice. This change has no impact on safety. This change is consistent with NUREG-1432.

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 a proposed amendment, would not 1) involve a significant increase in the probability or consequences of an accident previously evaluated; 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:

TECHNICAL CHANGES - LESS RESTRICTIVE
(ITS 3.9.3 Discussion of Changes Labeled L.1) (continued)

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

The proposed change modifies the SR for containment purge valves during refueling operations by removing the requirement to test the system with a manual initiation and allowing the use of an actual or simulated CPIAS to test the automatic actuation of the purge valves. The purge valves are required to be tested in accordance with the Inservice Testing Program (reference Specification 5.5.8). The stroke timing required by this program is adequate to assure proper function of the manual actuation. Allowing the use of an actual CPIAS allows PVNGS to take credit for purge valve SRs if actuation occurs in the event an actual actuation signal is received. In this case, if an actual signal is received and the containment purge isolation valves properly actuate per the safety analysis, there would be no need to perform the SRs at the original interval. PVNGS would have the option to either start the SR frequency from receipt of the actual actuation or to retest the purge valves at the 18 month Frequency. Operability is adequately demonstrated in either case since the components are not capable of discriminating between an actual or simulated actuation signal.

This change is consistent with NUREG-1432. This change does not result in any hardware changes or changes to plant operating practices nor does it affect plant operation. Therefore this change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

TECHNICAL CHANGES - LESS RESTRICTIVE
(ITS 3.9.3 Discussion of Changes Labeled L.1) (continued)

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

The proposed change modifies the SR for containment purge valves during refueling operations by removing the requirement to test the system with a manual initiation and allowing the use of an actual or simulated CPIAS to test the automatic actuation of the purge valves. The purge valves are required to be tested in accordance with the Inservice Testing Program (reference Specification 5.5.8). The stroke timing required by this program is adequate to assure proper function of the manual actuation. Allowing the use of an actual CPIAS allows PVNGS to take credit for purge valve SRs if actuation occurs in the event an actual actuation signal is received. In this case, if an actual signal is received and the containment purge isolation valves properly actuate per the safety analysis, there would be no need to perform the SRs at the original interval. PVNGS would have the option to either start the SR frequency from receipt of the actual actuation or to retest the purge valves at the 18 month Frequency. Operability is adequately demonstrated in either case since the components are not capable of discriminating between an actual or simulated actuation signal.

This change is consistent with NUREG-1432. This change will not alter the plant configuration (no new or different type of equipment will be installed) or change the methods of governing normal plant operation. This change will not alter assumptions made in the safety analysis or licensing basis. Therefore, this change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

TECHNICAL CHANGES - LESS RESTRICTIVE
(ITS 3.9.3 Discussion of Changes Labeled L.1) (continued)

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

The proposed change modifies the SR for containment purge valves during refueling operations by removing the requirement to test the system with a manual initiation and allowing the use of an actual or simulated CPIAS to test the automatic actuation of the purge valves. The purge valves are required to be tested in accordance with the Inservice Testing Program (reference Specification 5.5.8). The stroke timing required by this program is adequate to assure proper function of the manual actuation. Allowing the use of an actual CPIAS allows PVNGS to take credit for purge valve SRs if actuation occurs in the event an actual actuation signal is received. In this case, if an actual signal is received and the containment purge isolation valves properly actuate per the safety analysis, there would be no need to perform the SRs at the original interval. PVNGS would have the option to either start the SR frequency from receipt of the actual actuation or to retest the purge valves at the 18 month Frequency. Operability is adequately demonstrated in either case since the components are not capable of discriminating between an actual or simulated actuation signal.

This change will not reduce a margin of safety since it has no impact on safety analysis assumptions. This change is consistent with NUREG-1432, which was approved by the NRC Staff. Therefore, this change does not result in a reduction in a margin of safety.

TECHNICAL CHANGES - LESS RESTRICTIVE
(ITS 3.9.3 Discussion of Changes Labeled L.2)

Arizona Public Service Company, Palo Verde Nuclear Generating Station (PVNGS), Units 1, 2, and 3 is converting to the ITS as outlined in NUREG-1432. The proposed change involves making the CTS less restrictive. Below is the description of this less restrictive change and the NSHC for conversion to NUREG-1432.

L.2 CTS 4.9.9 requires that containment purge valve automatic isolation be demonstrated within 72 hours prior to the start of and at least once per 7 days during Core Alterations or movement of irradiated fuel assemblies in containment. ITS SR 3.9.3.2 requires that the same Surveillance be performed once per 18 months. Performing this Surveillance once per 18 months is acceptable since the potential for pressurizing containment during a refueling accident are greatly reduced compared to the potential in Modes 1, 2, 3, and 4. Automatic actuation of containment purge valves and other automatic containment isolation valves is required to be verified once per 18 months for containment isolation in MODES 1, 2, 3, and 4. Increasing the interval for the refueling Surveillance of the containment purge valves is consistent with the isolation requirements. This change is consistent with NUREG-1432.

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 a proposed amendment, would not 1) involve a significant increase in the probability or consequences of an accident previously evaluated; 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:

TECHNICAL CHANGES - LESS RESTRICTIVE
(ITS 3.9.3 Discussion of Changes Labeled L.2) (continued)

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

The proposed change modifies the frequency specified for the containment purge isolation valve SRs during refueling operations. The current Frequency for verification of automatic actuation is within 72 hours prior to the start of Core Alterations or movement of irradiated fuel in the containment building and every 7 days thereafter. This Frequency has been changed to 18 months in the ITS. The containment purge isolation valves are required to close automatically in the event of a fuel handling accident to restrict the escape of containment atmosphere. Since the potential for pressurizing the containment is greatly decreased during refueling operations compared to power operations, the controls required to close the containment should also be reduced. Testing the automatic actuation of the containment purge isolation valves on an 18 month Frequency is consistent with the requirements placed on other automatic CIVs during power operation. This Frequency has proven to be adequate to demonstrate the Operability of these CIVs. There is no reason to suspect that the purge valves will degrade any faster in the less severe environment that exists during refueling. Therefore, applying an 18 month Frequency to these valves is acceptable and is consistent with other SRs of this type.

This change is consistent with NUREG-1432. This change does not result in any hardware changes or changes to plant operating practices nor does it affect plant operation. Therefore this change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

TECHNICAL CHANGES - LESS RESTRICTIVE
(ITS 3.9.3 Discussion of Changes Labeled L.2) (continued)

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

The proposed change modifies the frequency specified for the containment purge isolation valve SRs during refueling operations. The current Frequency for verification of automatic actuation is within 72 hours prior to the start of Core Alterations or movement of irradiated fuel in the containment building and every 7 days thereafter. This Frequency has been changed to 18 months in the ITS. The containment purge isolation valves are required to close automatically in the event of a fuel handling accident to restrict the escape of containment atmosphere. Since the potential for pressurizing the containment is greatly decreased during refueling operations compared to power operations, the controls required to close the containment should also be reduced. Testing the automatic actuation of the containment purge isolation valves on an 18 month Frequency is consistent with the requirements placed on other automatic CIVs during power operation. This Frequency has proven to be adequate to demonstrate the Operability of these CIVs. There is no reason to suspect that the purge valves will degrade any faster in the less severe environment that exists during refueling. Therefore, applying an 18 month Frequency to these valves is acceptable and is consistent with other SRs of this type.

This change is consistent with NUREG-1432. This change will not alter the plant configuration (no new or different type of equipment will be installed) or change the methods of governing normal plant operation. This change will not alter assumptions made in the safety analysis or licensing basis. Therefore, this change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

TECHNICAL CHANGES - LESS RESTRICTIVE
(ITS 3.9.3 Discussion of Changes Labeled L.2) (continued)

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

The proposed change modifies the frequency specified for the containment purge isolation valve SRs during refueling operations. The current Frequency for verification of automatic actuation is within 72 hours prior to the start of Core Alterations or movement of irradiated fuel in the containment building and every 7 days thereafter. This Frequency has been changed to 18 months in the ITS. The containment purge isolation valves are required to close automatically in the event of a fuel handling accident to restrict the escape of containment atmosphere. Since the potential for pressurizing the containment is greatly decreased during refueling operations compared to power operations, the controls required to close the containment should also be reduced. Testing the automatic actuation of the containment purge isolation valves on an 18 month Frequency is consistent with the requirements placed on other automatic CIVs during power operation. This Frequency has proven to be adequate to demonstrate the Operability of these CIVs. There is no reason to suspect that the purge valves will degrade any faster in the less severe environment that exists during refueling. Therefore, applying an 18 month Frequency to these valves is acceptable and is consistent with other SRs of this type.

This change will not reduce a margin of safety since it has no impact on safety analysis assumptions. This change is consistent with NUREG-1432, which was approved by the NRC Staff. Therefore, this change does not result in a reduction in a margin of safety.

TECHNICAL CHANGES - LESS RESTRICTIVE
(ITS 3.9.3 Discussion of Changes Labeled L.3)

Arizona Public Service Company, Palo Verde Nuclear Generating Station (PVNGS), Units 1, 2, and 3 is converting to the ITS as outlined in NUREG-1432. The proposed change involves making the CTS less restrictive. Below is the description of this less restrictive change and the NSHC for conversion to NUREG-1432.

L.3 CTS 3.9.4.c.1 requires piping penetrations (other than purge valve penetrations) which provide direct access from the containment atmosphere to outside atmosphere to be closed by, "... an isolation valve, blind flange, or manual valve ..." ITS LCO 3.9.3.c.1 requires these penetrations to be closed by, "... a manual or automatic isolation valve, blind flange, or equivalent ..." The intent of the LCO is to assure that radioactive material released within the containment as a result of a fuel handling accident is restricted from release to the outside atmosphere. Using devices equivalent to the other specifically listed devices for containment closure is acceptable due to the lack of potential to pressurize containment under these conditions. This change does not impact safety and is consistent with NUREG-1432.

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 a proposed amendment, would not 1) involve a significant increase in the probability or consequences of an accident previously evaluated; 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:

TECHNICAL CHANGES - LESS RESTRICTIVE
(ITS 3.9.3 Discussion of Changes Labeled L.3) (continued)

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

The proposed change allows the use of equivalent devices for containment closure. In the event of a fuel handling accident, the containment must be closed to restrict containment atmosphere from escaping to the outside atmosphere. Because the potential for pressurizing containment during this type of event is minimal, the requirements for containment closure are much less stringent than those for containment isolation. Devices other than the permanent valves, blind flanges or other seals can provide an acceptable barrier for containment closure. These alternate devices must be evaluated and determined to be equivalent prior to use.

This change is consistent with NUREG-1432. This change does not result in any hardware changes or changes to plant operating practices nor does it affect plant operation. Therefore this change will not involve a significant increase in the probability or 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?

The proposed change allows the use of equivalent devices for containment closure. In the event of a fuel handling accident, the containment must be closed to restrict containment atmosphere from escaping to the outside atmosphere. Because the potential for pressurizing containment during this type of event is minimal, the requirements for containment closure are much less stringent than those for containment isolation. Devices other than the permanent valves, blind flanges or other seals can provide an acceptable barrier for containment closure. These alternate devices must be evaluated and determined to be equivalent prior to use.

This change is consistent with NUREG-1432. This change will not alter the plant configuration (no new or different type of equipment will be installed) or change the methods of governing normal plant operation. This change will not alter assumptions made in the safety analysis or licensing basis. Therefore, this change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

TECHNICAL CHANGES - LESS RESTRICTIVE
(ITS 3.9.3 Discussion of Changes Labeled L.3) (continued)

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

The proposed change allows the use of equivalent devices for containment closure. In the event of a fuel handling accident, the containment must be closed to restrict containment atmosphere from escaping to the outside atmosphere. Because the potential for pressurizing containment during this type of event is minimal, the requirements for containment closure are much less stringent than those for containment isolation. Devices other than the permanent valves, blind flanges or other seals can provide an acceptable barrier for containment closure. These alternate devices must be evaluated and determined to be equivalent prior to use.

This change will not reduce a margin of safety since it has no impact on safety analysis assumptions. This change is consistent with NUREG-1432, which was approved by the NRC Staff. Therefore, this change does not result in a reduction in a margin of safety.

<DOC>
<CTS>

## 3.9 REFUELING OPERATIONS

### 3.9.4 Shutdown Cooling (SDC) and Coolant Circulation—High Water Level

<3.9.8.1> LCO 3.9.4 One SDC loop shall be OPERABLE and in operation. (3)

<3.9.8.1"*">
<DOC M.1>

-----------------------------NOTE-----------------------------
The required SDC loop may be removed from operation for ≤ 1 hour per ~~8~~ hour period, provided no operations are permitted that would cause reduction of the Reactor Coolant System boron concentration.

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APPLICABILITY: MODE 6 with the water level ≥ 23 ft above the top of reactor vessel flange.

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| <3.9.8.1 ACT> A. SDC loop requirements not met. | A.1 Suspend operations involving a reduction in reactor coolant boron concentration. | Immediately |
| | AND | |
| <DOC A.3> | A.2 Suspend loading irradiated fuel assemblies in the core. | Immediately |
| | AND | |
| | A.3 Initiate action to satisfy SDC loop requirements. | Immediately |
| | AND | |
| | | (continued) |

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ACTIONS

| CONDITION | REQUIRED ACTION | | COMPLETION TIME |
|---|---|---|---|
| <3.9.8.1ACT> A. (continued) | A.4 | Close all containment penetrations providing direct access from containment atmosphere to outside atmosphere. | 4 hours |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| <4.9.8.1> SR 3.9.4.1 | Verify one SDC loop is OPERABLE and in operation ~~and circulating reactor coolant at a flow rate of ≥ [2200] gpm.~~ 3780 (2) (4) | 12 hours |

CE STS
NUREG-1432 REV. 1
SPECIFICATION 3.9.4
BASES MARK UP

## B 3.9 REFUELING OPERATIONS

## B 3.9.4 Shutdown Cooling (SDC) and Coolant Circulation—High Water Level

BASES

BACKGROUND

The purposes of the SDC System in MODE 6 are to remove decay heat and sensible heat from the Reactor Coolant System (RCS), as required by GDC 34, to provide mixing of borated coolant, to provide sufficient coolant circulation to minimize the effects of a boron dilution accident, and to prevent boron stratification (Ref. 1). Heat is removed from the RCS by circulating reactor coolant through the SDC heat exchanger(s), where the heat is transferred to the Component Essential Cooling Water System via the SDC heat exchanger(s). The coolant is then returned to the RCS via the RCS cold leg(s). Operation of the SDC System for normal cooldown or decay heat removal is manually accomplished from the control room. The heat removal rate is adjusted by controlling the flow of reactor coolant through the SDC heat exchanger(s) and bypassing the heat exchanger(s). Mixing of the reactor coolant is maintained by this continuous circulation of reactor coolant through the SDC System.

2

APPLICABLE SAFETY ANALYSES

If the reactor coolant temperature is not maintained below 200°F, boiling of the reactor coolant could result. This could lead to inadequate cooling of the reactor fuel due to a resulting loss of coolant in the reactor vessel. Additionally, boiling of the reactor coolant could lead to a reduction in boron concentration in the coolant due to the boron plating out on components near the areas of the boiling activity, and because of the possible addition of water to the reactor vessel with a lower boron concentration than is required to keep the reactor subcritical. The loss of reactor coolant and the reduction of boron concentration in the reactor coolant would eventually challenge the integrity of the fuel cladding, which is a fission product barrier. One train of the SDC System is required to be operational in MODE 6, with the water level ≥ 23 ft above the top of the reactor vessel flange, to prevent this challenge. The LCO does permit de-energizing of the SDC pump for short durations under the condition that the boron concentration is not diluted. This conditional de-energizing of the SDC pump does not result in a challenge to the fission product barrier.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

SDC and Coolant Circulation—High Water Level satisfies Criterion 2 of the NRC Policy Statement.
10 CFR 50.36 (c)(2)(ii) 1

LCO

Only one SDC loop is required for decay heat removal in MODE 6, with water level ≥ 23 ft above the top of the reactor vessel flange. Only one SDC loop is required because the volume of water above the reactor vessel flange provides backup decay heat removal capability. At least one SDC loop must be in operation to provide:

a. Removal of decay heat;

b. Mixing of borated coolant to minimize the possibility of a criticality; and

c. Indication of reactor coolant temperature.

An OPERABLE SDC loop includes an SDC pump, a heat exchanger, valves, piping, instruments, and controls to ensure an OPERABLE flow path and to determine the low end temperature. The flow path starts in one of the RCS hot legs and is returned to the RCS cold legs.

The LCO is modified by a Note that allows the required operating SDC loop to be removed from service for up to 1 hour in each 8 hour period, provided no operations are permitted that would cause a reduction of the RCS boron concentration. Boron concentration reduction is prohibited because uniform concentration distribution cannot be ensured without forced circulation. This permits operations such as core mapping or alterations in the vicinity of the reactor vessel hot leg nozzles, and RCS to SDC isolation valve testing. During this 1 hour period, decay heat is removed by natural convection to the large mass of water in the refueling cavity.
canal 2

APPLICABILITY

One SDC loop must be in operation in MODE 6, with the water level ≥ 23 ft above the top of the reactor vessel flange, to provide decay heat removal. The 23 ft level was selected because it corresponds to the 23 ft requirement established for fuel movement in LCO 3.9.6, "Refueling Water Level." Fuel Assemblies 2

(continued)

BASES

APPLICABILITY (continued)

Requirements for the SDC System in other MODES are covered by LCOs in Section 3.4, Reactor Coolant System (RCS), and Section 3.5, Emergency Core Cooling Systems (ECCS). SDC loop requirements in MODE 6, with the water level < 23 ft above the top of the reactor vessel flange, are located in LCO 3.9.5, "Shutdown Cooling (SDC) and Coolant Circulation—Low Water Level."

ACTIONS

SDC loop requirements are met by having one SDC loop OPERABLE and in operation, except as permitted in the Note to the LCO.

A.1

If SDC loop requirements are not met, there will be no forced circulation to provide mixing to establish uniform boron concentrations. Reduced boron concentrations can occur through the addition of water with a lower boron concentration than that contained in the RCS. Therefore, actions that reduce boron concentration shall be suspended immediately.

A.2

If SDC loop requirements are not met, actions shall be taken immediately to suspend loading irradiated fuel assemblies in the core. With no forced circulation cooling, decay heat removal from the core occurs by natural convection to the heat sink provided by the water above the core. A minimum refueling water level of 23 ft above the reactor vessel flange provides an adequate available heat sink. Suspending any operation that would increase the decay heat load, such as loading ~~a~~ an irradiated fuel assembly, is a prudent action under this condition. 2

A.3

If SDC loop requirements are not met, actions shall be initiated and continued in order to satisfy SDC loop requirements.

(continued)

BASES

ACTIONS (continued)

A.4

If SDC loop requirements are not met, all containment penetrations to the outside atmosphere must be closed to prevent fission products, if released by a loss of decay heat event, from escaping the containment building. The 4 hour Completion Time allows fixing most SDC problems without incurring the additional action of violating the containment atmosphere.

2 4

SURVEILLANCE REQUIREMENTS

SR 3.9.4.1

This Surveillance demonstrates that the SDC loop is in operation and circulating reactor coolant at a flowrate of greater than or equal to 3780 gpm. The flow rate is determined by the flow rate necessary to provide sufficient decay heat removal capability and to prevent thermal and boron stratification in the core. The Frequency of 12 hours is sufficient, considering the flow, temperature, pump control, and alarm indications available to the operator in the control room for monitoring the SDC System.

REFERENCES

UFSAR ~~1. FSAR~~, Section ~~X.X.~~ 5.4.7

1

NUREG-1432 EXCEPTIONS
SPECIFICATION 3.9.4

1. Grammar and/or editorial changes have been made to enhance clarity. No technical or intent changes to the Specification are made by this change.

2. The plant specific titles, nomenclature, number parameter/value, reference, system description, system design, operating practices or analysis description was used (additions, deletions, and/or changes are included). Plant specific parameters/values were directly transferred from the CTS to the ITS, or from the plant design basis to the ITS. The Bases have been revised to be consistent with the LCO/Surveillance.

3. CTS 3.9.8.1 requires that one shutdown cooling loop be Operable and in operation. NUREG LCO 3.9.4 requires that one SDC loop be in operation. The Bases for NUREG LCO 3.9.4 indicates that one SDC loop must be operating and that it must be Operable. ITS LCO 3.9.4 requires that one SDC loop be Operable and in operation. This change corrects an inconsistency between the LCO and Bases. This change is also consistent with the current licensing basis.

4. NUREG SR 3.9.4.1 specifies the quantitative value for the flowrate of the SDC loop. DOC LA.2 relocates this value to the ITS Bases since adequate controls exist per 10 CFR 50.59 and TS Bases Control Program. Removal of this detail makes ITS SR 3.9.4.1 consistent with the level of detail of ITS SR 3.9.5.1, which also requires a SDC loop to be in operation and specifies the quantitative value of the flowrate in the ITS Bases. This is acceptable since it corrects an inconsistency between two ITS Specifications and because it does not change the surveillance requirement to verify an SDC loop is in operation. It merely relocates the quantitative value of the flowrate to the Bases.

PVNGS CTS
SPECIFICATION 3.9.4
MARK UP

Specification 3.9.4

A.1

3.9 REFUELING OPERATIONS

3.9.4 3/4.9.8 SHUTDOWN COOLING (SDC) AND COOLANT CIRCULATION (–)

HIGH WATER LEVEL

~~LIMITING CONDITION FOR OPERATION~~

LCO3.9.4 ~~3.9.8.1~~ At least one shutdown cooling loop shall be OPERABLE and in operation*.

APPLICABILITY: MODE 6 when the water level above the top of the reactor pressure vessel flange is greater than or equal to 23 feet.

ACT A ACTION: A.2

With no shutdown cooling loop OPERABLE and in operation, suspend ~~all~~ loading of irradiated fuel in the core, and operations involving an ~~increase in~~ the ~~reactor decay heat load or a~~ reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required shutdown cooling loop to OPERABLE and operating status ~~as soon as possible~~. Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.

SURVEILLANCE REQUIREMENTS

SR3.9.4.1 4.9.8.1 ~~At least~~ one shutdown cooling loop IS OPERABLE and shall be verified to be in operation, and circulating reactor coolant at a flow rate of greater than or equal to 3780 gpm at least once per 12 hours. LA.2

NOTE *The required shutdown cooling loop may be removed from operation for up to 1 hour per 8-hour period during the performance of CORE ALTERATIONS in the vicinity of the reactor pressure vessel hot legs or during surveillance testing of ECCS pumps. LA.1

, provided no operations are permitted that would cause reduction of the Reactor Coolant System boron concentration. M.1

# DISCUSSION OF CHANGES
# SPECIFICATION 3.9.4

ADMINISTRATIVE CHANGES

A.1 All reformatting and renumbering is in accordance with the Combustion Engineering Plant (CEOG) Standard Technical Specifications NUREG-1432, Rev. 1 (NUREG-1432). As a result, the Palo Verde Nuclear Generating Station (PVNGS) Improved Technical Specifications (ITS) should be more readable, and therefore understandable, by plant operators as well as other users. During the reformatting and renumbering of the ITS, no technical changes (either actual or interpretational) to the Current Technical Specification (CTS) were made unless they were identified and justified.

Editorial rewording (either adding or deleting) is made consistent with NUREG-1432. During NUREG-1432 development, certain wording preferences or English language conventions were adopted which resulted in no technical changes (either actual or interpretational) to the CTS.

Additional information has also been added to more fully describe each subsection. This wording is consistent with NUREG-1432. Since the design is already approved by the NRC, adding more detail does not result in a technical change.

A.2 CTS 3.9.8.1 Action states in part, "... suspend all operations involving an increase in the reactor decay heat load or a reduction in boron concentration ...." ITS LCO 3.9.4 Required Actions A.1 and A.2 require suspending operations involving a reduction in reactor coolant boron concentration, and suspending loading of irradiated fuel assemblies in the core. The requirement to suspend operations involving an increase in decay heat load is equivalent to suspending reload of irradiated fuel since that is the only way to increase decay heat load. Changing the logical connector from "or" to "and" clarifies the intent of the Actions that any which are applicable be applied. This clarification does not alter the intent or application of the TS. This change does not impact safety and is consistent with NUREG-1432.

TECHNICAL CHANGES - MORE RESTRICTIVE

M.1 CTS 3.9.8.1 LCO statement is modified by an asterisk which allows removing the SDC loop from operation for up to 1 hour per 8 hour period. ITS LCO 3.9.4 also contains this provision as a Note but is restricted by the statement, "... provided no operations are permitted that would cause reduction of the Reactor Coolant System boron concentration." Suspending operations that could cause a reduction in boron concentration during the 1 hour when shutdown cooling is allowed out of service is reasonable since there is no other means of forced circulation available to assure uniform boron concentration. The addition of this requirement constitutes a more restrictive change to PVNGS CTS requirements. This change is consistent with NUREG-1432.

TECHNICAL CHANGES - RELOCATIONS

LA.1 CTS 3.9.8.1 LCO statement is modified by an asterisk which states, "... The shutdown cooling loop may be removed from operation for up to 1 hour per 8-hour period during the performance of CORE ALTERATIONS in the vicinity of the reactor pressure vessel hot legs or during surveillance testing of ECCS pumps." ITS LCO 3.9.4 is modified by a Note which also allows the shutdown cooling loop to be removed from operation for up to 1 hour per 8 hour period but does not specify the activities during which this exclusion is allowed. During the 1 hour period when shutdown cooling is not in operation, no reduction in boron concentration is allowed as discussed in DOC M.1. Decay heat removal during this period is provided by natural convection to the large mass of water in the refueling pool. With adequate decay heat removal, and no means of boron reduction, it is acceptable to allow the shutdown cooling loop to be removed from operation for reasons other than those allowed by CTS 3.9.8.1. In this condition, there would not be sufficient decay heat to drastically increase the temperature of the total volume of water over a 1 hour period. These requirements are not required to determine the OPERABILITY of a system, component or structure and therefore are being relocated to the Bases.

Any change to the requirements in the Bases will be governed by the provisions of the Bases Control Program. This provides an equivalent level of regulatory control and is an administrative change with no impact on the margin of safety. This requirement does not need to be in the ITS to provide adequate protection to the public health and safety. Therefore, relocation of this requirement to the Bases is acceptable and is consistent with NUREG-1432.

TECHNICAL CHANGES - RELOCATIONS (continued)

LA.2 CTS 4.9.8.1 states in part, "At least one shutdown cooling loop shall be verified to be in operation and circulating reactor coolant at a flowrate of greater than or equal to 3780 gpm..." ITS SR 3.9.4.1 requires verification that one SDC loop is operable and in operation. Removal of the quantitative value of the flowrate is acceptable since there are adequate controls of the TS Bases to ensure that changes are appropriately evaluated.

The bases for this specification indicates that the pumps are required Operable in order to maintain OPERABILITY of the SDC System. If the pumps are Operable and the system is properly aligned, the resulting flowrate will be sufficient to assure adequate mixing and temperature control of the RCS. These requirements are being relocated to the ITS Bases.

Any change to the requirements in the Bases will be governed by the provisions of the TS Bases Control Program. This provides an equivalent level of control and is an administrative change with no impact on the margin of safety. This requirement does not need to be in the ITS to provide adequate protection to the public health and safety. Therefore, relocation of this requirement is acceptable and is consistent with NUREG-1432.

TECHNICAL CHANGES - LESS RESTRICTIVE

None

# NO SIGNIFICANT HAZARDS CONSIDERATION
# SPECIFICATION 3.9.4

ADMINISTRATIVE CHANGES
(ITS 3.9.4 Discussion of Changes Labeled A.1 and A.2)

Arizona Public Service Company, Palo Verde Nuclear Generating Station (PVNGS), Units 1, 2, and 3, is converting to the ITS as outlined in NUREG-1432, "Standard Technical Specifications, Combustion Engineering Plants." The proposed changes involve the reformatting, renumbering, rewording of the Technical Specifications (TS) and Bases with no change in intent, and the incorporation of current operating practices consistent with NUREG-1432. These changes, since they do not involve technical changes to the Current TS (CTS), are administrative. Below are the No Significant Hazards Consideration (NSHC) for the conversion of this Section/Chapter to NUREG-1432.

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 a proposed amendment, would not 1) involve a significant increase in the probability or consequences of an accident previously evaluated; 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?

> The proposed changes involve reformatting, renumbering, and rewording of the CTS and Bases along with incorporation of PVNGS current operating practices and other changes to the CTS as discussed in the specific Discussion of Changes listed above in order to be consistent with NUREG-1432. The reformatting, renumbering, and rewording along with the other changes listed above, involves no technical changes to the CTS. Specifically, there will be no change in the requirements imposed on PVNGS due to these changes. During development of NUREG-1432, certain wording preferences or English language conventions were adopted. The proposed changes to this Section/Chapter are administrative in nature and do not impact initiators of any analyzed events. They also do not impact the assumed mitigation of accidents or transient events. Therefore, these changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

ADMINISTRATIVE CHANGES
(ITS 3.9.4 Discussion of Changes Labeled (A.1 and A.2) (continued)

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

The proposed changes involve reformatting, renumbering, and rewording of the CTS, along with the incorporation of PVNGS current operating practices and other changes, as discussed, in order to be consistent with NUREG-1432. The proposed changes do not involve a physical alteration of the plant (no new or different type of equipment will be installed) or change the methods governing normal plant operation. The proposed changes will not impose any new or different requirements or eliminate any existing requirements. Therefore, these changes do 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?

The proposed changes involve reformatting, renumbering, and rewording of the CTS, along with the incorporation of PVNGS current operating practices and other changes, as discussed, in order to be consistent with NUREG-1432. The proposed changes are administrative in nature and will not involve any technical changes. The proposed changes will not reduce a margin of safety because they have no impact on any safety analysis assumptions. Also, because these changes are administrative in nature, no question of safety is involved. Therefore, these changes do not involve a significant reduction in a margin of safety.

TECHNICAL CHANGES - MORE RESTRICTIVE
(ITS 3.9.4 Discussion of Changes Labeled M.1)

Arizona Public Service Company, Palo Verde Nuclear Generating Station (PVNGS), Units 1, 2, and 3 is converting to the ITS as outlined in NUREG-1432. This particular NSHC is for the changes labeled "Technical Changes - More Restrictive" described in the specific Discussion of Changes listed above. The proposed changes incorporate more restrictive changes into the CTS by either making current requirements more stringent or adding new requirements which currently do not exist.

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 a proposed amendment, would not 1) involve a significant increase in the probability or consequences of an accident previously evaluated; 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?

> The proposed changes provide more stringent requirements than previously existed in the CTS. The more stringent requirements will not result in operation that will increase the probability of initiating an analyzed event. If anything, the new requirements may decrease the probability or consequences of an analyzed event by incorporating the more restrictive changes discussed in the specific Discussion of Changes listed above. These changes will not alter assumptions relative to mitigation of an accident or transient event. The more restrictive requirements will not alter the operation and will continue to ensure process variables, structures, systems, or components are maintained consistent with safety analyses and licensing basis. These changes have been reviewed to ensure that no previously evaluated accident has been adversely affected. Therefore, these changes will not involve a significant increase in the probability or consequences of an accident evaluated.

TECHNICAL CHANGES - MORE RESTRICTIVE
(ITS 3.9.4 Discussion of Changes Labeled M.1) (continued)

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

Making existing requirements more restrictive and adding more restrictive requirements to the CTS will not alter the plant configuration (no new or different type of equipment will be installed) or change the methods governing normal plant operation. These changes do impose different requirements. However, they are consistent with the assumptions made in the safety analyses, licensing basis, and NUREG-1432. Therefore, these changes will 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?

The proposed changes provide more stringent requirements than previously existed in the CTS. An evaluation of these changes concluded that adding these more restrictive requirements either increases or has no impact on the margin of safety. The changes provide additional restrictions which may enhance plant safety. These changes maintain requirements of the safety analysis, licensing basis, and NUREG-1432. As such, no question of safety is involved. Therefore, these changes will not involve a significant reduction in a margin of safety.

TECHNICAL CHANGES - RELOCATIONS
(ITS 3.9.4 Discussion of Changes Labeled LA.1 and LA.2)

Arizona Public Service Company, Palo Verde Nuclear Generating Station (PVNGS), Units 1, 2, and 3 is converting to the ITS as outlined in NUREG-1432. The proposed changes, since detail is being removed from the CTS to a Licensee Controlled Document, are less restrictive. The descriptions of these changes are in the Discussion of Changes listed above.

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 a proposed amendment, would not 1) involve a significant increase in the probability or consequences of an accident previously evaluated; 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?

> The proposed changes relocate requirements from the CTS to a Licensee Controlled Document. These changes do not result in any hardware changes or changes to plant operating practices. The details being relocated are not assumed to be an initiator of any analyzed event. The Licensee Controlled Document containing the relocated requirements will be maintained using the provisions of 10 CFR 50.59 or other specified control processes and is subject to the change control process in the Administrative Controls Section of the ITS. Since any changes to a Licensee Controlled Document will be evaluated, no increase in the probability or consequences of an accident previously evaluated will be allowed. Therefore, these changes will not involve a significant increase in the probability or consequences of an accident previously evaluated.

TECHNICAL CHANGES - RELOCATIONS
(ITS 3.9.4 Discussion of Changes Labeled LA.1 and LA.2) (continued)

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

The proposed changes relocate requirements from the CTS to a Licensee Controlled Document. These changes will not alter the plant configuration (no new or different type of equipment will be installed) or change the methods governing normal plant operation. These changes will not impose different requirements and adequate control of information will still be maintained. These changes will not alter assumptions made in the safety analysis or licensing basis. Therefore, these changes will 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?

The proposed changes relocate requirements from the CTS to a Licensee Controlled Document. These changes will not reduce a margin of safety since they have no impact on any safety analysis assumptions. In addition, the requirements to be transposed from the CTS to the Licensee Controlled Document are the same as the CTS. Since any future changes to this Licensee Controlled Document will be evaluated per the requirements of 10 CFR 50.59, or other specified control processes, no reduction (significant or insignificant) in a margin of safety will be allowed. Therefore, these changes will not involve a significant reduction in a margin of safety.

The NRC review provides a certain margin of safety, and although this review will no longer be performed prior to submittal, the NRC still inspects the 10 CFR 50.59 process. The proposed changes are consistent with NUREG-1432, which was approved by the NRC Staff. The change controls for proposed relocated details and requirements provide an acceptable level of regulatory authority. Revising the CTS to reflect the approved level of detail per NUREG-1432 reinforces the conclusion that there is not a significant reduction in the margin of safety. Therefore, revising the CTS to reflect the NRC accepted level of detail and requirements ensures no reduction in a margin of safety.

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NUREG-1432 REV. 1

SPECIFICATION 3.9.5

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<CTS>

## 3.9 REFUELING OPERATIONS

### 3.9.5 Shutdown Cooling (SDC) and Coolant Circulation—Low Water Level

<3.9.8.2> LCO 3.9.5 Two SDC loops shall be OPERABLE, and one SDC loop shall be in operation.

Insert 1 1

APPLICABILITY: MODE 6 with the water level < 23 ft above the top of reactor vessel flange.

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| <3.9.8.2 ACT a> A. One SDC loop inoperable. | A.1 Initiate action to restore SDC loop to OPERABLE status. | Immediately |
| | OR | |
| | A.2 Initiate action to establish ≥ 23 ft of water above the top of reactor vessel flange. | Immediately |
| <3.9.8.2 ACT b> B. No SDC loop OPERABLE or in operation. | B.1 Suspend operations involving a reduction in reactor coolant boron concentration. | Immediately |
| | AND | |
| <DOC A.3> | B.2 Initiate action to restore one SDC loop to OPERABLE status and to operation. | Immediately |
| | AND | |
| | | (continued) |

INSERT FOR ITS 3.9.5
LCO SECTION

LCO 3.9.5

---------------------------------------------------NOTE---------------------------------------------------

The required SDC loop may be removed from operation for ≤ 1 hour per 8 hour period, provided no operations are permitted that would cause reduction of the Reactor Coolant System boron concentration.

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<DOC>
<CTS>

ACTIONS

| | CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|---|
| <3.9.8.2 ACT b> | B. (continued) | B.3 Close all containment penetrations providing direct access from containment atmosphere to outside atmosphere. | 4 hours |

SURVEILLANCE REQUIREMENTS

| | SURVEILLANCE | | FREQUENCY |
|---|---|---|---|
| <4.9.8.2> <DOC M.2> | SR 3.9.5.1 | Verify required SDC loops are OPERABLE and one SDC loop is in operation. | 12 hours |
| <DOC M.3> | SR 3.9.5.2 | Verify correct breaker alignment and indicated power available to the required SDC pump that is not in operation. | 7 days |

CE STS
NUREG-1432 REV. 1
SPECIFICATION 3.9.5
BASES MARK UP

# B 3.9 REFUELING OPERATIONS

## B 3.9.5 Shutdown Cooling (SDC) and Coolant Circulation—Low Water Level

BASES

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BACKGROUND

The purposes of the SDC System in MODE 6 are to remove decay heat and sensible heat from the Reactor Coolant System (RCS), as required by GDC 34, to provide mixing of borated coolant, to provide sufficient coolant circulation to minimize the effects of a boron dilution accident, and to prevent boron stratification (Ref. 1). Heat is removed from the RCS by circulating reactor coolant through the SDC heat exchanger(s), where the heat is transferred to the Component (Essential) (3) Cooling Water System via the SDC heat exchanger(s). The coolant is then returned to the RCS via the RCS cold leg(s). Operation of the SDC System for normal cooldown or decay heat removal is manually accomplished from the control room. The heat removal rate is adjusted by controlling the flow of reactor coolant through the SDC heat exchanger(s) and bypassing the heat exchanger(s). Mixing of the reactor coolant is maintained by this continuous circulation of reactor coolant through the SDC System.

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APPLICABLE SAFETY ANALYSES

If the reactor coolant temperature is not maintained below 200°F, boiling of the reactor coolant could result. This could lead to inadequate cooling of the reactor fuel due to the resulting loss of coolant in the reactor vessel. Additionally, boiling of the reactor coolant could lead to a reduction in boron concentration in the coolant due to the boron plating out on components near the areas of the boiling activity, and because of the possible addition of water to the reactor vessel with a lower boron concentration than is required to keep the reactor subcritical. The loss of reactor coolant and the reduction of boron concentration in the reactor coolant would eventually challenge the integrity of the fuel cladding, which is a fission product barrier. Two trains of the SDC System are required to be OPERABLE, and one train is required to be in operation in MODE 6, with the water level < 23 ft above the top of the reactor vessel flange, to prevent this challenge.

SDC and Coolant Circulation—Low Water Level satisfies Criterion 2 of the NRC Policy Statement.

10CFR 50.36(c)(2)(ii) (2)

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(continued)

BASES (continued)

LCO

In MODE 6, with the water level < 23 ft above the top of the reactor vessel flange, both SDC loops must be OPERABLE. Additionally, one loop of the SDC System must be in operation in order to provide:

a. Removal of decay heat;

b. Mixing of borated coolant to minimize the possibility of a criticality; and

c. Indication of reactor coolant temperature.

3

Both SDC Pumps may be aligned to the Refueling Water Tank (RWT) to support filling the refueling cavity or for performance of required testing.

An OPERABLE SDC loop consists of an SDC pump, a heat exchanger, valves, piping, instruments, and controls to ensure an OPERABLE flow path and to determine the low end temperature. The flow path starts in one of the RCS hot legs and is returned to the RCS cold legs.

Insert 1 1

APPLICABILITY

Two SDC loops are required to be OPERABLE, and one SDC loop must be in operation in MODE 6, with the water level < 23 ft above the top of the reactor vessel flange, to provide decay heat removal. Requirements for the SDC System in other MODES are covered by LCOs in Section 3.4, Reactor Coolant System. MODE 6 requirements, with a water level ≥ 23 ft above the reactor vessel flange, are covered in LCO 3.9.4, "Shutdown Cooling and Coolant Circulation—High Water Level."

ACTIONS

A.1 and A.2

If one SDC loop is inoperable, action shall be immediately initiated and continued until the SDC loop is restored to OPERABLE status and to operation, or until ≥ 23 ft of water level is established above the reactor vessel flange. When the water level is established at ≥ 23 ft above the reactor vessel flange, the Applicability will change to that of LCO 3.9.4, "Shutdown Cooling and Coolant Circulation—High Water Level," and only one SDC loop is required to be OPERABLE and in operation. An immediate Completion Time is necessary for an operator to initiate corrective actions.

(continued)

INSERT FOR ITS 3.9.5 BASES MARKUP
LCO SECTION

INSERT 1

The LCO is modified by a Note that allows a required operating SDC loop to be removed from service for up to 1 hour in each 8 hour period, provided no operations are permitted that would cause a reduction of the RCS boron concentration. Boron concentration reduction is prohibited because uniform concentration distribution cannot be ensured without forced circulation. This permits operations such as core mapping or alterations in the vicinity of the reactor vessel hot leg nozzles, and RCS to SDC isolation valve testing. During this 1 hour period, decay heat is removed by natural convection to the large mass of water in the refueling cavity.

BASES

ACTIONS (continued)

B.1

If no SDC loop is in operation or no SDC loops are OPERABLE, there will be no forced circulation to provide mixing to establish uniform boron concentrations. Reduced boron concentrations can occur by the addition of water with lower boron concentration than that contained in the RCS. Therefore, actions that reduce boron concentration shall be suspended immediately.

B.2

If no SDC loop is in operation or no SDC loops are OPERABLE, action shall be initiated immediately and continued without interruption to restore one SDC loop to OPERABLE status and operation. Since the unit is in Conditions A and B concurrently, the restoration of two OPERABLE SDC loops and one operating SDC loop should be accomplished expeditiously.

B.3

SDC    or no SDC loops are OPERABLE,

3

If no RHR loop is in operation, all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere must be closed within 4 hours. With the RHR loop requirements not met, the potential exists for the coolant to boil and release radioactive gas to the containment atmosphere. Closing containment penetrations that are open to the outside atmosphere ensures that dose limits are not exceeded.

SDC

The Completion Time of 4 hours is reasonable, based on the low probability of the coolant boiling in that time.

SURVEILLANCE REQUIREMENTS

SR 3.9.5.1

This Surveillance demonstrates that one SDC loop is operating and circulating reactor coolant. The flow rate is determined by the flow rate necessary to provide sufficient decay heat removal capability and to prevent thermal and boron stratification in the core. In addition, this Surveillance demonstrates that the other SDC loop is OPERABLE.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.9.5.1 (continued)

In addition, during operation of the SDC loop with the water level in the vicinity of the reactor vessel nozzles, the SDC loop flow rate determination must also consider the SDC pump suction requirements. The Frequency of 12 hours is sufficient, considering the flow, temperature, pump control, and alarm indications available to the operator to monitor the SDC System in the control room.

Verification that the required loops are OPERABLE and in operation ensures that loops can be placed in operation as needed, to maintain decay heat and retain forced circulation. The Frequency of 12 hours is considered reasonable, since other administrative controls are available and have proven to be acceptable by operating experience.

SR 3.9.5.2 (3) that is not in operation

Verification that the required pump is OPERABLE ensures that an additional SDC pump can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available to the required pump. The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.

REFERENCES UFSAR ~~1. FSAR~~, Section ~~X X.~~ 5.4.7

(1)

NUREG-1432 EXCEPTIONS
SPECIFICATION 3.9.5

**SPECIFICATION 3.9.5 - Shutdown Cooling (SDC) And Coolant Circulation - Low Water Level**

1. NUREG-1432 LCO 3.9.5 does not contain provisions for removing the operating loop of shutdown cooling from service during performance of ECCS pump testing or during Core Alterations in the vicinity of the reactor pressure vessel hot legs. ITS LCO 3.9.5 is modified by a note that allows the operating SDC loop to be removed for up to 1 hour in each 8 hour period, provided no operations are permitted that would cause a reduction of the RCS boron concentration. Boron concentration reduction is prohibited because uniform concentration distribution cannot be ensured without forced circulation. During this 1 hour period, decay heat is removed by natural convection to the water in the refueling cavity. This note is identical to the note contained in NUREG-1432 LCO 3.9.4 and ITS LCO 3.9.4. This change does not impact safety and is consistent with current licensing bases.

2. Grammar and/or editorial changes have been made to enhance clarity. No technical or intent changes to the Specification are made by this change.

3. The plant specific titles, nomenclature, number parameter/value, reference, system description, system design, operating practices or analysis description was used (additions, deletions, and/or changes are included). Plant specific parameters/values were directly transferred from the CTS to the ITS, or from the plant design basis to the ITS. The Bases have been revised to be consistent with the LCO/Surveillance.

PVNGS CTS
SPECIFICATION 3.9.5
MARK UP

Specification 3.9.5

(A.1)

(3.9) REFUELING OPERATIONS

(3.9.5) ~~LOW WATER LEVEL~~ Shutdown Cooling (SDC) and Coolant Circulation—

~~LIMITED CONDITION FOR OPERATION~~

(LCO3.9.5) ~~3.9.8.2~~ Two ~~independent~~ shutdown cooling loops shall be OPERABLE and ~~at least~~ one shutdown cooling loop shall be in operation~~*~~.

APPLICABILITY: MODE 6 when the water level above the top of the reactor pressure vessel flange is less than 23 feet.

ACTION:

(ACT A) ~~a.~~ ~~With less than the required shutdown cooling loops OPERABLE~~ one shutdown cooling loop inoperable, (A.2) immediately initiate corrective action to return the ~~required~~ loop~~s~~ to OPERABLE status, or to establish greater than or equal to 23 feet of water above the reactor pressure vessel flange, ~~as soon as possible.~~

(ACT B) ~~b.~~ With no shutdown cooling loop OPERABLE or (A.3) in operation, immediately (A.5) suspend all operations involving ~~an increase in the reactor decay heat load or~~ a reduction (A.4) in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required shutdown cooling loop to OPERABLE status and to (A.3) operation. Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.

SURVEILLANCE REQUIREMENTS

(SR3.9.5.1) ~~4.9.8.2~~ ~~At least~~ Verify required shutdown cooling loops are OPERABLE and (M.2) one shutdown cooling loop shall be verified to be in operation and circulating reactor coolant at a flow rate of greater than or equal to 3780 gpm at least once per 12 hours. (LA.1)

(SR3.9.5.2) Verify correct breaker alignment and indicated power available to the required SDC pump that is not in operation. (M.3)

(LCO3.9.5 NOTE) ~~*~~The shutdown cooling loop may be removed from operation for up to 1 hour per 8-hour period, provided no operations are permitted that would cause reduction of the Reactor Coolant System boron concentration, (M.1) during the performance of CORE ALTERATIONS in the vicinity of the reactor pressure vessel hot legs or during surveillance testing of ECCS pumps. (LA.2)

# DISCUSSION OF CHANGES
# SPECIFICATION 3.9.5

ADMINISTRATIVE CHANGES

A.1 All reformatting and renumbering is in accordance with the Combustion Engineering Plant (CEOG) Standard Technical Specifications NUREG-1432, Rev. 1 (NUREG-1432). As a result, the Palo Verde Nuclear Generating Station (PVNGS) Improved Technical Specifications (ITS) should be more readable, and therefore understandable, by plant operators as well as other users. During the reformatting and renumbering of the ITS, no technical changes (either actual or interpretational) to the Current Technical Specification (CTS) were made unless they were identified and justified.

Editorial rewording (either adding or deleting) is made consistent with NUREG-1432. During NUREG-1432 development, certain wording preferences or English language conventions were adopted which resulted in no technical changes (either actual or interpretational) to the CTS.

Additional information has also been added to more fully describe each subsection. This wording is consistent with NUREG-1432. Since the design is already approved by the NRC, adding more detail does not result in a technical change.

A.2 CTS 3.9.8.2 Action a states in part, "With less than the required shutdown cooling loops OPERABLE ... ." ITS LCO 3.9.5 Condition A states, "One SDC loop inoperable." There are two shutdown cooling loops and both are required to be OPERABLE by both CTS 3.9.8.2 and ITS LCO 3.9.5. The requirement of CTS 3.9.8.2 Action a, "With less than the required shutdown cooling loops OPERABLE ... ," means that one or both loops are inoperable. Application of ITS Actions dictate that if no loops are OPERABLE, the Actions for both one SDC loop inoperable and no SDC loop OPERABLE must be complied with. Both ITS LCO 3.9.5 and CTS 3.9.8.2 contain additional Actions for no loops OPERABLE or in operation. This change does not impact safety. This change is consistent with NUREG-1432.

A.3 CTS 3.9.8.2 Action b states in part, "With no shutdown cooling loop in operation ... ." ITS LCO 3.9.5 Action Condition B states, "No SDC loop OPERABLE or in operation." The LCO for this Specification requires two shutdown cooling loops to be OPERABLE one of which must be in operation. With no SDC loops OPERABLE, the Actions are the same as for no SDC loop in operation since with both loops inoperable, no credit can be taken for operating loops. Adding this clarification to the Action for no SDC loops OPERABLE does not impact safety and is consistent with NUREG-1432.

ADMINISTRATIVE CHANGES (continued)

A.4 CTS 3.9.8.2 Action b states in part, "With no shutdown cooling loop in operation, suspend all operations involving an increase in the reactor decay heat load or ... ." ITS LCO 3.9.5 does not contain a Required Action for suspending operations involving an increase in reactor decay heat load (loading of irradiated fuel assemblies in the core). ITS LCO 3.9.6 (Refueling Water Level - Fuel Assemblies) requires the water level to be at least 23 feet above the reactor vessel flange during movement of fuel assemblies within the reactor pressure vessel when either the fuel assemblies being moved or the fuel assemblies seated within the reactor pressure vessel are irradiated. With the water level less than 23 feet above the reactor vessel flange, ITS LCO 3.9.6 Required Action A.1 requires suspending movement of fuel assemblies within the reactor pressure vessel immediately. Since ITS LCO 3.9.6 prohibits movement of fuel with less than 23 feet of water above the reactor vessel flange, movement of fuel is also not allowed when in the Applicability of ITS LCO 3.9.5; therefore, an Action requiring suspending loading of irradiated fuel is not necessary. This change does not impact safety and is consistent with NUREG-1432.

A.5 CTS 3.9.8.2 Action b states in part, "With no shutdown cooling loop in operation, suspend all operations involving a ... reduction in boron concentration of the Reactor Coolant System ... ." ITS LCO 3.9.5 Required Action B.1 requires that with no SDC loop Operable or in operation, operations involving a reduction in reactor coolant boron concentration be suspended immediately. Although CTS 3.9.8.2 Action b does not specify a Completion Time of immediately, the intent is that the Required Action be initiated immediately to eliminate the possibility of localized dilution of boron concentration with the mixing action of the SDC unavailable. This change does not impact safety and is consistent with NUREG-1432.

TECHNICAL CHANGES - MORE RESTRICTIVE

M.1 CTS 3.9.8.2 LCO statement is modified by an asterisk which allows removing the SDC loop from operation for up to 1 hour per 8 hour period. ITS LCO 3.9.5 also contains this provision as a Note but is restricted by the statement, "... provided no operations are permitted that would cause reduction of the Reactor Coolant System boron concentration." Suspending operations that could cause a reduction in boron concentration during the 1 hour during which shutdown cooling is allowed out of service is reasonable since there is no other means of forced circulation available to assure uniform boron concentration. The addition of this requirement constitutes a more restrictive change to PVNGS CTS requirements. This change is consistent with NUREG-1432.

M.2 CTS 4.9.8.2 states in part, "At least one shutdown cooling loop shall be verified to be in operation ... ." ITS SR 3.9.5.1 states, "Verify required SDC loops are OPERABLE and one SDC loop is in operation." The ITS Surveillance requires that both loops be verified to be Operable and the required loop in operation while the CTS only requires verification that one loop is in operation. Since both CTS 3.9.8.2 and ITS LCO 3.9.5 require two SDC loops to be Operable, it is necessary to perform a Surveillance to verify that both loops are Operable. This verification is performed administratively. The addition of this requirement constitutes a more restrictive change to PVNGS current operating practices. This change is consistent with NUREG-1432.

M.3 CTS 4.9.8.2 does not contain a requirement to verify power to the SDC pump that is not in operation. ITS SR 3.9.5.2 states, "Verify correct breaker alignment and indicated power available to the required SDC pump that is not in operation." Verification that the required pump which is not in operation is Operable ensures that an additional SDC pump can be placed in operation if needed to maintain decay heat removal and reactor coolant circulation. The addition of this requirement constitutes a more restrictive change to PVNGS current operating practices. This change is consistent with NUREG-1432.

TECHNICAL CHANGES - RELOCATIONS

LA.1 CTS 4.9.8.2 states in part, "At least one shutdown cooling loop shall be verified to be in operation and circulating reactor coolant at a flow rate of greater than or equal to 3780 gpm ... ." ITS SR 3.9.5.1 requires verification that the required shutdown cooling loops are OPERABLE and one loop is in operation. Removal of the quantitative value of the flow rate is acceptable since there are adequate controls of the TS Bases to ensure that changes are appropriately evaluated. The bases for this specification indicates that the pumps are required Operable in order to maintain OPERABILITY of the SDC System. If the pumps are Operable and the system is properly aligned, the resulting flowrate will be sufficient to assure adequate mixing and temperature control of the RCS. These requirements are being relocated to the ITS Bases.

Any change to the requirements in the Bases will be governed by the provisions of the TS Bases Control Program. This provides an equivalent level of control and is an administrative change with no impact on the margin of safety. This requirement does not need to be in the ITS to provide adequate protection to the public health and safety. Therefore, relocation of this requirement is acceptable and is consistent with NUREG-1432.

TECHNICAL CHANGES - RELOCATIONS (continued)

LA.2 CTS 3.9.8.2 LCO statement is modified by an asterisk which states, "... The shutdown cooling loop may be removed from operation for up to 1 hour per 8-hour period during the performance of CORE ALTERATIONS in the vicinity of the reactor pressure vessel hot legs or during surveillance testing of ECCS pumps." ITS LCO 3.9.5 is modified by a Note which also allows the shutdown cooling loop to be removed from operation for up to 1 hour per 8 hour period but does not specify the activities during which this exclusion is allowed. During the 1 hour period when shutdown cooling is not in operation, no reduction in boron concentration is allowed as discussed in DOC M.1. Decay heat removal during this period is provided by natural convection to the large mass of water in the refueling pool. With adequate decay heat removal, and no means of boron reduction, it is acceptable to allow the shutdown cooling loop to be removed from operation for reasons other than those allowed by CTS 3.9.8.2. In this condition, there would not be sufficient decay heat to drastically increase the temperature of the total volume of water over a 1 hour period. These requirements are not required to determine the OPERABILITY of a system,component or structure and therefore are being relocated to the Bases.

Any change to the requirements in the Bases will be governed by the provisions of the Bases Control Program. This provides an equivalent level of regulatory control and is an administrative change with no impact on the margin of safety. This requirement does not need to be in the ITS to provide adequate protection to the public health and safety. Therefore, relocation of this requirement to the Bases is acceptable and is consistent with NUREG-1432.

TECHNICAL CHANGES - LESS RESTRICTIVE

None

# NO SIGNIFICANT HAZARDS CONSIDERATION
# SPECIFICATION 3.9.5

ADMINISTRATIVE CHANGES
(ITS 3.9.5 Discussion of Changes Labeled A.1, A.2, A.3, A.4, and A.5)

Arizona Public Service Company, Palo Verde Nuclear Generating Station (PVNGS), Units 1, 2, and 3, is converting to the ITS as outlined in NUREG-1432, "Standard Technical Specifications, Combustion Engineering Plants." The proposed changes involve the reformatting, renumbering, rewording of the Technical Specifications (TS) and Bases with no change in intent, and the incorporation of current operating practices consistent with NUREG-1432. These changes, since they do not involve technical changes to the Current TS (CTS), are administrative. Below are the No Significant Hazards Consideration (NSHC) for the conversion of this Section/Chapter to NUREG-1432.

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 a proposed amendment, would not 1) involve a significant increase in the probability or consequences of an accident previously evaluated; 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?

> The proposed changes involve reformatting, renumbering, and rewording of the CTS and Bases along with incorporation of PVNGS current operating practices and other changes to the CTS as discussed in the specific Discussion of Changes listed above in order to be consistent with NUREG-1432. The reformatting, renumbering, and rewording along with the other changes listed above, involves no technical changes to the CTS. Specifically, there will be no change in the requirements imposed on PVNGS due to these changes. During development of NUREG-1432, certain wording preferences or English language conventions were adopted. The proposed changes to this Section/Chapter are administrative in nature and do not impact initiators of any analyzed events. They also do not impact the assumed mitigation of accidents or transient events. Therefore, these changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

ADMINISTRATIVE CHANGES
(ITS 3.9.5 Discussion of Changes Labeled (A.1, A.2, A.3, A.4, and A.5) (continued)

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

The proposed changes involve reformatting, renumbering, and rewording of the CTS, along with the incorporation of PVNGS current operating practices and other changes, as discussed, in order to be consistent with NUREG-1432. The proposed changes do not involve a physical alteration of the plant (no new or different type of equipment will be installed) or change the methods governing normal plant operation. The proposed changes will not impose any new or different requirements or eliminate any existing requirements. Therefore, these changes do 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?

The proposed changes involve reformatting, renumbering, and rewording of the CTS, along with the incorporation of PVNGS current operating practices and other changes, as discussed, in order to be consistent with NUREG-1432. The proposed changes are administrative in nature and will not involve any technical changes. The proposed changes will not reduce a margin of safety because they have no impact on any safety analysis assumptions. Also, because these changes are administrative in nature, no question of safety is involved. Therefore, these changes do not involve a significant reduction in a margin of safety.

TECHNICAL CHANGES - MORE RESTRICTIVE
(ITS 3.9.5 Discussion of Changes Labeled M.1, M.2 and M.3)

Arizona Public Service Company, Palo Verde Nuclear Generating Station (PVNGS), Units 1, 2, and 3 is converting to the ITS as outlined in NUREG-1432. This particular NSHC is for the changes labeled "Technical Changes - More Restrictive" described in the specific Discussion of Changes listed above. The proposed changes incorporate more restrictive changes into the CTS by either making current requirements more stringent or adding new requirements which currently do not exist.

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 a proposed amendment, would not 1) involve a significant increase in the probability or consequences of an accident previously evaluated; 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?

The proposed changes provide more stringent requirements than previously existed in the CTS. The more stringent requirements will not result in operation that will increase the probability of initiating an analyzed event. If anything, the new requirements may decrease the probability or consequences of an analyzed event by incorporating the more restrictive changes discussed in the specific Discussion of Changes listed above. These changes will not alter assumptions relative to mitigation of an accident or transient event. The more restrictive requirements will not alter the operation and will continue to ensure process variables, structures, systems, or components are maintained consistent with safety analyses and licensing basis. These changes have been reviewed to ensure that no previously evaluated accident has been adversely affected. Therefore, these changes will not involve a significant increase in the probability or consequences of an accident evaluated.

TECHNICAL CHANGES - MORE RESTRICTIVE
(ITS 3.9.5 Discussion of Changes Labeled M.1, M.2 and M.3) (continued)

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

Making existing requirements more restrictive and adding more restrictive requirements to the CTS will not alter the plant configuration (no new or different type of equipment will be installed) or change the methods governing normal plant operation. These changes do impose different requirements. However, they are consistent with the assumptions made in the safety analyses, licensing basis, and NUREG-1432. Therefore, these changes will 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?

The proposed changes provide more stringent requirements than previously existed in the CTS. An evaluation of these changes concluded that adding these more restrictive requirements either increases or has no impact on the margin of safety. The changes provide additional restrictions which may enhance plant safety. These changes maintain requirements of the safety analysis, licensing basis, and NUREG-1432. As such, no question of safety is involved. Therefore, these changes will not involve a significant reduction in a margin of safety.

TECHNICAL CHANGES - RELOCATIONS
(ITS 3.9.5 Discussion of Changes Labeled LA.1 and LA.2)

Arizona Public Service Company, Palo Verde Nuclear Generating Station (PVNGS), Units 1, 2, and 3 is converting to the ITS as outlined in NUREG-1432. The proposed changes, since detail is being removed from the CTS to a Licensee Controlled Document, are less restrictive. The descriptions of these changes are in the Discussion of Changes listed above.

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 a proposed amendment, would not 1) involve a significant increase in the probability or consequences of an accident previously evaluated; 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:

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

> The proposed changes relocate requirements from the CTS to a Licensee Controlled Document. These changes do not result in any hardware changes or changes to plant operating practices. The details being relocated are not assumed to be an initiator of any analyzed event. The Licensee Controlled Document containing the relocated requirements will be maintained using the provisions of 10 CFR 50.59 or other specified control processes and is subject to the change control process in the Administrative Controls Section of the ITS. Since any changes to a Licensee Controlled Document will be evaluated, no increase in the probability or consequences of an accident previously evaluated will be allowed. Therefore, these changes will not involve a significant increase in the probability or consequences of an accident previously evaluated.

TECHNICAL CHANGES - RELOCATIONS
(ITS 3.9.5 Discussion of Changes Labeled LA.1 and LA.2) (continued)

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

The proposed changes relocate requirements from the CTS to a Licensee Controlled Document. These changes will not alter the plant configuration (no new or different type of equipment will be installed) or change the methods governing normal plant operation. These changes will not impose different requirements and adequate control of information will still be maintained. These changes will not alter assumptions made in the safety analysis or licensing basis. Therefore, these changes will 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?

The proposed changes relocate requirements from the CTS to a Licensee Controlled Document. These changes will not reduce a margin of safety since they have no impact on any safety analysis assumptions. In addition, the requirements to be transposed from the CTS to the Licensee Controlled Document are the same as the CTS. Since any future changes to this Licensee Controlled Document will be evaluated per the requirements of 10 CFR 50.59, or other specified control processes, no reduction (significant or insignificant) in a margin of safety will be allowed. Therefore, these changes will not involve a significant reduction in a margin of safety.

The NRC review provides a certain margin of safety, and although this review will no longer be performed prior to submittal, the NRC still inspects the 10 CFR 50.59 process. The proposed changes are consistent with NUREG-1432, which was approved by the NRC Staff. The change controls for proposed relocated details and requirements provide an acceptable level of regulatory authority. Revising the CTS to reflect the approved level of detail per NUREG-1432 reinforces the conclusion that there is not a significant reduction in the margin of safety. Therefore, revising the CTS to reflect the NRC accepted level of detail and requirements ensures no reduction in a margin of safety.

CE STS
NUREG-1432 REV. 1
SPECIFICATION 3.9.6
MARK UP

<DOC>
<CTS>

## 3.9 REFUELING OPERATIONS

### 3.9.6 Refueling Water Level - Fuel Assemblies 1

<3.9.10.1> LCO 3.9.6 Refueling water level shall be maintained ≥ 23 ft above the top of reactor vessel flange.

APPLICABILITY: ~~During CORE ALTERATIONS, except during latching and unlatching of control rod drive shafts,~~ 1
During movement of ~~irradiated~~ fuel assemblies within containment, when either the fuel assemblies being moved or the fuel assemblies seated within the reactor vessel are irradiated.

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| <3.9.10.1 ACT> A. Refueling water level not within limit. | ~~A.1 Suspend CORE ALTERATIONS.~~ | ~~Immediately~~ 1 |
| | ~~AND~~ | |
| | ~~A.2~~ A.1 Suspend movement of ~~irradiated~~ fuel assemblies within containment. | Immediately 1 |
| | ~~AND~~ | |
| | ~~A.3 Initiate action to restore refueling cavity water level to within limit.~~ | ~~Immediately~~ 2 [TSTF-20] |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | FREQUENCY |
|---|---|
| <4.9.10.1> SR 3.9.6.1 Verify refueling water level is ≥ 23 ft above the top of reactor vessel flange. | 24 hours |

CE STS
NUREG-1432 REV. 1
SPECIFICATION 3.9.6
BASES MARK UP

# B 3.9 REFUELING OPERATIONS

## B 3.9.6 Refueling Water Level - Fuel Assemblies (4)

BASES

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BACKGROUND

(4) The movement of irradiated fuel assemblies ~~or performance of CORE ALTERATIONS, except during latching and unlatching of control rod drive shafts,~~ within containment requires a minimum water level of 23 ft above the top of the reactor vessel flange. During refueling this maintains sufficient water level in ~~the containment~~ the refueling canal, the fuel transfer canal, the refueling cavity, and the spent fuel pool. (4) Sufficient water is necessary to retain iodine fission product activity in the water in the event of a fuel handling accident (Refs. 1 and 2). Sufficient iodine activity would be retained to limit offsite doses from the accident to < ~~25%~~ 33% (4) of 10 CFR 100 limits, ~~as provided by~~ which meets the intent of (4) the guidance of Reference 3.

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APPLICABLE SAFETY ANALYSES

During ~~core alterations and during~~ movement of irradiated fuel assemblies, the water level in the refueling canal and refueling cavity is an initial condition design parameter in the analysis of the fuel handling accident in containment postulated by Regulatory Guide 1.25 (Ref. 1). A minimum water level of 23 ft (Regulatory Position C.1.c of Ref. 1) allows a decontamination factor of 100 (Regulatory Position C.1.g of Ref. 1) to be used in the accident analysis for iodine. This relates to the assumption that 99% of the total iodine released from the pellet to cladding gap of all the dropped fuel assembly rods is retained by the refueling ~~cavity~~ Canal (4) water. The fuel pellet to cladding gap is assumed to contain 10% of the total fuel rod iodine inventory (Ref. 1).

The fuel handling accident analysis inside containment is described in Reference 2. With a minimum water level of 23 ft and a minimum decay time of ~~72~~ 100 (4) hours prior to fuel handling, the analysis and test programs demonstrate that the iodine release due to a postulated fuel handling accident is adequately captured by the water and offsite doses are maintained within allowable limits (Ref. 4).

Refueling water level satisfies Criterion 2 of ~~the NRC Policy Statement~~ 10 CFR 50.36 (c)(2)(ii) (3).

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(continued)

BASES (continued)

LCO

A minimum refueling water level of 23 ft above the reactor vessel flange is required to ensure that the radiological consequences of a postulated fuel handling accident inside containment are within acceptable limits as provided by the guidance of Reference 3.

APPLICABILITY

LCO 3.9.6 is applicable ~~during CORE ALTERATIONS, except during latching and unlatching of control rod drive shafts, and when moving fuel assemblies in the presence of irradiated fuel assemblies.~~ (4) when moving fuel assemblies within containment when either the fuel assemblies being moved or the fuel assemblies seated in the reactor vessel are irradiated. The LCO minimizes the possibility of a fuel handling accident in containment that is beyond the assumptions of the safety analysis. If irradiated fuel is not present in containment, there can be no significant radioactivity release as a result of a postulated fuel handling accident. Requirements for fuel handling accidents in the spent fuel pool are covered by LCO 3.7.~~10~~ 14, "Fuel Storage Pool Water Level."

ACTIONS

A.1 ~~and A.2~~

With a water level of < 23 ft above the top of the reactor vessel flange, all operations involving ~~CORE ALTERATIONS or~~ movement of irradiated fuel assemblies shall be suspended immediately to ensure that a fuel handling accident cannot occur. (4)

The suspension of ~~CORE ALTERATIONS and~~ fuel movement shall not preclude completion of movement of a component to a safe position.

~~A.3~~

~~In addition to immediately suspending CORE ALTERATIONS or movement of irradiated fuel, action to restore refueling cavity water level must be initiated immediately.~~ (2)

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.9.6.1

Verification of a minimum water level of 23 ft above the top of the reactor vessel flange ensures that the design basis for the postulated fuel handling accident analysis during refueling operations is met. Water at the required level above the top of the reactor vessel flange limits the consequences of damaged fuel rods that are postulated to result from a fuel handling accident inside containment (Ref. 2).

The Frequency of 24 hours is based on engineering judgment and is considered adequate in view of the large volume of water and the normal procedural controls of valve positions, and HVAC operations which make significant unplanned level changes unlikely. 4

REFERENCES

1. Regulatory Guide 1.25, March 23, 1972.

2. FSAR, Section [ ]. 15.7.4 3

3. NUREG-0800, Section 15.7.4.

4. 10 CFR 100.10.

NUREG-1432 EXCEPTIONS
SPECIFICATION 3.9.6

1. NUREG-1432 LCO 3.9.6 is Applicable during Core Alterations (except during latching and unlatching of CEAs), and during movement of irradiated fuel assemblies within containment. ITS LCO 3.9.6 is Applicable during movement of fuel assemblies within containment when either the fuel assemblies being moved or the fuel assemblies seated in the reactor vessel are irradiated. The design of the PVNGS CEA (upper guide-structure) lift rig is such that the platform that refueling personnel stand on while latching and unlatching CEAs would be submerged with the refueling water level 23 ft. above the top of the reactor vessel flange. The current operating practice is to maintain refueling water level 23 ft. above the top of irradiated fuel assemblies seated in the reactor vessel while latching and unlatching CEAs. The water level is then raised as the CEA lift rig and CEAs are withdrawn. By not submerging the CEA lift rig top plate, personnel contaminations due to hot particles are reduced.

   The ITS has been modified by addition of Specification ITS 3.9.7 (Refueling Water Level - CEAs) which requires a refueling water level of 23 ft. above the top of irradiated fuel seated within the reactor vessel during movement of CEAs within the reactor vessel. Performance of these activities with 23 ft. of water above the top of the irradiated fuel does not violate the initial conditions of the fuel handling accident analyses. This change does not impact safety. The PVNGS current licensing bases allows movement of CEAs within the reactor vessel with 23 ft. of water above the top of irradiated fuel seated in the reactor vessel.

2. NUREG-1432 LCO 3.9.6 Required Action A.3 requires initiating action to restore refueling cavity water level to within limit immediately. This requirement has been deleted from ITS LCO 3.9.6 since performance of Required Action A.2 results in exiting the Condition specified in the Applicability. This change does not impact safety and is consistent with the Actions specified in the current licensing bases. This change has been submitted as Generic Change TSTF - 20.

3. Grammar and/or editorial changes have been made to enhance clarity. No technical or intent changes to the Specification are made by this change.

4. The plant specific titles, nomenclature, number parameter/value, reference, system description, system design, operating practices or analysis description was used (additions, deletions, and/or changes are included). Plant specific parameters/values were directly transferred from the CTS to the ITS, or from the plant design basis to the ITS. The Bases have been revised to be consistent with the LCO/Surveillance.

PVNGS CTS
SPECIFICATION 3.9.6
MARK UP

Specification 3.9.6

A.1

3.9 REFUELING OPERATIONS Refueling

3.9.6 3/4.9.10 WATER LEVEL - REACTOR VESSEL Fuel Assemblies

FUEL ASSEMBLIES

LIMITING CONDITION FOR OPERATION

LCO 3.9.6 3.9.10.1 At least 23 feet of water shall be maintained over the top of the reactor pressure vessel flange.

Containment. M.1

APPLICABILITY: During movement of fuel assemblies within the reactor pressure vessel when either the fuel assemblies being moved or the fuel assemblies seated within the reactor pressure vessel are irradiated.

ACTION: Refueling water level not within limit, immediately

ACT A ~~With the requirements of the above specification not satisfied,~~ suspend ~~all operations involving~~ movement of fuel assemblies within ~~the pressure vessel~~.

A2 containment M.1

SURVEILLANCE REQUIREMENTS

L.1

SR 3.9.6.1 4.9.10.1 ~~The water level shall be determined to be at least its minimum required depth within 2 hours prior to the start of and at least once per 24 hours thereafter during movement of fuel assemblies.~~

Verify refueling water level is ≥ 23ft above the top of reactor vessel flange once per 24 hours.

DISCUSSION OF CHANGES
SPECIFICATION 3.9.6

ADMINISTRATIVE CHANGES

A.1 All reformatting and renumbering is in accordance with the Combustion Engineering Plant (CEOG) Standard Technical Specifications NUREG-1432, Rev. 1 (NUREG-1432). As a result, the Palo Verde Nuclear Generating Station (PVNGS) Improved Technical Specifications (ITS) should be more readable, and therefore understandable, by plant operators as well as other users. During the reformatting and renumbering of the ITS, no technical changes (either actual or interpretational) to the Current Technical Specification (CTS) were made unless they were identified and justified.

Editorial rewording (either adding or deleting) is made consistent with NUREG-1432. During NUREG-1432 development, certain wording preferences or English language conventions were adopted which resulted in no technical changes (either actual or interpretational) to the CTS.

Additional information has also been added to more fully describe each subsection. This wording is consistent with NUREG-1432. Since the design is already approved by the NRC, adding more detail does not result in a technical change.

A.2 CTS 3.9.10 Action states in part, "... suspend all operations involving movement of fuel assemblies ... ." ITS 3.9.6 Required Action A.1 states, "Suspend movement of fuel assemblies within containment." Rewording the Required Action provides clarification of the intent of the Action and is consistent with the Action necessary to exit the Applicability. Adding clarification does not alter the intent or application of the Required Action and does not impact safety. This change is consistent with NUREG-1432.

TECHNICAL CHANGES - MORE RESTRICTIVE

M.1 CTS 3/4.9.10 Applicability states, "During movement of fuel assemblies within the reactor pressure vessel when either the fuel assemblies being moved or the fuel assemblies seated within the reactor pressure vessel are irradiated." ITS 3.9.6 Applicability states, "During movement of fuel assemblies within containment when either the fuel assemblies being moved or the fuel assemblies seated within the reactor vessel are irradiated." CTS 3/4.9.10 is Applicable while moving fuel within the reactor vessel while ITS 3.9.6 is Applicable while moving fuel within containment. The Actions are reworded to suspend fuel movement within containment accordingly. The Applicability of the CTS is bounded by the Applicability of the ITS. This change in Applicability is acceptable since a fuel handling accident is possible while handling irradiated fuel outside the reactor vessel. The addition of this requirement constitutes a more restrictive change to PVNGS current operating practices. This change is consistent with NUREG-1432.

TECHNICAL CHANGES - RELOCATIONS

None

TECHNICAL CHANGES - LESS RESTRICTIVE

L.1 CTS 4.9.10.1 states, "The water level shall be determined to be at least its minimum required depth within 2 hours prior to the start of and at least once per 24 hours thereafter during movement of irradiated fuel assemblies." The requirement to perform this Surveillance within 2 hours prior to the start of fuel movement is not specified in ITS SR 3.9.6.1. The Surveillance must be performed within 24 hours prior to the start of fuel movement as specified by the ITS SR 3.9.6.1 Frequency. The basis for this time interval is contained in ITS 1.4 Example 1.4-1. This example states in part, "...The measurement of this interval continues at all times, even when the SR is not required to be met per SR 3.0.1 (such as when the equipment is inoperable, a variable is outside specified limits, or the unit is outside the Applicability of the LCO). ..." This example further states, "... If the interval as specified by SR 3.0.2 is exceeded while the unit is not in a MODE or other specified condition in the Applicability of the LCO for which performance of the SR is required, the Surveillance must be performed within the Frequency requirements of SR 3.0.2 prior to entry into the MODE or other specified condition. ..." Allowing the Surveillance to be performed within its specified Frequency prior to entering the Applicability is acceptable since it is consistent with the performance interval required when the unit is in the Applicability. This interval is also consistent with the application of ITS SR 3.0.1 for other Specifications. This change does not impact safety and is consistent with NUREG-1432.

# NO SIGNIFICANT HAZARDS CONSIDERATION
# SPECIFICATION 3.9.6

ADMINISTRATIVE CHANGES
(ITS 3.9.6 Discussion of Changes Labeled A.1 and A.2)

Arizona Public Service Company, Palo Verde Nuclear Generating Station (PVNGS), Units 1, 2, and 3, is converting to the ITS as outlined in NUREG-1432, "Standard Technical Specifications, Combustion Engineering Plants." The proposed changes involve the reformatting, renumbering, rewording of the Technical Specifications (TS) and Bases with no change in intent, and the incorporation of current operating practices consistent with NUREG-1432. These changes, since they do not involve technical changes to the Current TS (CTS), are administrative. Below are the No Significant Hazards Consideration (NSHC) for the conversion of this Section/Chapter to NUREG-1432.

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 a proposed amendment, would not 1) involve a significant increase in the probability or consequences of an accident previously evaluated; 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?

> The proposed changes involve reformatting, renumbering, and rewording of the CTS and Bases along with incorporation of PVNGS current operating practices and other changes to the CTS as discussed in the specific Discussion of Changes listed above in order to be consistent with NUREG-1432. The reformatting, renumbering, and rewording along with the other changes listed above, involves no technical changes to the CTS. Specifically, there will be no change in the requirements imposed on PVNGS due to these changes. During development of NUREG-1432, certain wording preferences or English language conventions were adopted. The proposed changes to this Section/Chapter are administrative in nature and do not impact initiators of any analyzed events. They also do not impact the assumed mitigation of accidents or transient events. Therefore, these changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

ADMINISTRATIVE CHANGES
(ITS 3.9.6 Discussion of Changes Labeled (A.1 and A.2) (continued)

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

The proposed changes involve reformatting, renumbering, and rewording of the CTS, along with the incorporation of PVNGS current operating practices and other changes, as discussed, in order to be consistent with NUREG-1432. The proposed changes do not involve a physical alteration of the plant (no new or different type of equipment will be installed) or change the methods governing normal plant operation. The proposed changes will not impose any new or different requirements or eliminate any existing requirements. Therefore, these changes do 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?

The proposed changes involve reformatting, renumbering, and rewording of the CTS, along with the incorporation of PVNGS current operating practices and other changes, as discussed, in order to be consistent with NUREG-1432. The proposed changes are administrative in nature and will not involve any technical changes. The proposed changes will not reduce a margin of safety because they have no impact on any safety analysis assumptions. Also, because these changes are administrative in nature, no question of safety is involved. Therefore, these changes do not involve a significant reduction in a margin of safety.

TECHNICAL CHANGES - MORE RESTRICTIVE
(ITS 3.9.6 Discussion of Changes Labeled M.1)

Arizona Public Service Company, Palo Verde Nuclear Generating Station (PVNGS), Units 1, 2, and 3 is converting to the ITS as outlined in NUREG-1432. This particular NSHC is for the changes labeled "Technical Changes - More Restrictive" described in the specific Discussion of Changes listed above. The proposed changes incorporate more restrictive changes into the CTS by either making current requirements more stringent or adding new requirements which currently do not exist.

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 a proposed amendment, would not 1) involve a significant increase in the probability or consequences of an accident previously evaluated; 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?

> The proposed changes provide more stringent requirements than previously existed in the CTS. The more stringent requirements will not result in operation that will increase the probability of initiating an analyzed event. If anything, the new requirements may decrease the probability or consequences of an analyzed event by incorporating the more restrictive changes discussed in the specific Discussion of Changes listed above. These changes will not alter assumptions relative to mitigation of an accident or transient event. The more restrictive requirements will not alter the operation and will continue to ensure process variables, structures, systems, or components are maintained consistent with safety analyses and licensing basis. These changes have been reviewed to ensure that no previously evaluated accident has been adversely affected. Therefore, these changes will not involve a significant increase in the probability or consequences of an accident evaluated.

TECHNICAL CHANGES - MORE RESTRICTIVE
(ITS 3.9.6 Discussion of Changes Labeled M.1) (continued)

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

Making existing requirements more restrictive and adding more restrictive requirements to the CTS will not alter the plant configuration (no new or different type of equipment will be installed) or change the methods governing normal plant operation. These changes do impose different requirements. However, they are consistent with the assumptions made in the safety analyses, licensing basis, and NUREG-1432. Therefore, these changes will 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?

The proposed changes provide more stringent requirements than previously existed in the CTS. An evaluation of these changes concluded that adding these more restrictive requirements either increases or has no impact on the margin of safety. The changes provide additional restrictions which may enhance plant safety. These changes maintain requirements of the safety analysis, licensing basis, and NUREG-1432. As such, no question of safety is involved. Therefore, these changes will not involve a significant reduction in a margin of safety.

TECHNICAL CHANGES - LESS RESTRICTIVE
(ITS 3.9.6 Discussion of Changes Labeled L.1)

Arizona Public Service Company, Palo Verde Nuclear Generating Station (PVNGS), Units 1, 2, and 3 is converting to the ITS as outlined in NUREG-1432. The proposed change involves making the CTS less restrictive. Below is the description of this less restrictive change and the NSHC for the conversion to NUREG 1432.

L.1 CTS 4.9.10.1 states, "The water level shall be determined to be at least its minimum required depth within 2 hours prior to the start of and at least once per 24 hours thereafter during movement of irradiated fuel assemblies." The requirement to perform this Surveillance within 2 hours prior to the start of fuel movement is not specified in ITS SR 3.9.6.1. The Surveillance must be performed within 24 hours prior to the start of fuel movement as specified by the ITS SR 3.9.6.1 Frequency. The basis for this time interval is contained in ITS 1.4 Example 1.4-1. This example states in part, "... The measurement of this interval continues at all times, even when the SR is not required to be met per SR 3.0.1 (such as when the equipment is inoperable, a variable is outside specified limits, or the unit is outside the Applicability of the LCO)..." This example further states, "... If the interval as specified by SR 3.0.2 is exceeded while the unit is not in a MODE or other specified condition in the Applicability of the LCO for which performance of the SR is required, the Surveillance must be performed within the Frequency requirements of SR 3.0.2 prior to entry into the MODE or other specified condition..." Allowing the Surveillance to be performed within its specified Frequency prior to entering the Applicability is acceptable since it is consistent with the performance interval required when the unit is in the Applicability. This interval is also consistent with the application of ITS SR 3.0.1 for other Specifications. This change does not impact safety and is consistent with NUREG-1432.

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 a proposed amendment, would not 1) involve a significant increase in the probability or consequences of an accident previously evaluated; 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:

TECHNICAL CHANGES - LESS RESTRICTIVE
(ITS 3.9.6 Discussion of Changes Labeled L.1) (continued)

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

The proposed change modifies the Frequency of the refueling water level SR by removing the requirement to perform within 2 hours prior to the start of moving irradiated fuel assemblies in the reactor pressure vessel. The Frequency for this SR is once per 24 hours which requires that the SR be performed within 24 hours prior to entering the Applicability. Removal of the requirement to perform this SR within 2 hours prior to entering the Applicability is consistent with the requirements for entry into the Applicability of other specifications. These requirements are delineated in ITS LCO 3.0 and ITS SR 3.0.

This change is consistent with NUREG-1432. This change does not result in any hardware changes or changes to plant operating practices nor does it affect plant operation. Therefore this change will not involve a significant increase in the probability or 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?

The proposed change modifies the Frequency of the refueling water level SR by removing the requirement to perform within 2 hours prior to the start of moving irradiated fuel assemblies in the reactor pressure vessel. The Frequency for this SR is once per 24 hours which requires that the SR be performed within 24 hours prior to entering the Applicability. Removal of the requirement to perform this SR within 2 hours prior to entering the Applicability is consistent with the requirements for entry into the Applicability of other specifications. These requirements are delineated in ITS LCO 3.0 and ITS SR 3.0.

This change is consistent with NUREG-1432. This change will not alter the plant configuration (no new or different type of equipment will be installed) or change the methods of governing normal plant operation. This change will not alter assumptions made in the safety analysis or licensing basis. Therefore, this change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

TECHNICAL CHANGES - LESS RESTRICTIVE
(ITS 3.9.6 Discussion of Changes Labeled L.1) (continued)

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

The proposed change modifies the Frequency of the refueling water level SR by removing the requirement to perform within 2 hours prior to the start of moving irradiated fuel assemblies in the reactor pressure vessel. The Frequency for this SR is once per 24 hours which requires that the SR be performed within 24 hours prior to entering the Applicability. Removal of the requirement to perform this SR within 2 hours prior to entering the Applicability is consistent with the requirements for entry into the Applicability of other specifications. These requirements are delineated in ITS LCO 3.0 and ITS SR 3.0.

This change will not reduce a margin of safety since it has no impact on safety analysis assumptions. This change is consistent with NUREG-1432, which was approved by the NRC Staff. Therefore, this change does not result in a reduction in a margin of safety.

CE STS

NUREG-1432 REV. 1

SPECIFICATION 3.9.7

MARK UP

<DOC>
<CTS>

## 3.9 REFUELING OPERATIONS

### 3.9.7 ~~3.9.6~~ Refueling Water Level - CEAS

<3.9.10.1>
<3.9.10.2>

LCO ~~3.9.6~~ 3.9.7 Refueling water level shall be maintained ≥ 23 ft above the top of irradiated fuel assemblies seated within the reactor vessel, ~~flange~~.

APPLICABILITY: ~~During CORE ALTERATIONS, except during latching and unlatching of control rod drive shafts,~~
~~During movement of irradiated fuel assemblies within containment.~~
During movement of CEA's within the reactor vessel when the fuel assemblies seated in the reactor vessel are irradiated.

ACTIONS

<3.9.10.1ACT>
<3.9.10.2ACT>

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. Refueling water level not within limit. | ~~A.1 Suspend CORE ALTERATIONS.~~ | ~~Immediately~~ |
| | ~~AND~~ | |
| | ~~A.2~~ A.1 Suspend movement of ~~irradiated fuel assemblies within containment.~~ CEAS within the reactor vessel | Immediately |
| | ~~AND~~ | |
| | ~~A.3 Initiate action to restore refueling cavity water level to within limit.~~ | ~~Immediately~~ |

SURVEILLANCE REQUIREMENTS

<4.9.10.1>
<4.9.10.2>

| SURVEILLANCE | FREQUENCY |
|---|---|
| SR ~~3.9.6.1~~ 3.9.7.1 Verify refueling water level is ≥ 23 ft above the top of irradiated fuel assemblies seated within the reactor vessel. ~~flange~~ | 24 hours |

CE STS
NUREG-1432 REV. 1
SPECIFICATION 3.9.7
BASES MARK UP

## B 3.9 REFUELING OPERATIONS

## B 3.9.~~6~~ 7 Refueling Water Level - CEAs 2

BASES

---

**BACKGROUND** 3 The movement of ~~irradiated fuel assemblies or performance of CORE ALTERATIONS, except during latching and unlatching of control rod drive shafts,~~ CEA'S within ~~containment requires a~~ the reactor vessel, when irradiated fuel assemblies are seated in the reactor vessel 1 minimum water level of 23 ft above the top of the ~~reactor vessel flange~~ irradiated fuel 1. During refueling this maintains sufficient water level in ~~the containment~~ 3, the refueling canal, the fuel transfer canal, the refueling cavity, and the spent fuel pool. Sufficient water is necessary to retain iodine fission product activity in the water in the event of a fuel handling accident (Refs. 1 and 2). Sufficient iodine activity would be retained to limit offsite doses from the accident to < ~~25%~~ 33% 3 of 10 CFR 100 limits, ~~as provided by~~ which meets the intent of 1 the guidance of Reference 3.

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**APPLICABLE SAFETY ANALYSES** 1 ~~During core alterations and during movement of irradiated fuel assemblies~~ movement of CEAs, the water level in the refueling canal and refueling cavity is an initial condition design parameter in the analysis of the fuel handling accident in containment postulated by Regulatory Guide 1.25 (Ref. 1). A minimum water level of 23 ft (Regulatory Position C.1.c of Ref. 1) allows a decontamination factor of 100 (Regulatory Position C.1.g of Ref. 1) to be used in the accident analysis for iodine. This relates to the assumption that 99% of the total iodine released from the pellet to cladding gap of all the dropped fuel assembly rods is retained by the refueling cavity water. The fuel pellet to cladding gap is assumed to contain 10% of the total fuel rod iodine inventory (Ref. 1).

The fuel handling accident analysis inside containment is described in Reference 2. With a minimum water level of 23 ft and a minimum decay time of ~~72~~ 100 3 hours prior to fuel handling, the analysis and test programs demonstrate that the iodine release due to a postulated fuel handling accident is adequately captured by the water and offsite doses are maintained within allowable limits (Ref. 4).

Refueling water level satisfies Criterion 2 of ~~the NRC Policy Statement~~ 10 CFR 50.36 (c)(2)(ii) 2.

---

(continued)

- CEAs



BASES (continued)

1

irradiated fuel assemblies seated within

LCO

A minimum refueling water level of 23 ft above the reactor vessel flange is required to ensure that the radiological consequences of a postulated fuel handling accident inside containment are within acceptable limits as provided by the guidance of Reference 3.

7 movement of CEAs 3

APPLICABILITY

LCO 3.9.6 is applicable during CORE ALTERATIONS, except during latching and unlatching of control rod drive shafts, and when moving fuel assemblies in the presence of irradiated fuel assemblies. The LCO minimizes the possibility of a fuel handling accident in containment that is beyond the assumptions of the safety analysis. If irradiated fuel is not present in containment, there can be no significant radioactivity release as a result of a postulated fuel handling accident. Requirements for fuel handling accidents in the spent fuel pool are covered by LCO 3.7.10, "Fuel Storage Pool Water Level." 14

within the reactor vessel when irradiated fuel assemblies are seated within the reactor vessel 3

ACTIONS

A.1 and A.2

1 irradiated fuel assemblies seated within

With a water level of < 23 ft above the top of the reactor vessel flange, all operations involving CORE ALTERATIONS or movement of irradiated fuel assemblies shall be suspended immediately to ensure that a fuel handling accident cannot occur.

movement of CEAs 1 within the reactor vessel

1

movement of CEAs 1

The suspension of CORE ALTERATIONS and fuel movement shall not preclude completion of movement of a component to a safe position.

1

~~A.3~~

~~In addition to immediately suspending CORE ALTERATIONS or movement of irradiated fuel, action to restore refueling cavity water level must be initiated immediately.~~

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.9.6.1 7

Verification of a minimum water level of 23 ft above the top of the reactor vessel flange irradiated fuel assemblies seated within 3 ensures that the design basis for the postulated fuel handling accident analysis during refueling operations is met. Water at the required level above the top of the ~~reactor vessel flange~~ irradiated fuel 3 limits the consequences of damaged fuel rods that are postulated to result from a fuel handling accident inside containment (Ref. 2).

The Frequency of 24 hours is based on engineering judgment and is considered adequate in view of the large volume of water and the normal procedural controls of valve positions, and HVAC operations 3 which make significant unplanned level changes unlikely.

REFERENCES

1. Regulatory Guide 1.25, March 23, 1972.
2. FSAR, Section [ ] 15.7.4 2
3. NUREG-0800, Section 15.7.4.
4. 10 CFR 100.10.

NUREG-1432 EXCEPTIONS
SPECIFICATION 3.9.7

1. NUREG-1432 LCO 3.9.6 is Applicable during Core Alterations (except during latching and unlatching of CEAs), and during movement of irradiated fuel assemblies within containment. ITS LCO 3.9.7 is Applicable only during movement of new fuel assemblies and handling of CEAs within the reactor vessel when irradiated fuel is seated within the reactor vessel. The design of the PVNGS CEA lift rig (upper guide structure) is such that the platform that refueling personnel stand on while latching and unlatching CEAs would be submerged with the refueling water level 23 ft. above the top of the reactor vessel flange as required by NUREG-1432 LCO 3.9.6. The current operating practice is to maintain refueling water level 23 ft. above the top of irradiated fuel assemblies seated in the reactor vessel while latching and unlatching CEAs. The water level is then raised as the CEA lift rig and CEAs are withdrawn. By not submerging the CEA lift rig top plate, personnel contaminations due to hot particles are reduced. During this operation, water level is maintained 23 ft above the top of irradiated fuel assemblies seated in the reactor vessel as required by CTS 3.9.10.2.

   The ITS has been modified by addition of Specification ITS 3.9.7 (Refueling Water Level - CEAs) which requires a refueling water level of 23 ft. above the top of irradiated fuel seated within the reactor vessel during movement or handling of CEAs within the reactor vessel. Performance of these activities with 23 ft. of water above the top of the irradiated fuel does not violate the initial conditions of the fuel handling accident analyses. This change does not impact safety. The PVNGS current licensing bases allows movement of CEAs within the reactor vessel with 23 ft. of water above the top of irradiated fuel seated in the reactor vessel.

2. Grammar and/or editorial changes have been made to enhance clarity. No technical or intent changes to the Specification are made by this change.

3. The plant specific titles, nomenclature, number parameter/value, reference, system description, system design, operating practices or analysis description was used (additions, deletions, and/or changes are included). Plant specific parameters/values were directly transferred from the CTS to the ITS, or from the plant design basis to the ITS. The Bases have been revised to be consistent with the LCO/Surveillance.

PVNGS CTS
SPECIFICATION 3.9.7
MARK UP

Specification 3.9.7

A.1

3.9 REFUELING OPERATIONS

3.9.7 ~~CEAS~~ Refueling Water Level - CEAs

~~LIMITING CONDITION FOR OPERATION~~

Irradiated A.2

LCO 3.9.7 ~~3.9.10.2~~ At least 23 feet of water shall be maintained over the top of ~~the~~ fuel seated in the reactor ~~pressure~~ vessel.
assemblies

APPLICABILITY: During movement of CEAs within the reactor pressure vessel, when the fuel assemblies seated within the reactor ~~pressure~~ vessel are irradiated.

ACT ACTION: Refueling water level not within limit, immediately

~~With the requirements of the above specification not satisfied,~~ suspend ~~all operations involving~~ movement of CEAs within the ~~pressure~~ vessel.
reactor

SURVEILLANCE REQUIREMENTS

SR 3.9.7.1 ~~4.9.10.2 The water level shall be determined to be at least its minimum required depth within 2 hours prior to the start of and at least once per 24 hours thereafter during movement of CEAs.~~

L.1

Verify refueling water level is ≥ 23 ft above the top of irradiated fuel assemblies seated within the reactor vessel once per 24 hours.

DISCUSSION OF CHANGES
SPECIFICATION 3.9.7

ADMINISTRATIVE CHANGES

A.1 All reformatting and renumbering is in accordance with the Combustion Engineering Plant (CEOG) Standard Technical Specifications NUREG-1432, Rev. 1 (NUREG-1432). As a result, the Palo Verde Nuclear Generating Station (PVNGS) Improved Technical Specifications (ITS) should be more readable, and therefore understandable, by plant operators as well as other users. During the reformatting and renumbering of the ITS, no technical changes (either actual or interpretational) to the Current Technical Specification (CTS) were made unless they were identified and justified.

Editorial rewording (either adding or deleting) is made consistent with NUREG-1432. During NUREG-1432 development, certain wording preferences or English language conventions were adopted which resulted in no technical changes (either actual or interpretational) to the CTS.

Additional information has also been added to more fully describe each subsection. This wording is consistent with NUREG-1432. Since the design is already approved by the NRC, adding more detail does not result in a technical change.

A.2 CTS 3.9.10.2 states, "At least 23 ft of water shall be maintained over the top of the fuel seated in the reactor pressure vessel." ITS LCO 3.9.7 states, "Refueling water level shall be maintained ≥ 23 ft above the top of irradiated fuel assemblies seated within the reactor vessel." The ITS adds clarification in the LCO statement that the water level is Applicable only if there is irradiated fuel seated in the reactor vessel. With only new fuel seated in the reactor vessel, there would be no release of iodine if a new fuel assembly or CEA was dropped. The Applicability of CTS 3.9.10.2 clearly states that the Specification applies when the fuel assemblies seated within the reactor pressure vessel are irradiated. This change provides clarification of existing requirements and does not alter the intent of the Specification. There is no impact to safety due to this change. This change is consistent with NUREG-1432.

TECHNICAL CHANGES - MORE RESTRICTIVE

None

TECHNICAL CHANGES - RELOCATIONS

None

TECHNICAL CHANGES - LESS RESTRICTIVE

L.1 CTS 4.9.10.2 states in part, "The water level shall be determined to be at least its minimum required depth within 2 hours prior to the start of ... ." The requirement to perform this Surveillance within 2 hours prior to the start of movement of CEAs is not specified in ITS SR 3.9.7.1. The Surveillance must be performed within 24 hours prior to the start of movement as specified by the ITS SR 3.9.7.1 Frequency. The basis for this time interval is contained in ITS 1.4 Example 1.4-1. This example states in part, "...The measurement of this interval continues at all times, even when the SR is not required to be met per SR 3.0.1 (such as when the equipment is inoperable, a variable is outside specified limits, or the unit is outside the Applicability of the LCO). ..." This example further states, "... If the interval as specified by SR 3.0.2 is exceeded while the unit is not in a MODE or other specified condition in the Applicability of the LCO for which performance of the SR is required, the Surveillance must be performed within the Frequency requirements of SR 3.0.2 prior to entry into the MODE or other specified condition. ..." Allowing the Surveillance to be performed within its specified Frequency prior to entering the Applicability is acceptable since it is consistent with the performance interval required when the unit is in the Applicability. This interval is also consistent with the application of ITS SR 3.0.1 for other Specifications. This change does not impact safety and is consistent with NUREG-1432.

# NO SIGNIFICANT HAZARDS CONSIDERATION
# SPECIFICATION 3.9.7

ADMINISTRATIVE CHANGES
(ITS 3.9.7 Discussion of Changes Labeled A.1 and A.2)

Arizona Public Service Company, Palo Verde Nuclear Generating Station (PVNGS), Units 1, 2, and 3, is converting to the ITS as outlined in NUREG-1432, "Standard Technical Specifications, Combustion Engineering Plants." The proposed changes involve the reformatting, renumbering, rewording of the Technical Specifications (TS) and Bases with no change in intent, and the incorporation of current operating practices consistent with NUREG-1432. These changes, since they do not involve technical changes to the Current TS (CTS), are administrative. Below are the No Significant Hazards Consideration (NSHC) for the conversion of this Section/Chapter to NUREG-1432.

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 a proposed amendment, would not 1) involve a significant increase in the probability or consequences of an accident previously evaluated; 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?

> The proposed changes involve reformatting, renumbering, and rewording of the CTS and Bases along with incorporation of PVNGS current operating practices and other changes to the CTS as discussed in the specific Discussion of Changes listed above in order to be consistent with NUREG-1432. The reformatting, renumbering, and rewording along with the other changes listed above, involves no technical changes to the CTS. Specifically, there will be no change in the requirements imposed on PVNGS due to these changes. During development of NUREG-1432, certain wording preferences or English language conventions were adopted. The proposed changes to this Section/Chapter are administrative in nature and do not impact initiators of any analyzed events. They also do not impact the assumed mitigation of accidents or transient events. Therefore, these changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

ADMINISTRATIVE CHANGES
(ITS 3.9.7 Discussion of Changes Labeled (A.1 and A.2) (continued)

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

The proposed changes involve reformatting, renumbering, and rewording of the CTS, along with the incorporation of PVNGS current operating practices and other changes, as discussed, in order to be consistent with NUREG-1432. The proposed changes do not involve a physical alteration of the plant (no new or different type of equipment will be installed) or change the methods governing normal plant operation. The proposed changes will not impose any new or different requirements or eliminate any existing requirements. Therefore, these changes do 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?

The proposed changes involve reformatting, renumbering, and rewording of the CTS, along with the incorporation of PVNGS current operating practices and other changes, as discussed, in order to be consistent with NUREG-1432. The proposed changes are administrative in nature and will not involve any technical changes. The proposed changes will not reduce a margin of safety because they have no impact on any safety analysis assumptions. Also, because these changes are administrative in nature, no question of safety is involved. Therefore, these changes do not involve a significant reduction in a margin of safety.

TECHNICAL CHANGES - LESS RESTRICTIVE
(ITS 3.9.7 Discussion of Changes Labeled L.1)

Arizona Public Service Company, Palo Verde Nuclear Generating Station (PVNGS), Units 1, 2, and 3 is converting to the ITS as outlined in NUREG-1432. The proposed change involves making the CTS less restrictive. Below is the description of this less restrictive change and the NSHC for conversion to NUREG-1432.

L.1 CTS 4.9.10.2 states in part, "The water level shall be determined to be at least its minimum required depth within 2 hours prior to the start of..." The requirement to perform this Surveillance within 2 hours prior to the start of movement of CEAs is not specified in ITS SR 3.9.7.1. The Surveillance must be performed within 24 hours prior to the start of movement as specified by the ITS SR 3.9.7.1 Frequency. The basis for this time interval is contained in ITS 1.4 Example 1.4-1. This example states in part, "...The measurement of this interval continues at all times, even when the SR is not required to be met per SR 3.0.1 (such as when the equipment is inoperable, a variable is outside specified limits, or the unit is outside the Applicability of the LCO)..." This example further states, "...If the interval as specified by SR 3.0.2 is exceeded while the unit is not in a MODE or other specified condition in the Applicability of the LCO for which performance of the SR is required, the Surveillance must be performed within the Frequency requirements of SR 3.0.2 prior to entry into the MODE or other specified condition..." Allowing the Surveillance to be performed within its specified Frequency prior to entering the Applicability is acceptable since it is consistent with the performance interval required when the unit is in the Applicability. This interval is also consistent with the application of ITS SR 3.0.1 for other Specifications. This change does not impact safety and is consistent with NUREG-1432.

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 a proposed amendment, would not 1) involve a significant increase in the probability or consequences of an accident previously evaluated; 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:

TECHNICAL CHANGES - LESS RESTRICTIVE
(ITS 3.9.7 Discussion of Changes Labeled L.1) (continued)

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

The proposed change modifies the Frequency of the refueling water level SR by removing the requirement to perform within 2 hours prior to the start of moving fuel assemblies or CEAs in the reactor pressure vessel. The Frequency for this SR is once per 24 hours which requires that the SR be performed within 24 hours prior to entering the Applicability. Removal of the requirement to perform this SR within 2 hours prior to entering the Applicability is consistent with the requirements for entry into the Applicability of other specifications. These requirements are delineated in ITS LCO 3.0 and ITS SR 3.0.

This change does not result in any hardware changes or changes to plant operating practices nor does it affect plant operation. Therefore this change will not involve a significant increase in the probability or 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?

The proposed change modifies the Frequency of the refueling water level SR by removing the requirement to perform within 2 hours prior to the start of moving fuel assemblies or CEAs in the reactor pressure vessel. The Frequency for this SR is once per 24 hours which requires that the SR be performed within 24 hours prior to entering the Applicability. Removal of the requirement to perform this SR within 2 hours prior to entering the Applicability is consistent with the requirements for entry into the Applicability of other specifications. These requirements are delineated in ITS LCO 3.0 and ITS SR 3.0.

This change will not alter the plant configuration (no new or different type of equipment will be installed) or change the methods of governing normal plant operation. This change will not alter assumptions made in the safety analysis or licensing basis. Therefore, this change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

TECHNICAL CHANGES - LESS RESTRICTIVE
(ITS 3.9.7 Discussion of Changes Labeled L.1) (continued)

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

The proposed change modifies the Frequency of the refueling water level SR by removing the requirement to perform within 2 hours prior to the start of moving fuel assemblies or CEAs in the reactor pressure vessel. The Frequency for this SR is once per 24 hours which requires that the SR be performed within 24 hours prior to entering the Applicability. Removal of the requirement to perform this SR within 2 hours prior to entering the Applicability is consistent with the requirements for entry into the Applicability of other specifications. These requirements are delineated in ITS LCO 3.0 and ITS SR 3.0.

This change will not reduce a margin of safety since it has no impact on safety analysis assumptions. Therefore, this change does not result in a reduction in a margin of safety.

## ENVIRONMENTAL ASSESSMENT

These proposed TS changes have been evaluated against the criteria for and identification of licensing and regulatory actions requiring environmental assessment in accordance with 10 CFR 51.21. It has been determined that the proposed changes meet the criteria for categorical exclusion as provided for under 10 CFR 51.22(c)(9). The following is a discussion of how the proposed TS changes meet the criteria for categorical exclusion.

10 CFR 51.22(c)(9): Although the proposed changes involve changes to requirements with respect to inspection or Surveillance Requirements with;

(i) the proposed changes involve No Significant Hazards Consideration (refer to the No Significant Hazards Consideration Section of this Technical Specification Change Request),

(ii) there is no significant change in the types or significant increase in the amounts of any effluent that may be released offsite since the proposed changes do not affect generation of any radioactive effluent not do they affect any of the permitted release paths, and

(iii) there is no significant increase in individual or cumulative occupational radiation exposure.

Accordingly, the proposed changes meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Based on the aforementioned and pursuant to 10 CFR 51.22(b), no environmental assessment or environmental impact statement need be prepared in connection with issuance of an amendment to the Technical Specifications incorporating the proposed changes of this request.