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SUBJECT: Forwards responses to NRC request for addl info re Generic
Ltr 86-06, "Reactor Coolant Pump Trip Strategy."

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References: (1) Letter from E. E. Van Brunt, Jr., ANPP, to USNRC Document Control Desk dated March 28, 1988 (161-00918). Subject: Schedule for Response to NRC Request for Information.
(2) Letter from E. A. Licitra, NRC, to E. E. Van Brunt, Jr., ANPP, dated February 23, 1988. Subject: Request for Additional Information, Palo Verde Response to Generic Letter 86-06.
(3) Letter from J. G. Haynes, ANPP, to E. A. Licitra, NRC, dated November 24, 1986 (ANPP-39138). Subject: Implementation of TMI Action Item II.K.3.5, "Automatic Trip of Reactor Coolant Pumps" (Generic Letter 86-06).
(4) Letter from F. J. Miraglia, NRC, to All Applicants and Licensees with Combustion Engineering (CE) Designed Nuclear Steam Supply Systems (NSSSs) (Except Maine Yankee) dated May 29, 1986. Subject: Implementation of TMI Action Item II.K.3.5, "Automatic Trip of Reactor Coolant Pumps" (Generic Letter 86-06).

Dear Sirs:

Subject: Palo Verde Nuclear Generating Station (PVNGS)
Units 1, 2 and 3
Response to NRC Request for Information -
Reactor Coolant Pump Trip Strategy
File: 88-A-056-026

By Reference (2), the NRC Staff has requested additional information concerning the ANPP response to Generic Letter 86-06 that had been previously submitted to the NRC by Reference (3). In response to this request, ANPP informed the NRC Staff that the requested information would be provided by June 30, 1988. The purpose of this letter is to provide the requested additional information. The ANPP responses to the NRC questions are provided in the attachment to this letter.

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If you have any additional questions on this matter, please contact Mr. A. C. Rogers at (602) 371-4041.

Very truly yours,

E. E. Van Brunt

E. E. Van Brunt, Jr.
Executive Vice President
Project Director

EEVB/BJA/jle
Attachment

cc: G. W. Knighton (all w/attachment)
M. J. Davis
T. J. Polich
J. B. Martin
A. C. Gehr

ATTACHMENT

ANPP Responses to NRC Questions -
Reactor Coolant Pump Trip Strategy

1. NRC QUESTION

Arizona Nuclear Power Project's (ANPP's) letter of November 24, 1986 did not clearly identify which of the criteria presented in CEN-268 was selected to trip the second set of pumps during a small break LOCA (SBLOCA) at Palo Verde, Units 1, 2, and 3 (PV-1, 2, 3). Identify the criterion selected and the setpoints used to determine when to trip the second set of pumps. Also, identify and provide justification for the pressure setpoint used to trip the first set of pumps if different from that recommended in CEN-268.

ANPP RESPONSE

To trip the first set of Reactor Coolant Pumps (RCPs), ANPP uses a pressure setpoint that is conservatively high with respect to the setpoint identified in CEN-268. CEN-268 recognizes that the pressure setpoint must be selected on a plant-specific basis. CEN-268 also presents a generic pressure setpoint of 1400 psia. ANPP has selected a value of 1837 psia to use as the setpoint for tripping the first set of RCPs. The selected setpoint of 1837 psia is also the setpoint used to generate a Safety Injection Actuation Signal (SIAS) and a low pressurizer pressure reactor trip. The advantage of using this setpoint is that the decision point is clear for the plant operators. This setpoint conservatively accounts for the post-accident instrument inaccuracies to ensure that the RCPs are tripped prior to when the actual Reactor Coolant System (RCS) pressure falls below the generic setpoint of 1400 psia.

The operators use two parameters to determine when to trip the second set of RCPs. The parameters are low subcooling margin and the presence of containment radiation alarms. During a LOCA, both of these parameters must be satisfied to trip the second set of RCPs. This is consistent with one of the trip strategies recommended by CEN-268. The containment radiation alarms are used in the diagnostic flow chart as one indicaton of a Loss of Coolant Accident (LOCA). The containment radiation alarms are set so as to indicate an abnormal increase in the containment radiation levels. As an example, containment area radiation monitors RU-148 and RU-149 are required to have an alarm setpoint of less than or equal to 10 R/hr. This required alarm setpoint is specified in Technical Specification Table 3.3-6. RCS subcooling is the other criterion used in determining whether to trip the second set of RCPs. ANPP has selected a setpoint of 28°F subcooling for this decision. This setpoint is conservatively high with respect to the setpoint identified in CEN-268. CEN-268 lists a generic subcooling setpoint of 20°F and also notes that a plant-specific setpoint is required in this case. The ANPP value of 28°F accounts for normal instrument inaccuracies to ensure that the second set of pumps are tripped prior to reaching the 20°F generic setpoint.

2. NRC QUESTION

For the setpoint identified in response to question 1, discuss how the effects of instrument uncertainty, as identified in ANPP's November 24, 1986 responses to Generic Letter (GL) 86-06 item 2, were included in determining the setpoints for pressure, subcooled margin, and the other parameters in the plant specific pump trip strategy.

ANPP RESPONSE

The instrument uncertainties shown in Table 1 of Reference (3) for adverse conditions were included in the calculations to determine the SIAS trip setpoint at PVNGS. This is the setpoint used to trip the first set of RCPs. The setpoint methodology uses measured worst case errors from equipment qualification testing. For example, the SIAS setpoint calculation adds a total of 225 psi to the safety analysis setpoint of 1600 psia. The actual SIAS setpoint is then established as 1837 psia. Note that this is very conservative with respect to the generic setpoint of 1400 psia for tripping the first set of RCPs.

For the containment radiation monitors, the specific alarm setpoint is not critical to the success of the RCP trip methodology. The alarm setpoints are chosen to indicate an abnormal increase in the containment radiation levels above the normal background levels. An abnormal increase would be indicative of a loss of RCS pressure boundary integrity.

The second parameter used as a decision to trip the second set of RCPs is RCS subcooling margin. During normal conditions inside the containment building (i.e., containment temperature less than 200°F), the subcooling margin setpoint is established as 28°F. However, during accident situations, where the containment internal temperature is greater than 200°F, a 40°F error is added to the CE generic setpoint of 20°F. This results in a subcooled margin trip setpoint of 60°F.

3. NRC QUESTION

Clarify how the normal conditions error of zero was determined for all the instruments listed in Table I of ANPP's November 24, 1986 response. The error appears to be unrealistic because any instrument has some uncertainty associated with the measurement it provides even under the best conditions. A footnote in the table states that the error provided is the tolerance added to an instruments reading by the operator. Is an error of zero used because the instrument circuit already accounts for possible instrument uncertainty?

ANPP RESPONSE

A brief explanation of the values provided in Table I of Reference (3) will help to clarify any misunderstandings concerning the table. The list of normal errors provided in the table was intended to indicate the correction that the operator makes to the instrument reading during normal conditions (i.e., containment temperature less than 200°F). Specifically, the operators do not apply any corrections to the instrument readings during normal conditions. The operators read the instrument and then compare that reading to the appropriate setpoints. However, the setpoints do account for the normal instrument errors. For instance, the 28°F subcooling margin setpoint accounts for 8°F of normal instrument error above the analysis setpoint of 20°F. Therefore, the operators do not need to correct the instrument readings since the setpoints already account for normal errors.

4. NRC QUESTION

ANPP did not provide sufficient information in its November 24, 1986 response to GL 86-06 item 3 to determine how the uncertainties in the generic analysis presented in CEN-268 affect the results as they apply to PV-1, 2, 3. Therefore, identify the PV-1, 2, 3 plant-specific features not representative of the reference plant used in the analyses presented in CEN-268. At a minimum discuss core power; decay heat; HPIS capacity; makeup flows; setpoints for steam generator secondary safety valves; setpoints for reactor trip, safety injection, and accumulator injection and show that the values used in the generic analysis are either representative of those at PV-1, 2, 3 or conservative. If a reference plant parameter is not representative for PV-1, 2, 3, discuss how this was considered in determining the plant specific setpoints. Alternatively, show that the pressure setpoint calculated from the equation provided in CEN-268, Supplement 1, in response to questions 48-55 using plant specific values as input is conservative relative to the 1400 psia setpoint recommended for PV-1, 2, 3 in CEN-268.

ANPP RESPONSE

To respond to this NRC question, ANPP has chosen to use the second alternative provided in the NRC request. This involves performing a plant-specific setpoint comparison using the equation provided in CEN-268, Supplement 1.

The 1400 psia pressure setpoint, determined in CEN-268, is a generic value for System 80 plants. To show that the 1400 psia value is conservative relative to the PVNGS specific setpoint, the equation given in Supplement 1 is solved using PVNGS specific values.

The equation is:

$$Q_{CORE} + Q_{RCP} - \frac{UA\ (T_{PRI} - T_{SEC})}{3600} = f(W_{LEAK}(P)) + q(W_{HPSI}(P))$$

As detailed in CEN-268, the RCP trip setpoint pressure is to be set above the RCS pressure during the pressure plateau phase of the transient. The secondary side pressure is determined by the lowest Main Steam Safety Valve (MSSV) setpoint. For PVNGS, the lowest MSSV setpoint is 1250 psia. At the pressure plateau, the two terms on the right hand side of the equation sum to zero. This leads to:

$$Q_{CORE} + Q_{RCP} = \frac{UA\ (T_{PRI} - T_{SEC})}{3600}$$

Rearranging these terms gives:

$$T_{PRI} = \frac{3600 \; (Q_{CORE} + Q_{RCP})}{UA} + T_{SEC}$$

The PVNGS specific values for the terms in this equation are as follows:

Q_{CORE} = This is the decay heat term. It is found by taking 4.3% of the 100% power level. The 4.3% decay heat value corresponds to the quickest time to reach the pressure plateau.

$$Q_{CORE} = (0.043)(3800 \text{ MW}) = 163.40 \text{ MW}$$

Q_{RCP} = This is the RCP heat input term. This term is equal to 22 MW.

A = This is the steam generator heat transfer area. Per CEN-268, fifty percent of the total steam generator heat transfer area is credited.

U = This is the steam generator overall heat transfer coefficient. This term is set at 600 BT/hr-ft^2-°F in accordance with the assumptions used in CEN-268.

T_{SEC} = This is the saturation temperature at the secondary side pressure of 1250 psia. The 1250 psia is determined by the lowest MSSV setpoint. This pressure leads to a saturation temperature of 572°F.

Substituting these values into the equation gives:

$$T_{PRI} = \frac{(163.40 \text{ MW} + 22 \text{ MW})}{(600 \text{ BTU/hr-ft-ft-°F})(\text{SG Area})} + (572°\text{F})$$

$$T_{PRI} = 580°\text{F}$$

The pressure corresponding to this saturation temperature is 1326 psia. Therefore, the PVNGS specific value is below the generic value of 1400 psia established in CEN-268. This demonstrates that the CEN-268 value conservatively bounds the PVNGS specific value.

The actual trip setpoint used at PVNGS is very conservative with respect to both of these values. The PVNGS RCS pressure setpoint, which is used to trip the first two RCPs, is 1837 psia. This value is determined in part by using the CEN-268 generic value of 1400 psia and 225 psia to account for instrument errors. The 1400 psia value was determined in CEN-268 to be the RCS pressure at which tripping the first two RCPs will minimize possible core uncovery. The resultant pressure of 1625 psia was then conservatively increased to match the SIAS setpoint of 1837 psia. This was done to facilitate and simplify operator response during depressurization events. The operator's are thus aware of the need to trip two RCPs whenever safety injection is automatically actuated.

Setting the pressure setpoint at a pressure above 1625 psia will satisfy the criteria of minimizing possible core uncovery with respect to RCP operation. The PVNGS specific RCS pressure setpoint for tripping the first two RCPs is therefore substantially more conservative than the generic setpoint of 1400 psia developed in CEN-268. In future SIAS setpoint determinations, the setpoint may vary. However, as long as the SIAS setpoint remains above the generic setpoint plus pressure instrument error, the CEN-268 criteria will be conservatively met.

5. NRC QUESTION

Additional information is needed to clarify ANPP's response to Item 4 of Generic Letter 86-06. This includes identifying the procedures which provide direction for use of individual steam generators with and without operating RCPs and discussing operator training for use of these procedures as well as all procedures requiring use of pump trip guidelines. Also, ANPP's response to Item 4 listed the following emergency operating procedures (EOPs) as those which require the use of reactor coolant pump trip guidelines:

1. 4[1,2, or 3]EP-[1,2, or 3]ZZ01: diagnostic aids.
2. 4[1,2, or 3]RO-[1,2, or 3]ZZ01 to 4[1,2, or 3]RO-[1,2, or 3]ZZ09: optimal recovery guidelines.
3. 4[1,2, or 3]RO-[1,2, or 3]ZZ10: functional recovery guideline.

Identify what situations, i.e., main steam line breaks, steam generator tube ruptures, small break LOCAs, etc., are covered by these EOPs.

ANPP RESPONSE

In response to the NRC request, the following list of procedures is provided. Note that this response is applicable to all three of the PVNGS units. Therefore, the unit specific designators have been removed from the list and have been replaced by the letter "X".

- 4XEP-XZZ01 - "Emergency Operations" - This procedure provides the standard post-trip actions in accordance with the guidance of CEN-152.
- 4XRO-XZZ01 - "Reactor Trip" - This procedure provides guidance for an uncomplicated reactor trip.
- 4XRO-XZZ02 - "Loss of Secondary Coolant" - This procedure provides guidance for a non-isolable break in a main steam line or a feedwater line. Guidance is provided for RCP trip, recovery of non-operating RCPs, and plant cooldown with a single steam generator (if required).
- 4XRO-XZZ03 - "Excessive Steam Demand" - This procedure provides guidance for an isolable main steam line break. Guidance is provided for RCP trip and recovery of non-operating RCPs.
- 4XRO-XZZ04 - "Loss of Forced Circulation" - This procedure provides guidance for an event involving the loss of all four RCPs.
- 4XRO-XZZ05 - "Loss of Feedwater" - This procedure provides guidance for a feedwater line break downstream of the check valves (i.e., isolable break). Guidance is provided for RCP trip as well as recovery of non-operating RCPs.

- 4XRO-XZZ06 - "Steam Generator Tube Rupture" - This procedure provides guidance for a steam generator tube rupture event. Guidance is provided for RCP trip, recovery of non-operating RCPs, cooldown with one or both steam generators, and cooldown with or without forced circulation.

- 4XRO-XZZ07 - "Loss of Coolant Accident" - This procedure provides guidance for a loss of RCS inventory event in which subcooling cannot be maintained. Guidance is provided for RCP trip, recovery of non-operating RCPs, and natural circulation cooldown.

- 4XRO-XZZ08 - "Small Loss of Coolant Accident" - This procedure provides guidance for small loss of RCS inventory events in which subcooling can be maintained. Guidance is provided for RCP trip and recovery of non-operating RCPs.

- 4XRO-XZZ09 - "Station Blackout" - This procedure provides guidance for an event involving the loss of all of the AC power sources. No RCPs will be available during a station blackout.

- 4XRO-XZZ10 - "Functional Recovery Procedure" - In accordance with CEN-152, this procedure provides guidance related to all of the above using a symptomatic approach.

- 4XAO-XZZ13 - "Natural Circulation Cooldown" - This procedure provides directions for performing a cooldown using one or both steam generators and no RCPs.

The PVNGS operators receive extensive training in the use of the EOPs. This training is conducted prior to initial operator licensing and periodically as part of the requalification training program. The training on the EOPs also covers the use of the RCP trip strategy discussed within this letter.

6. NRC QUESTION

Table I of ANPP's November 24, 1986 letter showed that two subcooled margin monitors were available, but two subcooled margin monitors were required for the pump trip criterion used at PV-1, 2, 3. Thus, this parameter had zero redundancy. Clarify what steps the operator would take to determine subcooling margin and trip the pumps, if one (or both) of the subcooled margin monitors should fail for any reason.

ANPP RESPONSE

There are several sources of information for determining subcooling margin. Subcooling margin is calculated and displayed on the two plasma display units of the Qualified Safety Parameter Display System (QSPDS) Trains A and B. Additionally, two recorders are provided on the main control board to display subcooling margin. One of the recorders displays the reactor vessel upper head saturation margin and the other recorder displays the core exit saturation margin. For both of these recorders, the input is from QSPDS Train A. Each train of QSPDS calculates and displays the following subcooling margins:

i) RCS saturation margin. This calculation uses pressurizer pressure and hot and cold leg temperatures.

ii) Reactor vessel upper head saturation margin. This calculation uses pressurizer pressure and the upper head temperatures from the Reactor Vessel Level Monitoring System (RVLMS) probes.

iii) Core exit saturation margin. This calculation uses pressurizer pressure and the representative core exit temperature from the core exit thermocouples.

There is no procedural or Technical Specification requirement for the operator to use a specific number of channels for his decision to trip the first or second set of RCPs. The operator's decision is based on the use of all available and operable instrumentation channels. This also enables him to identify a malfunctioning channel. Table I from the Reference (3) letter needs clarification in this respect. There is no procedural or other requirement that directs the operator to use a minimum of two subcooling margin channels in determining when to trip the second set of RCPs.

In the unlikely event that all of the subcooling margin indicators are unavailable, the operator can manually calculate subcooling margin. The operator can use the lowest reading of the pressurizer pressure instruments, the highest RCS temperature indication, and the steam tables to determine subcooling margin.

7. NRC QUESTION

Item 2 of Generic Letter 86-06 asked licensees to address local conditions such as fluid jets or pipe whip which might influence instrument reliability. ANPP's response to Item 2 in its November 24, 1986 letter did not address this issue. Provide this information for review.

ANPP RESPONSE

The effects of pipe whip and fluid jets were addressed during the design process at PVNGS. In general, a design review was conducted using the PVNGS model to ensure that equipment necessary to safely shut down the reactor was adequately protected from the effects of fluid jets and pipe whip. The protection of the essential equipment is accomplished by locating the equipment away from pipe break locations, enclosing the vital equipment within protective barriers, or by equipment redundancy considerations. For additional information, concerning the protection of equipment against local conditions such as fluid jets and pipe whip, refer to USAR Section 3.6.