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KNIGHTON,G.W. Licensing Branch 3

SUBJECT: Forwards response to NRC 841128 request for addl info re steam generator tube rupture analysis.

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TITLE: Licensing Submittal: PSAR/FSAR Amdts & Related Correspondence

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Arizona Public Service Company

ANPP-31324-EEVB/WFQ/KLM
December 4, 1984

Director of Nuclear Reactor Regulation
Attention: Mr. George W. Knighton, Chief
Licensing Branch No. 3
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: Palo Verde Nuclear Generating Station (PVNGS)
Units 1, 2 and 3
Docket Nos. STN-50-528/529/530
Steam Generator Tube Rupture Analysis
File: 84-056-026; G.1.01.10

Reference: (1) Letter to G.W. Knighton, NRC, from E.E. Van Brunt, Jr., APS, dated September 19, 1984; Subject: Steam Generator Tube Rupture Analysis.

(2) Letter to E.E. Van Brunt, Jr., APS, from G.W. Knighton, NRC, dated November 28, 1984.

Dear Mr. Knighton:

Attached are the responses to your questions regarding the Steam Generator Tube Rupture Analysis as requested by Reference (2).

If you should have any questions, please contact me.

Very truly yours,

E E Van Brunt /BSK

E. E. Van Brunt, Jr.
APS Vice President
Nuclear Production
ANPP Project Director

EEVBJr/KLM/no
Attachment

cc: E. A. Licitra w/a
A. C. Gehr w/a
R. P. Zimmerman w/a

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STATE OF ARIZONA)
) ss.
COUNTY OF MARICOPA)

I, Donald B. Karner, represent that I am Assistant Vice President, Nuclear Production of Arizona Public Service Company, that the foregoing document has been signed by me on behalf of Arizona Public Service Company with full authority to do so, that I have read such document and know its contents, and that to the best of my knowledge and belief, the statements made therein are true.

Donald B. Karner

Donald B. Karner

Sworn to before me this 4th day of December, 1984.

Nora E. Meador

Notary Public

My Commission Expires:
My Commission Expires April 6, 1987

RESPONSE TO NRC
REQUEST FOR ADDITIONAL INFORMATION

PALO VERDE NUCLEAR GENERATING STATION
UNITS 1, 2 AND 3

STEAM GENERATOR TUBE RUPTURE ANALYSIS

NRC QUESTION 1

In the response to question 1, it states that feeding the affected steam generator is not a deviation from CEN-152. Our position is that since the Palo Verde SG isolation strategy is different from the approved generic CEN-152 strategy, this is a deviation which should be justified.

RESPONSE

The PVNGS Procedure for mitigating a steam generator tube rupture (SGTR) has been augmented to include direction to the operator to feed the affected steam generator in order to keep the tubes covered and maintain the iodine partition coefficient. APS acknowledges that this is a deviation from CEN-152, and our justification is provided below.

By directing the operator to feed the affected steam generator, to cover the tubes, the iodine partition coefficient is maintained. By maintaining the iodine partition coefficient, offsite doses will be reduced. With this new procedural consideration, offsite doses will be reduced for any postulated SGTR. That is, a reduction in offsite doses would be realized for those scenarios which do not include a full open atmospheric dump valve failure.

This modification to the PVNGS Procedure will be incorporated before fuel load of PVNGS Unit 1. Training of the operators will commence soon after approval of the procedure modification, and will require approximately 3 months to train all of the Unit 1 shifts. This training should be complete by initial criticality. Training will include simulator time and will emphasize the reduction of offsite releases and the potential for overfill of the affected steam generator.

We believe this justifies the deviation from CEN-152, for the PVNGS SGTR Procedure.

NRC QUESTION 2

For a steam generator tube rupture, initially the secondary side of the affected steam generator will be fed by both the primary-to-secondary side leak and feedwater. This influx of water creates the potential for overfilling the steam generator. Discuss the information available to the operator to prevent overfilling the steam generator and the sensitivity of the time period from the start of the accident to the time when there could be an overfill problem, assuming the operator does not take any action to prevent overfilling. Alternately, show that the consequences of SG overfill are not significant.

RESPONSE

For a SGTR with loss of offsite power and a fully stuck open ADV, the steaming rate through the stuck open ADV is significantly higher than the primary-to-secondary side leak during the entire reported period of the transient. Therefore, the influx of leak flow from the primary system alone will not create the potential for overfilling the steam generator. The operator, after taking manual control of the auxiliary feedwater system, first raises the affected SG level above the top of the U-tubes. Thereafter, the auxiliary feedwater flow is throttled to maintain the level above the top of the SG U-tubes at about 71.5% wide range. This level prevents any overfilling of the affected steam generator. The operator will continuously rely on the SG level measurements (12 Class IE indicators per SG) for information all through the transient to keep the level below the acceptable limit of about 90% wide range in both generators. There are audio and visual alarms that actuate when it appears that the SG is being overfilled. It is these alarms and MSIS actuation which would provide overfill protection assuming no operator action. If at anytime there is a concern regarding SG overfill, the auxiliary feedwater to the affected steam generator will be temporarily terminated.

Combustion Engineering Emergency Procedure Guidelines (CEN-152) provides suggestions regarding steam generator overfill during a SGTR event mitigation. These suggestions include draining to radwaste system and steaming the generator.

NRC QUESTION 3

Evaluate the sensitivity of the time period the SG tubes could be uncovered to the increase in radiological consequences. Relate this study to the amount of tube uncovery without credit for manual SG level control.

RESPONSE

In the analysis of a SGTR with loss of offsite power and a fully stuck open ADV, the first operator action taken to recover the level in the affected SG was assumed to occur two minutes after isolation of the auxiliary feedwater flow to the affected SG. The action consisted of overriding the auxiliary feedwater isolation signal in order to start feeding the affected steam generator again. Two minutes subsequent to this action the operator takes manual control of the auxiliary feedwater system and starts feeding the affected steam generator with both auxiliary feedwater pumps. The actions were taken to raise the level in the affected SG above the top of the U-tubes as quickly as allowed by the emergency operating procedure since the magnitude of the offsite radiation dose is sensitive to the duration of SG tube uncovery.

In order to limit the doses within acceptable limits the operator needs to take timely actions. The current analysis assumed the operator takes manual control of the auxiliary feedwater system approximately 5 minutes after opening the ADV on each SG or 2 minutes after overriding the auxiliary feedwater system isolation signal. Calculations performed indicate that to limit the offsite doses to 10CFR100 guidelines, the operator will need to take manual control of the auxiliary feedwater system no later than approximately 12 minutes after opening of the ADV on each generator. The time interval between overriding the auxiliary feedwater isolation signal and taking manual control of the system is again 2 minutes. Therefore, within the constraints and conservatisms inherent in the current model, the operator can delay taking manual control of the SG level by approximately 12 minutes after the opening of the ADVs, and still limit the offsite doses to 10CFR100 guidelines.

NRC QUESTION 4

In your discussion of the steam generator tube rupture (Appendix); it states that 460 seconds (about 7 1/2 minutes) is the earliest possible time that the operator can take an adverse action. The bases for this statement was given as reference to ANSI Standard N660. Since the purpose of ANSI Standard N660 is not to determine the earliest time for operator to take "adverse" actions, our position is that inadequate bases have been provided to justify that the operator could not have opened the ADV earlier than 460 seconds. Therefore, determine the radiological consequences of the steam generator tube rupture with loss of offsite power assuming the operator opens an ADV on each steam generator at the earliest time possible that would result in the maximum radiological consequences.

RESPONSE

The largest contribution of the offsite dose during the event occurs in the time period between the opening of the ADVs and the time of recovery of the affected SG level above the U-tubes. This time period is greatly influenced by the inventory in the affected steam generator at the time that the ADVs are opened. In the current analysis the auxiliary feedwater flow to the affected SG is actuated on low SG level at about 177 seconds. Thereafter, the level is maintained in the narrow band between 25 and 30 percent wide range by the automatic operation of the auxiliary feedwater system. The SG level in the affected generator will be higher than 25 percent wide range prior to the auxiliary feedwater system actuation. Hence, opening of the ADVs at a prior time (that is, before 177 seconds) results in the inventory in the affected steam generator being higher than that calculated for the current analysis at the time the ADVs were opened. This means quicker recovery of the level in the affected steam generator since the inventory will be less depleted than for the current analysis.

Opening the ADV at an earlier time, when primary system pressure is higher, also causes increased primary-to-secondary flow through the postulated tube rupture. This offsets the level effects described above. Therefore, the overall impact on offsite doses is expected to be minimal. Analysis has verified that the most limiting offsite dose (pre-existing iodine spike) is increased by less than 5%. For the offsite dose with an event generated iodine spike, analysis has verified an increase of approximately 8%. This assumes that the sequence of events between the opening of the ADVs and the operator taking manual control of the auxiliary feedwater system is the same for both cases. Therefore, even if the operator was to open the ADVs at a time prior to that assumed in the analysis, resulting offsite doses would still be within 10CFR100 guidelines.