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SUBJECT: Application for amends to licenses NPF-41,NPF-51 & NPF-74, reducing spurious reactor trip hazards associated with setpoints while maintaining plant protection.

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10 CFR 50.90
10 CFR 50.91

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U.S. Nuclear Regulatory Commission
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Dear Sirs:

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

Arizona Public Service Company (APS) requests an amendment to Technical Specification 3.3.1, Reactor Protective System (RPS) Instrumentation - Operating, for each Palo Verde Nuclear Generating Station (PVNGS) Unit. The proposed amendment would change the allowable values in Technical Specification section 3.3.1, Table 3.3.1-1, Item 12 "Reactor Coolant Flow, Steam Generator #1-Low" and Item 13 "Reactor Coolant Flow, Steam Generator #2-Low." This change is required to reduce spurious reactor trip hazards associated with these setpoints while maintaining plant protection.

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

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

A001

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

APS requests 60 days to implement the approved Technical Specification amendment. The 60 days is required to implement setpoint changes in all three units.

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

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

Sincerely,

James M Levine

JML/SAB/RKR/mah

Enclosure

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

STATE OF ARIZONA)
) ss.
COUNTY OF MARICOPA)

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

J. M. Levine

Sworn To Before Me This 26 Day Of May, 1999.

Notary Public

My Commission Expires

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

ENCLOSURE

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

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

A. NEED FOR THE AMENDMENT

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

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

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

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

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

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

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

B. DESCRIPTION OF THE PROPOSED TECHNICAL SPECIFICATION AMENDMENT

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

C. PURPOSE OF THE TECHNICAL SPECIFICATION

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

D. SAFETY ANALYSIS OF THE PROPOSED TECHNICAL SPECIFICATION AMENDMENT

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

One other event was also evaluated to verify that the consequences of the event are not affected by this change. This second event involves a large steam line break during full power operation with concurrent loss of offsite power (LOP) in combination with a single failure, and a stuck control element assembly (UFSAR 15.1.5). As described in UFSAR Section 15.1.5.1, large steam line breaks in containment are analyzed to maximize the potential for a post-trip return to power, and steam line breaks outside containment are

analyzed to maximize the potential fuel degradation and to maximize exclusion area boundary (EAB) dose. Taking credit for the low reactor coolant flow trip rather than the low RCP shaft speed trip for the large steam line break in containment reduces the potential for a post trip return to power as described in the UFSAR, but may increase the potential for pretrip fuel degradation. Therefore, the large steam line break during full power operation with concurrent LOP in combination with a single failure, and a stuck control element assembly event was reanalyzed to determine the effect of the delayed reactor trip on the analysis described in UFSAR Chapter 15.

<u>RCP Sheared Shaft</u>

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

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

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

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

The doppler and void coefficients are primarily responsible for turning DNBR around once three RCP flow has been reached. The time of control rod insertion (i.e., timing of the reactor trip) primarily influences the rate of DNBR recovery and thus relates to DNBR

propagation. These two distinct cause-and-effect relationships are fundamental to the sheared shaft event.

The reanalysis of the sheared shaft event using the same methodology as the original analysis, determined that the overall effect of the changes to the allowable values for the low reactor coolant flow trip function was to delay the RPS initiated low reactor coolant flow reactor trip for this event from the current value of approximately 1.2 seconds after event initiation to approximately 2.5 seconds after event initiation. The reanalysis of the sheared shaft event concluded that delaying the reactor trip would result in approximately the same minimum DNBR as previously analyzed. This is expected since the RCP coastdown characteristics are not being changed. Furthermore, although the time-in-DNB-condition (DNBR propagation) increases from approximately 2.6 seconds to approximately 3.9 seconds as a result of the delay in reactor trip, it remains below the limiting time (4.5 seconds) for the strain limit to be reached. Thus, DNBR propagation is also not a concern.

The reanalysis also evaluated the impact of extending the total trip time from approximately 1.2 seconds to approximately 2.5 seconds and assuming a LOP at approximately 3 seconds after the trip. The reanalysis showed that the minimum DNBR was relatively unchanged. This is expected because at 2 seconds into the event - close to the time of minimum DNBR - flow reaches the flow for three RCPs operating. Therefore, the reanalysis concluded that the UFSAR-documented case of 25% fuel failure and EAB dose consequences of 240 Rem remains bounding for the sheared shaft event.

<u>Large Steam Line Break</u>

The large steam line break described in UFSAR Section 15.1.5 is a limiting fault event. The specific cases described maximize the potential for a post-trip return to power, the potential for degradation in fuel cladding performance, and the potential dose at the site EAB. For this event, the change in the allowable values for the low reactor coolant flow trip function results in a delay in the reactor trip from the current time delay of approximately six seconds after event initiation to a delay of approximately 11 seconds after event initiation. A delay in the reactor trip will not affect the large steam line break post-trip return to power case, because the delayed reactor trip delays the RCS cooldown and thus dampens power increase. Normally the low RCP shaft speed trip is credited in this event since it results in a faster reactor trip and maximizes the potential for a post-trip return to power.

The large steam line break event was reanalyzed to determine if the dose consequences associated with this event would still be bounded by the analysis of record in the UFSAR. Fuel failure is assumed to occur when the minimum DNBR during the event, falls below the DNBR SAFDL value. The conclusion in UFSAR Section

15.1.5.4 states that "the 2-hour inhalation thyroid dose at the EAB is calculated to be no more than 42 rem, which is within the 10CFR100 guidelines." The 2-hour inhalation thyroid dose at the EAB was originally calculated based on 0.7% fuel failure. The UFSAR analysis assumed the 0.7% fuel failure in the specific dose calculations in order to provide conservatively bounding results. Subsequent cycle specific reload analysis of steam line break events maintain a higher Required Over Power Margin (ROPM) in the Core Operating Limit Supervisory System (COLSS) to prevent fuel failure. The reanalysis (using the same methodology as the cycle specific reload analysis) of the steam line break event using the proposed delay in the reactor trip determined that there would be no fuel failures. Since the UFSAR determination of the dose consequences of this event assumed 0.7% fuel failure, the dose consequences of this event remain bounded by the UFSAR analysis. Therefore, the reanalysis of the large steam line break event, with the reactor trip delayed from six seconds to 11 seconds, demonstrates that the consequences of this event remain within acceptable limits.

E. <u>NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION</u>

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

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

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

These changes will result in an increased time delay for the RPS low reactor coolant flow trip. The reanalysis of the affected UFSAR Chapter 15 events (UFSAR 15.3.4, Reactor Coolant Pump Shaft Break with Loss of Offsite Power and UFSAR 15.1.5, Steam System Piping Failures Inside and Outside Containment - Modes 1 and 2 Operations), with the increased time delay, shows that the dose consequences for these events remain

bounded by the UFSAR analysis. Therefore, this change does not involve a significant increase in the consequences of an accident previously evaluated.

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

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

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

No. The proposed change will change the RPS reactor coolant flow trip setpoints. The reanalysis of the affected UFSAR Chapter 15 events (UFSAR 15.3.4, Reactor Coolant Pump Shaft Break with Loss of Offsite Power and UFSAR 15.1.5, Steam System Piping Failures Inside and Outside Containment - Modes 1 and 2 Operations), with the revised reactor coolant flow trip setpoints, shows that the minimum DNBR and SAFDLs for these events remain bounded by the UFSAR analysis. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

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

F. <u>ENVIRONMENTAL CONSIDERATION</u>

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