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FACIL: STN-50-528 Palo Verde Nuclear Station, Unit 1, Arizona Publi 05000528
STN-50-529 Palo Verde Nuclear Station, Unit 2, Arizona Publi 05000529

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RECIP. NAME	RECIPIENT AFFILIATION
KNIGHTON, G. W.	PWR Project Directorate 7

SUBJECT: Forwards addl info in support of 860723 application for amends to Licenses NPF-41 & NPF-51, changing Tech Spec Tables 2.2-1 & 3.3-2 under exigent circumstances to avoid spurious reactor trips.

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Arizona Nuclear Power Project
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August 26, 1986
ANPP-37978-JGH/BJA/98.05

Director of Nuclear Reactor Regulation
Attention: Mr. George W. Knighton, Project Director
PWR Project Directorate #7
Division of Pressurized Water Reactor Licensing - B
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: Palo Verde Nuclear Generating Station (PVNGS)
Units 1 and 2
Docket Nos. STN 50-528 (License No. NPF-41)
STN 50-529 (License No. NPF-51)
Additional Information on Technical Specification
Change Request
File: 86-F-005-419.05; 86-E-056-026; 86-F-056-026

Reference: (1) Letter from J. G. Haynes, ANPP, to G. W. Knighton, NRC, dated July 23, 1986 (ANPP-37463). Subject: Request for Exigent Technical Specification Change.

Dear Mr. Knighton:

Reference (1) requested changes to PVNGS Units 1 and 2 Technical Specification Tables 2.2-1 and 3.3-2 under exigent circumstances in order to avoid spurious reactor trips and lower the probability of the Units being in a transient condition. The proposed changes involved setpoint changes to the low reactor coolant flow reactor trip function. Subsequent to the ANPP submittal of the proposed changes, the NRC Staff has requested additional information on these changes. The requested additional information is provided in the attachment to this letter along with a justification for exigent classification.

Your prompt attention to this matter is appreciated. If you have any additional questions, please contact Mr. W. F. Quinn of my staff.

Very truly yours,

JG Haynes

J. G. Haynes
Vice President
Nuclear Production

JGH/BJA/dlm
Attachment

cc: O. M. DeMichele (all w/a)
E. E. Van Brunt, Jr.
E. A. Licitra
R. P. Zimmerman
A. C. Gehr

A001
1/1

A. REVISED DESCRIPTION OF PROPOSED AMENDMENT REQUEST

The purpose of this amendment request is to change the setpoints involved with the Low Reactor Coolant Flow (LRCF) reactor trip function at PVNGS Units 1 and 2. The reason for this change is that process noise in the impulse lines of the differential pressure sensors has tended to cause unnecessary pre-trip alarms and channel trips at PVNGS. The specific Technical Specifications affected by this amendment request are Tables 2.2-1 and 3.3-2 of the PVNGS Units 1 and 2 Technical Specifications. The reactor trip setpoints for the LRCF trip function are to be adjusted within the bounds of the current safety analyses so that process noise can be accommodated without tripping the reactor.

The LRCF trip function provides protection for a Reactor Coolant Pump (RCP) sheared shaft event and a main steam line break with a concurrent loss of offsite power. In both of these events, the reduction in Reactor Coolant System (RCS) flow causes a reduction in the differential pressure across the primary side of the affected steam generator. The LRCF trip function uses a Rate Limited Variable Setpoint module to initiate a reactor trip based on the differential pressure input signal.

Under steady state conditions, the trip setpoint will stay below the differential pressure input signal by the trip function parameter STEP. During a transient, the trip setpoint will move away from the decreasing differential pressure input signal to try and maintain the separation defined by STEP. The rate of decrease of the trip setpoint is fixed by the trip function parameter RATE. If the rate of decrease of the differential pressure input signal is greater than RATE, a trip will occur when the differential pressure input signal eventually equals the trip setpoint. The minimum value that the trip setpoint can have is defined by the trip function parameter FLOOR.

Both loss of flow events are over quickly. The setpoint calculation uses a combination of the STEP, RATE, and FLOOR trip function parameters to provide the protection required. The trip function parameter FLOOR is used to provide protection for both loss of flow events whenever the Steam Generator differential pressure is less than or equal to 22.5 psid. The trip function parameters STEP and RATE are used to provide protection for both loss of flow events whenever the Steam Generator differential pressure is greater than 22.5 psid.

The total channel response time used in the safety analysis has been selected to permit initiation of a reactor trip during both loss of flow events at the lowest possible differential pressure. This permitted a decrease in the trip function parameter FLOOR and an increase in the trip function parameter STEP. The decrease in the FLOOR permitted increased operating space between the trip setpoint and the differential pressure input signal at lower operating differential pressures. The increase in STEP permitted increased operating space between the trip setpoint and the differential pressure input signal at higher operating differential pressures. A larger STEP will move the trip setpoint further away from the peak-to-peak variations in the process and decrease the trip function's sensitivity to process noise. In addition, the bistable delay time has been increased by 0.100 seconds to decrease the trip function's sensitivity to high frequency process noise.

These Technical Specification changes to the LRCF setpoints are expected to eliminate the frequent pretrips and channel trips that have been experienced at PVNGS. Thus, the changes will prevent spurious trips and lower the probability of the PVNGS units being in a transient condition.

B. REVISED BASIS FOR NO SIGNIFICANT HAZARDS DETERMINATION

The proposed changes to the LRCF trip setpoints do not involve a Significant Hazards Consideration because:

(A) The operation of PVNGS Units 1 and 2 in accordance with this change would not:

1) Involve a significant increase in the probability or the consequences of an accident previously evaluated. The accidents that have the potential for being impacted by the proposed changes are the reactor coolant pump shaft break event and the main steam line break with loss of offsite power event. The proposed setpoint changes are still within the safety analysis requirements for the two impacted events. Therefore, the probability or consequences of the previously analyzed accidents are not affected by this change.

2) Create the possibility of a new or different kind of accident from any accident previously analyzed. The proposed changes affect an existing reactor trip function. No change in plant hardware is involved. Operation of the facility with the proposed setpoint change will not create the possibility of a new or different kind of accident from any accident previously evaluated since the changes are still within the existing safety analysis.

3) Involve a significant reduction in a margin of safety. Operation of the facility with the proposed setpoint changes may reduce in some way the safety margin that was provided by the existing setpoints. The safety analyses ensure that the acceptance criteria for a postulated RCP shaft break or a steam line break with loss of offsite power are met by specification of an implicit time interval in which the reactor must trip. While some of the proposed setpoints are less conservative than the existing setpoints, the proposed setpoints are within the requirements of the existing safety analyses and all safety analysis acceptance criteria are met with the proposed setpoints.

(B) The NRC has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists. This proposed amendment matches one of the examples presented in 48FR14864. Specifically, the proposed amendment is a change which either may result in some increase to the probability or consequences of a previously-analyzed accident or may reduce in some way a safety margin, but where the results of the change are clearly within all acceptable criteria with respect to the system or component specified in the Standard Review Plan. As described under item #3 above, this change may in some way reduce the margin of safety provided by the existing setpoints, but the acceptance criteria of the affected safety analyses are still satisfied with these setpoint changes.

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[illegible]

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[illegible]

[illegible]

[illegible]

C. JUSTIFICATION FOR EXIGENT CLASSIFICATION

The NRC regulation 10CFR50.91(a)(6) provides that the commission may act quickly when an amendment involves no significant hazards considerations and exigent circumstances exist. The NRC's Statements of Consideration for 10CFR50.91(a)(6) provide two examples of exigent circumstances and describe exigency as "a situation other than an emergency where swift action is necessary". NCR's Responses to Comments on this Interim Final Rule, 51FR7744 (March 6, 1986), explains that these examples of exigent circumstances "were meant merely as guidance and were meant to cover circumstance where a net safety benefit might be lost if an amendment were not issued in a timely manner" and explain that "the examples should be read as also covering those circumstances where there is a net increase in safety or reliability...".

We believe exigent circumstances exist since the operating reactor could experience a plant shutdown due to potential spurious channel trips. The potential plant trips could subject the units to unnecessary transients which are not in the best interests of nuclear safety and also cause a loss of power generation capability. Thus, a net safety benefit exists in accordance with NRC Commission criteria. PVNGS Unit 1 has been experiencing single channel trips and double channel pre-trips including a previous trip which occurred on July 12. On August 9, with Unit 1 in MODE 3, a reactor trip signal was received when 2 channels simultaneously generated trip signals. If the unit had been at power, it would have tripped. This potential for plant trips will exist until the requested change has been approved and implemented.

D. NRC QUESTION

Are the measurement channel response times presented on page 19 of the July 23, 1986 letter worst case values?

ANPP RESPONSE

The response times given on page 19 are worst case values based on information from the vendors. The total of 580 msec. is less than the present Technical Specification requirement of 650 msec. However, it should be noted the test procedure that is used to periodically verify this response time requires a response time of less than 500 msec. Testing experience thus far has shown that the worst case response time has been approximately 350 msec.

[illegible]

[illegible]
[illegible]

[illegible]

[illegible]

E. NRC QUESTION

Explain the inconsistency between the sequence of events for the Main Steam Line Break (MSLB) analyses with concurrent loss of offsite power presented in FSAR Section 15.1. Specifically, the SOE states that a CPC trip or RCS low flow trip is received at .6 seconds after event initiation. The new setpoint analysis shows that the RCS low flow trip is not received until approximately 6 seconds after event initiation.

ANPP RESPONSE

As background, it should be noted that the LRCF trip function provides a secondary protection function for mitigation of the MSLB events with concurrent loss of offsite power. The primary protection for this event is provided by the Core Protection Calculators (CPCs). The LRCF trip function is only needed in the unlikely event of a CPC failure to trip. There are not any analyzed conditions that would result in the inability of the safety-grade CPCs to provide the required protection for this event.

Steam line breaks inside containment are analyzed to observe the potential for post-trip return to criticality. Early reactor trips tend to increase the potential for a post-trip return to criticality because of the large cooldown which takes place during the early part of the MSLB. Consequently, the CPC trip at .6 sec is employed.

Steam line breaks outside containment are analyzed to observe pre-trip fuel failure and subsequent offsite dose consequences. Breaks outside containment bound breaks inside containment with respect to offsite dose due to the direct path to the outside atmosphere.

FSAR Section 15.1 should not imply that the LRCF reactor trip is set to be coincident with the CPC trip. The CPC trip represents the worst case for post-trip return to criticality since it is earlier than the LRCF reactor trip.

The analysis trip setpoint value for the LRCF trip was chosen such that a coolable core geometry is maintained for this limiting fault.

This requirement for the LRCF trip function in relation to MSLB events with concurrent loss of offsite power has been maintained with these trip setpoint changes.

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

F. NRC QUESTION

Technical Specification setpoints and allowable values are very close (closer than existing). Explain why this margin was reduced and how the allowable values were arrived at.

ANPP RESPONSE

The proposed setpoints and allowable values are closer than the existing values for STEP and FLOOR. The proposed values for RATE are different than the existing RATE but the difference between the setpoint and the allowable value is the same as shown in the following table.

		SETPOINT	ALLOWABLE VALUE	DIFFERENCE
RATE	(old)	0.034 V/sec	0.035 V/sec	0.001
	(new)	0.016 V/sec	0.017 V/sec	0.001
STEP	(old)	1.284 V	1.351 V	0.067
	(new)	1.429 V	1.457 V	0.028
FLOOR	(old)	2.240 V	2.025 V	0.215
	(new)	1.700 V	1.671 V	0.029

ANPP has evaluated the Rate Limited Variable Setpoint (RLVS) card functions and operation. This allowed us to take credit for the way that the card operates in the setpoint calculation. The main improvement allowed by the operation of the card was the removal of a 2% drift component from the worst case signal error computation. The drift component was eliminated because of the ability of the trip setpoint to follow the process measurement. This is explained and justified further on page 8 of the proprietary attachment of the July 23, 1986 letter to the NRC.

[illegible]

[illegible]

[illegible]	[illegible]	[illegible]
[illegible]	[illegible]	[illegible]
[illegible]	[illegible]	[illegible]
[illegible]	[illegible]	[illegible]

[illegible]

G. <u>NRC QUESTION</u>

How was the 22.5 psid value determined?

<u>ANPP RESPONSE</u>

The transition point of 22.5 psid was selected after evaluating the operating data for averaged differential pressure and was selected to optimize the step and floor trip setpoints.

[illegible]

[illegible]

[illegible]

[illegible]

H. NRC QUESTION

Please provide a sequence of events for the RCP sheared shaft event.

ANPP RESPONSE

The sequence of events for this event is provided below.

RCP SHEARED SHAFT

	Time (Seconds)
Event Initiation (shaft break)	0.0
Process reaches trip setpoint	0.2
Trip Signal Generated	1.05
Reactor Trip Breakers Open	1.2
CEAs begin to drop	1.5

Arizona Nuclear Power Project
P.O. BOX 52034 • PHOENIX, ARIZONA 85072-2034

August 26, 1986
ANPP-37978-JGH/BJA?98.05

Director of Nuclear Reactor Regulation
Attention: Mr. George W. Knighton, Project Director
PWR Project Directorate #7
Division of Pressurized Water Reactor Licensing - B
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: Palo Verde Nuclear Generating Station (PVNGS)
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JH 8/26

J. G. Haynes
Vice President
Nuclear Production

JGH/BJA/dlm
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cc: O. M. DeMichele (all w/a)
E. E. Van Brunt, Jr.
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Mr. George W. Knighton
Additional Information on Technical Specification
Change Request
ANPP- 37978
Page 2

bcc: A. C. Rogers (all w/a)
W. F. Quinn WFQ 8/26/86
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LCTS Coordinator
J. M. Allen
J. R. Bynum JRB 8/26/86
O. J. Zeringue
L. G. Papworth
J. R. Webb
R. A. Bernier RAB 8-25-86
C. Lewis
G. W. Sowers
S. Troisi
R. M. Butler RMB 8/26/86

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ANPP RESPONSE

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RCP SHEARED SHAFT

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Event Initiation (shaft break)	0.0
Process reaches trip setpoint	0.2
Trip Signal Generated	1.05
Reactor Trip Breakers Open	1.2
CEAs begin to drop	1.5

[illegible]