From: <Glenn.Michael@aps.com>
To: <MTM@nrc.gov>
Date: 02/21/2007 9:03:21 AM
Subject: FW: Statement in U2 Power Uprate Safety Evaluation

Mike,
Tom's note below contains the Unit 2 pages; I've attached the Units 1 and 3 SE pages. Our submittal for Unit 1&3 simply referenced the Unit 2 submittal for this section.

Glenn

<<U1,U3 SER.pdf>>

> ____
> From: Weber, Thomas N(Z00499)
> Sent: Friday, January 26, 2007 1:51 PM
> To: Mike Markley (NRC PM) (mtm@nrc.gov); Mel Fields (NRC Project
> Manager)
> Cc: Michael, Glenn A(Z01119)
> Subject: Statement in U2 Power Uprate Safety Evaluation
>
>
> Mike,
> The attached files show the sentence that may be in error (depending
> on how it is interpreted) from the U2 Power Uprate amendment (refer to
> page 53 of the U2 Safety Evaluation). I mentioned this last week in
> our phone call and Glenn is adding it to our list of topics to discuss
> on Wednesday. This was discovered by our safety analysis group
> recently as they were doing some routine safety analysis work and
> happened to stumble onto this in the PUR safety evaluation. The
> second file is from our submittal showing that the sentence in
> question was not something repeated from our submittal. We can wait
> to discuss this when I return from business travel.
>
>
>
> <<U2 PUR SER.pdf>> <<U2 PUR SUBMITTAL.pdf>>
>
>
> TN Weber...
> Licensing Section Leader
> PVNGS

CC: Thomas.N.Weber@aps.com

Mail Envelope Properties (45DC511F.4E3 : 11 : 13539)

Subject:	FW: Statement in U2 Power Uprate Safety Evaluation
Creation Date	02/21/2007 9:02:58 AM
From:	<Glenn.Michael@aps.com>
Created By:	Glenn.Michael@aps.com

Recipients
nrc.gov
TWGWPO01.HQGWDO01
MTM (Michael Markley)

aps.com
Thomas.N.Weber CC

Post Office	**Route**
TWGWPO01.HQGWDO01	nrc.gov
	aps.com

Files	**Size**	**Date & Time**
MESSAGE	1362	02/21/2007 9:02:58 AM
TEXT.htm	2896	
U1,U3 SER.pdf	224015	
U2 PUR SER.pdf	227604	
U2 PUR SUBMITTAL.pdf	120190	
Mime.822	790067	

Options

Expiration Date:	None
Priority:	Standard
ReplyRequested:	No
Return Notification:	None
Concealed Subject:	No
Security:	Standard

Junk Mail Handling Evaluation Results
Message is eligible for Junk Mail handling
This message was not classified as Junk Mail

Junk Mail settings when this message was delivered
Junk Mail handling disabled by User
Junk Mail handling disabled by Administrator
Junk List is not enabled
Junk Mail using personal address books is not enabled
Block List is not enabled

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

Received
NOV 29 2005
J.M Levine

November 16, 2005

James M. Levine
Executive Vice President, Generation
Mail Station 7602
Arizona Public Service Company
PO Box 52034
Phoenix, Arizona 85072-2034

SUBJECT: PALO VERDE NUCLEAR GENERATING STATION, UNITS 1, 2, AND 3 - ISSUANCE OF AMENDMENTS RE: REPLACEMENT OF STEAM GENERATORS AND UPRATED POWER OPERATIONS AND ASSOCIATED ADMINISTRATIVE CHANGES (TAC NOS. MC3777, MC3778, AND MC3779)

Dear Mr. Levine:

The Commission has issued the enclosed Amendment No. 157 to Facility Operating License No. NPF-41, Amendment No. 157 to Facility Operating License No. NPF-51, and Amendment No. 157 to Facility Operating License No. NPF-74 for the Palo Verde Nuclear Generating Station (PVNGS), Units 1, 2, and 3, respectively. The amendments consist of changes to the Technical Specifications (TSs) in response to your application dated July 9, 2004, as supplemented by letters dated June 2, June 3 (two letters), June 17, July 9 (two letters), July 19 (two letters), August 3, September 29, October 21, and November 1, 2005.

The amendments change the facility operating licenses and TSs to support replacement of the steam generators and subsequent operation at an increased maximum power level of 3990 megawatts thermal (MWt), a 2.94 percent increase from the current 3876 MWt for PVNGS Unit 1 and PVNGS Unit 3. The amendments also make administrative changes to the PVNGS Unit 2 TSs so that the changed pages would apply to the three PVNGS units.

A copy of the related Safety Evaluation (SE) is also enclosed. A draft version of the SE was issued for comment on November 2, 2005, and comments were provided by Arizona Public Service Company in a letter dated November 10, 2005. These comments have been incorporated into the final SE, as appropriate. The Notice of Issuance will be included in the Commission's next biweekly *Federal Register* notice.

Sincerely,

Mel B. Fields

Mel B. Fields, Senior Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. STN 50-528, STN 50-529,
and STN 50-530

Enclosures:
1. Amendment No. 157 to NPF-41
2. Amendment No. 157 to NPF-51
3. Amendment No. 157 to NPF-74
4. Safety Evaluation

cc w/encls: See next page

However, SGTR will be followed by reactor and turbine trip and loss of normal FW flow, forced RCS flow, and condenser vacuum. Cool-down is maintained by using the AFW and releasing steam through the ADVs.

Radiation monitors will initiate alarms to notify the operator and aid in event diagnosis. The EOPs include explicit instructions to guide the operator to a reactor cool-down. The objectives of the EOP guidance and corresponding operator actions are: use the ADVs to control pressure and avoid challenge to the MSSVs, diagnose the event and stabilize the plant, cool-down the plant using the ADVs on both SGs before SG isolation, do a manual main steam isolation, isolate the affected SG, and cooldown and maintain adequate RCS inventory.

The results indicate that the plant will not over pressurize and the MDNBR will remain well above the SAFDL limits. (The licensee states that the atmospheric radioactivity release will be within the 10 CFR Part 100 limits.)

The NRC staff has reviewed the licensee's analysis of the SGTR accident and concludes that the licensee's analysis has adequately accounted for operation of the plant at the proposed power level and was performed using acceptable analytical methods and approved computer codes. The NRC staff further concludes that the assumptions used in this analysis are conservative and that the event does not result in an overfill of the SG. Therefore, the NRC staff finds the proposed PUR acceptable with respect to the SGTR event.

4.3.7 Limiting Infrequent Events

4.3.7.1 AOOs in Combination With a Single Active Failure

The limiting infrequent event is designed to test the plant's capability to respond to extreme transient conditions. The acceptance criteria are based on (1) GDC 10, which requires that the RCS be designed with appropriate margin to ensure that the SAFDLs are not exceeded, (2) GDC 15, which requires that the RCS be designed with appropriate margin to ensure that the design conditions of the RCPB will not be exceeded, (3) GDC 26, which requires that the control rods be capable of reliably controlling reactivity changes to ensure that the SAFDLs are not exceeded, (4) GDC 27, which requires that the reactivity control systems be designed with appropriate margin for stuck rods to ensure that the capability to cool the core is maintained, (5) GDC 28, which requires that the reactivity control systems be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents do not result in damage to the RCPB greater than limited local yielding and do not cause sufficient damage to significantly impair the capability to cool the core, and (6) GDC 31, which requires that the RCS be designed with sufficient margin to ensure that the RCPB behaves in a non-brittle manner and that the probability of propagating fracture is minimized.

The licensee created a composite limiting transient to bound the MDNBR of infrequent events (including AOOs in combination with a single failure). It is assumed that the unspecified event degrades the DNBR to the SAFDL level. The most limiting single failure is LOP, resulting in the coast-down of all RCPs. This is combined with the maximum linear heat rate (LHR) produced by a CEAW. No single failures or operator errors can degrade DNBR more than the above circumstances; therefore, no other failures are assumed. Initial conditions conservatively assume a 116 percent power level due to a preexisting condition from the undefined AOO and a turbine trip coincident with reactor trip, although a 3 second delay exists. No operator action is assumed for 30 minutes after transient initiation.

The acceptance criteria are those for infrequent events (including AOOs with single failure), i.e., limited fuel damage and maximum RCS pressure within 110 percent of the RCS design value.

The analysis is based on the CENTS code supplemented by CETOP-D for DNBR (using the CE-1 CHF correlation), and the HERMITE code for the calculation of the initial conditions. The MDNBR is calculated using the more detailed TORC code.

Because such events are heat-up transients, it is implicitly postulated that the PSVs will keep the maximum pressure within acceptable limits. The results are comparable to those for a broken RCP shaft, i.e., limited fuel damage and no MDNBR propagation are predicted. The maximum pressure is within acceptance limits because the PSVs have sufficient capacity to relieve overpressure. These transient analyses provides confidence that the limiting infrequent events (including AOOs in combination with a single failure) are well within prescribed limits.

The NRC staff has reviewed the licensee's analyses of the limiting hypothetical AOO transient with LOP. The NRC staff concludes that the licensee's analyses have adequately accounted for operation of the plant at the proposed power level and that the analyses were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the RPS will continue to ensure that the MDNBR and the peak RCS pressure will remain within the acceptance limits for this hypothetical event. In addition, core geometry and LTC will remain within acceptable limits for such an event. On this basis, the NRC staff concludes that the plant will continue to meet the requirements of GDCs 10, 15, 26, 27, and 31 for this hypothetical event. Therefore, the NRC staff finds the proposed bounding transient acceptable.

4.3.7.2 Anticipated Transient Without Scram (ATWS)

The pressure vessel has a certain depressurization capability and a maximum pressure design value. Certain vessels have been judged to have designed pressures marginally below the maximum pressure expected to develop in an ATWS. Pressure vessels in CE plants are in this category. The ATWS rule requires an independent and diverse shutdown system. With such a system, the probability of an ATWS event is thought to be acceptably low. Therefore, the NRC staff review for an ATWS event in a CE plant is to assure that a diverse and independent shutdown system is available.

According to 10 CFR 50.62, PWRs manufactured by CE must be equipped with systems diverse and independent from the reactor trip system to scram the reactor, trip the turbine, and initiate AFW under conditions of an ATWS. The licensee indicates that the existing system does not need any modification or resetting due to the PUR and installation of the RSGs. Inasmuch as the effects of the ATWS transient depend only on the vessel peak pressure design value, the vessel venting capability, and the presence of the diverse and independent shutdown system, the PUR and the RSGs, which do not affect these items, are irrelevant to the reactor's response to an ATWS.

U2 PUR SER

September 29, 2003

Mr. Gregg R. Overbeck
Senior Vice President, Nuclear
Arizona Public Service Company
P.O. Box 52034
Phoenix, AZ 85072-2034

SUBJECT: PALO VERDE NUCLEAR GENERATING STATION, UNIT 2 (PVNGS-2) - ISSUANCE OF AMENDMENT ON REPLACEMENT OF STEAM GENERATORS AND UPRATED POWER OPERATIONS (TAC NO. MB3696)

Dear Mr. Overbeck:

The Commission has issued the enclosed Amendment No. 149 to Facility Operating License No. NPF-51 for the Palo Verde Nuclear Generating Station, Unit 2. The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated December 21, 2001, as supplemented by letters dated March 13, August 27, August 29, September 4, September 6, October 11, November 21, December 10, December 23, 2002, and March 11, June 10, July 25, and August 22, 2003.

The amendment changes the operating license and TSs to support replacement of the steam generators and subsequent operation at an increased maximum power level of 3990 MWt, a 2.94 percent increase from the current 3876 MWt. The amendment shall be implemented prior to entry into Mode 4 during the restart from the Fall 2003 refueling outage.

A copy of the related Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's next biweekly *Federal Register* notice.

Sincerely,

/RA/
Bo M. Pham, Project Manager, Section 2
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. STN 50-529

Enclosures: 1. Amendment No. 149 to NPF-51
2. Safety Evaluation

cc w/encls: See next page

Radiation monitors, low pressurizer level, and high SG level let the operator diagnose SGTR and trip the plant manually before reaching the reactor trip point. This will keep the ADV open for a longer period of time and maximize emissions.

The analysis of an SGTR is based on the CENTS code. The analysis covers two events, SGTR and a stuck-open ADV creating an excess steam demand. This transient is a limiting event. The plant EOPs provide operator instructions for plant recovery.

The results indicate that the behavior of the PUR plant configuration with the RSGs is similar to the existing plant configuration. EOPs are designed to preclude pressurization and challenge to the MSSVs, aid diagnosis and plant stabilization, accomplish functional recovery, provide post-tube-rupture tube coverage, maintain adequate RCS inventory, and accomplish shutdown and depressurization.

From a reactor protection point of view, the results are acceptable because the plant does not over pressurize nor does it sustain any fuel damage during the transient. As mentioned above, the radiation consequences of this event are discussed in Section 4.4 of this SE.

4.3.6.2.1 Steam Generator Tube Rupture With Concurrent Loss of Offsite Power (No Stuck-Open ADV)

As in the previous case, upon tube rupture, primary water will mix on the shell side and reach the atmosphere via the turbine, the condenser, and the condenser air removal pumps. However, SGTR will be followed by reactor and turbine trip and loss of normal FW flow, forced RCS flow, and condenser vacuum. Cool-down is maintained by using the AFW and releasing steam through the ADVs.

Radiation monitors will initiate alarms to notify the operator and aid in event diagnosis. The EOPs include explicit instructions to guide the operator to a reactor cool-down. The objectives of the EOP guidance and corresponding operator actions are: use the ADVs to control pressure and avoid challenge to the MSSVs, diagnose the event and stabilize the plant, cool-down the plant using the ADVs on both SGs before SG isolation, do a manual main steam isolation, isolate the affected SG, and cooldown and maintain adequate RCS inventory.

The results indicate that the plant will not over pressurize and the MDNBR will remain well above the SAFDL limits. (The licensee states that the atmospheric radioactivity release will be within the 10 CFR Part 100 limits.)

The NRC staff has reviewed the licensee's analysis of the SGTR accident and concludes that the licensee's analysis has adequately accounted for operation of the plant at the proposed power level and was performed using acceptable analytical methods and approved computer codes. The NRC staff further concludes that the assumptions used in this analysis are conservative and that the event does not result in an overfill of the SG. Therefore, the NRC staff finds the proposed PUR acceptable with respect to the SGTR event.

4.3.7 Limiting Infrequent Events

4.3.7.1 AOOs in Combination With a Single Active Failure

The limiting infrequent event is designed to test the plant's capability to respond to extreme transient conditions. The acceptance criteria are based on (1) GDC 10, which requires that the

RCS be designed with appropriate margin to ensure that the SAFDLs are not exceeded, (2) GDC 15, which requires that the RCS be designed with appropriate margin to ensure that the design conditions of the RCPB will not be exceeded, (3) GDC 26, which requires that the control rods be capable of reliably controlling reactivity changes to ensure that the SAFDLs are not exceeded, (4) GDC 27, which requires that the reactivity control systems be designed with appropriate margin for stuck rods to ensure that the capability to cool the core is maintained, (5) GDC 28, which requires that the reactivity control systems be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents do not result in damage to the RCPB greater than limited local yielding and do not cause sufficient damage to significantly impair the capability to cool the core, and (6) GDC 31, which requires that the RCS be designed with sufficient margin to ensure that the RCPB behaves in a non-brittle manner and that the probability of propagating fracture is minimized.

The licensee created a composite limiting transient to bound the MDNBR of infrequent events (including AOOs in combination with a single failure). It is assumed that the unspecified event degrades the DNBR to the SAFDL level. The most limiting single failure is LOP, resulting in the coast-down of all RCPs. This is combined with the maximum linear heat rate produced by a CEAW. No single failures or operator errors can degrade DNBR more than the above circumstances; therefore, no other failures are assumed. Initial conditions conservatively assume a 116 percent power level due to a preexisting condition from the undefined AOO and a turbine trip coincident with reactor trip, although a 3 second delay exists. No operator action is assumed for 30 minutes after transient initiation. INCORRECT STATEMENT

The acceptance criteria are those for infrequent events (including AOOs with single failure), i.e., limited fuel damage and maximum RCS pressure within 110 percent of the RCS design value.

The analysis is based on the CENTS code supplemented by CETOP-D for DNBR (using the CE-1 CHF correlation), and the HERMITE code for the calculation of the initial conditions. The MDNBR is calculated using the more detailed TORC code.

Because such events are heat-up transients, it is implicitly postulated that the PSVs will keep the maximum pressure within acceptable limits. The results are comparable to those for a broken RCP shaft, i.e., limited fuel damage and no MDNBR propagation are predicted. The maximum pressure is within acceptance limits because the PSVs have sufficient capacity to relieve overpressure. These transient analyses provides confidence that the limiting infrequent events (including AOOs in combination with a single failure) are well within prescribed limits.

The NRC staff has reviewed the licensee's analyses of the limiting hypothetical AOO transient with LOP. The NRC staff concludes that the licensee's analyses have adequately accounted for operation of the plant at the proposed power level and that the analyses were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection system will continue to ensure that the MDNBR and the peak RCS pressure will remain within the acceptance limits for this hypothetical event. In addition, core geometry and long-term cooling will remain within acceptable limits for such an event. On this basis, the NRC staff concludes that the plant will continue to meet the requirements of GDCs 10, 15, 26, 27, and 31 for this hypothetical event. Therefore, the NRC staff finds the proposed bounding transient acceptable.

U2 PUR SUBMITTAL

Section 6.3.8 Limiting Infrequent Events

Section 6.3.8.1 Anticipated Operational Occurrences in Combination with a Single Active Failure

As an analytical simplification, a composite event was created to bound the DNBR degradation for all infrequent events - including AOOs in combination with a single active failure. When determining the actual limiting infrequent event, all combinations of initiating events and single active failures need to be evaluated. To avoid evaluating all of the potential initiating AOOs, the composite event assumes that an unspecified initiating event degrades all the thermal margins preserved by COLSS and brings core conditions to the DNBR SAFDL. This assumption is conservative since the AOOs are specifically analyzed to ensure that SAFDLs are not violated and the necessary thermal margin is preserved by the LCOs. The most limiting single active failure for DNBR degradation is a LOP, resulting in the coastdown of all four RCPs. Therefore, the composite event is defined as a LOF from SAFDL conditions.

The limiting AOOs for peak linear heat rate are the bank CEAW events presented in Section 6.3.4.1. There are no single active failures nor postulated operator errors that could occur with these events that would produce more severe consequences.

Section 6.3.8.1.1 Acceptance Criteria

As defined in the SRP Section 15.1.1, the specific acceptance criterion is:

- An incident of moderate frequency in combination with any single active component failure, or single operator error, should not result in loss of function of any barrier other than the fuel cladding. A limited number of fuel rod cladding perforations are acceptable.

Offsite radiological consequences must be limited to a small fraction of 10 CFR Part 100 guidelines (i.e., 30 REM thyroid, 2.5 REM whole body).

Section 6.3.8.1.2.1 Transient Simulation

The limiting AOO with single failure event is modeled as a LOF from SAFDL conditions. An undefined "Limiting AOO" is assumed to degrade all available COLSS margin and forces the hot channel DNBR to the SAFDL. At this point the limiting single failure, LOP, occurs and further degrades DNB. The SAFDL conditions include an assumed, pre-existing power of 116%, due to the undefined limiting AOO.

Although a LOP would not occur for at least three seconds following a turbine trip, this evaluation conservatively assumes a coincident turbine trip and LOP. The RCP coastdown leads to a CPC DNBR reactor trip. RCS flow coastdown degrades DNBR below the initial SAFDL conditions. DNBR degradation is terminated when the mitigating effects of the scram CEA insertion dominate the flow coastdown.

Section 6.3.8.1.3 Input Parameters, Initial Conditions, and Assumptions

Table 6.3-53 contains the initial conditions used for the LOF from SAFDL event.

The following assumptions are made in this analysis:

1. In accordance with Section 6.3.0.2, the initial conditions for the process variables were varied within the ranges of steady state operational configurations including the uncertainties to determine the set of initial conditions and input parameters that would produce the most adverse consequences.
2. There is no operator action for the first 30 minutes of the event.