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See Proposed Change to Tech Spec.

SUBJECT: Application for amends to Licenses NPF-41,NPF-51 & NPF-74, revising Tech Specs 3.4.1.3.3.4.1.4.1.3.4.8.1 & 3.4.8.3.

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WILLIAM F. CONWAY
EXECUTIVE VICE PRESIDENT
NUCLEAR

161-02990-WFC/JST
March 13, 1990

Docket Nos. STN 50-528/529/530

Document Control Desk
U. S. Nuclear Regulatory Commission
Mail Station P1-37
Washington, D. C. 20555

Dear Sirs:

Subject: Palo Verde Nuclear Generating Station (PVNGS)
Units 1, 2, and 3
Proposed Technical Specification Amendment to Sections 3.4.1.3, 3.4.1.4.1, 3.4.8.1, 3.4.8.3, 4.4.8.3.1, and B3/4.4.8 to incorporate the requirements of Generic Letter 88-11 "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and its Impact on Plant Operations"
File: 90-056-026

This letter requests an Amendment to the PVNGS Units 1, 2, and 3 Technical Specifications Sections 3.4.1.3, 3.4.1.4.1, 3.4.8.1, 3.4.8.3, 4.4.8.3.1 and B3/4.4.8. The proposed change would update the Reactor Vessel Pressure-Temperature (P-T) curves and Low Temperature Overpressure Protection (LTOP) enable temperatures, in accordance with the irradiation damage prediction methodology of Revision 02 of Regulatory Guide 1.99 "Radiation Embrittlement of Reactor Vessel Materials." This reanalysis utilizing Revision 2 of Regulatory Guide 1.99 was required by Generic Letter 88-11 and resulted in new Reactor Vessel P-T curves and LTOP enable temperatures. The proposed amendment would incorporate these changes into the PVNGS Technical Specifications. Arizona Public Service respectfully requests 45 days from the date of issuance to the effective date of the amendment to update the affected plant procedures.

Enclosed with this amendment request are:

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

Pursuant to 10 CFR 50.91(b)(1) a copy of this request has been forwarded to the Arizona Radiation Regulatory Agency.

A008
1/1

If there are any questions concerning this request, please contact A. C. Rogers of my staff at (602) 340-4041.

Sincerely,

WFC/JST/jle

Attachments

cc: T. L. Chan (all w/attachments)
J. B. Martin
T. J. Polich
A. C. Gehr
A. H. Gutterman
C. E. Tedford

STATE OF ARIZONA)
) ss.
COUNTY OF MARICOPA)

I, W. F. Conway, represent that I am Executive Vice President - Nuclear, 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 and correct.

WFConway
W. F. Conway

Sworn To Before Me This 13 Day Of March, 1990.

Linda B. Spell
Notary Public

My Commission Expires

My Commission Expires June 5, 1992

Attachments

A. Description of Amendment Request

Technical Specifications 3.4.1.3, note **, and 3.4.1.4.1, note ##, are changed to read:

> A reactor coolant pump shall not be started with one or more of the Reactor Coolant System cold leg temperatures less than or equal to 214°F during cooldown, or 291°F during heatup, unless the secondary water temperature (saturation temperature corresponding to steam generator pressure) of each steam generator is less than 100°F above each of the Reactor Coolant System cold leg temperatures.

A note is added to both specifications to maintain the analysis assumptions of flow induced pressure correction factors. The note reads as follows:

> Reactor Coolant Pump operation is limited to 2 Reactor Coolant Pumps with RCS cold leg temperature less than or equal to 200°F, 3 Reactor Coolant Pumps with RCS cold leg temperature greater than 200°F but less than or equal to 500°F.

Technical Specification 3.4.8.3 is changed to the new LTOP enable temperatures. It should be noted that for simplification of operation the LTOP enable temperatures for 32 EFPY were used since they are conservative and did not present any operational difficulties.

The new values of LTOP enable temperatures are based on the methodology of Regulatory Guide 1.99 Revision 2 and NRC Standard Review Plan 5.2.2. They are taken from an analysis performed by Combustion Engineering Inc.. The note added to each specification preserves the assumptions of flow induced pressure correction factors due to Reactor Coolant Pump operation used in the analysis.

Technical Specification 3.4.8.1 is changed to delete figure 3.4-2 and provide new P-T limit curves 3.4-2a and 3.4-2b and add maximum allowable heatup and cooldown rate table 3.4-3 for less than 8 and 8-32 EFPY, respectively. Subparagraphs (a) and (b) of the Limiting Condition for Operation are changed to read as follows:

> a. Maximum heatup and cooldown rates as specified in Table 3.4-3.
>
> b. A maximum temperature change of 10°F in any 1-hour period during inservice hydrostatic testing operations.

The new heatup and cooldown rate limit table and P-T limit curves represent the changes introduced by utilizing the methodology of Revision 2 of Regulatory Guide 1.99. The heatup and cooldown rate limits are presented in a tabular format to improve clarity.

Technical Specification Bases Section B3/4.4.8 is substantially changed to reflect the methodology used in the current analysis. Figure B 3/4.4-1 is eliminated as it is no longer applicable. The new Basis Section reads as follows:

The various categories of load cycles used for design purposes are provided in Chapters 3 and 5 of the FSAR. During startup and shutdown, the rates of temperature and pressure changes are limited so as not to exceed the limit lines of Figures 3.4-2a and 3.4-2b. This ensures that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation.

Reactor vessel pressure-temperature limitations and Low Temperature Overpressure Protection requirements for the Palo Verde Nuclear Generating Station are calculated to meet the regulations of 10 CFR Part 50 Appendix A, Design Criterion 14 and Design Criteria 31. These design criteria require that the reactor coolant pressure boundary be designed, fabricated, erected, and tested in order to have an extremely low probability of abnormal leakage, of rapid failure, and of gross rupture. The criteria also require that the reactor coolant pressure boundary be designed with sufficient margin to assure that when stressed under operation, maintenance, and testing; the boundary behaves in a non-brittle manner and the probability of a rapidly propagating fracture is minimized.

The pressure-temperature limits are developed using the requirements of 10 CFR 50 Appendix G. This appendix describes the requirements for developing the pressure-temperature limits and provides the general basis for these limitations. The margins of safety against fracture provided by the pressure-temperature limits using the requirements of 10 CFR Part 50 Appendix G are equivalent to those recommended in the ASME Boiler and Pressure Vessel Code Section III, Appendix G, "Protection Against Nonductile Failure." The general guidance provided in those procedures has been utilized to develop the Palo Verde pressure-temperature limits with the requisite margins of safety for heatup and cooldown conditions.

The pressure-temperature limits account for the temperature differential between the reactor vessel base metal and the reactor coolant bulk fluid temperature. Correction for elevation and RCS flow induced pressure differences between the reactor vessel beltline and pressurizer, are included in the development of the pressure-temperature limits as are instrumentation uncertainties for pressure and temperature measurement. Consequently the P-T limits are provided on coordinates of indicated pressurizer pressure versus indicated RCS temperature.

The pressure correction factors are based upon the differential pressure due to the elevation difference between the reactor vessel wall adjacent to the active core region and the pressurizer pressure instrument nozzle. This term of the pressure correction factor is equal to 29.62 psi. The pressure correction factors are also based upon flow induced pressure drops across the reactor core through the hot leg pipe up to the surge line nozzle. This term of the pressure correction factor has two values which are dependent upon the Reactor Coolant Pump (RCP) combination utilized during operation. At temperatures of $T_c < 200°F$, the flow induced pressure drop is based upon RCS flow rates resulting from two operating RCPs and is equal to 55.02 psi using post-core hot functional test data. At

[illegible]

temperatures of $T_c \geq 200°F$, the flow induced pressure drop is based upon the RCS flow rates resulting from three operating RCPs and is equal to 64.39 psi using post-core hot functional test data. The pressure correction factors also account for pressurizer pressure measurement uncertainty.

The Reactor Pressure Vessel beltline pressure-temperature limits are based upon the irradiation damage prediction method of Regulatory Guide 1.99 Revision 02. This methodology has been used to calculate the limiting material Adjusted Reference Temperatures (ART) for Palo Verde Units 1, 2, and 3. The adjusted reference temperatures of reactor vessel beltline materials for Palo Verde Units 1, 2, and 3 have been calculated at the 1/4T and 3/4T locations after 10 and 40 calendar years operation. By comparing the ART data for each material, the controlling materials for all three Palo Verde units, have been determined.

The analytical procedure for developing reactor vessel pressure-temperature limits utilizes the methods of Linear Elastic Fracture Mechanics (LEFM) found in the ASME Boiler and Pressure Vessel Code Section III, Appendix G in accordance with the requirements of 10 CFR Part 50 Appendix G. For these analyses, the Mode I (opening mode) stress intensity factors are used for the solution basis. The general method utilizes LEFM procedures. LEFM relates the size of a flaw with the allowable loading which precludes crack initiation. This relation is based upon a mathematical stress analysis of the beltline material fracture toughness properties as prescribed in Appendix G to Section III of the ASME code.

The reactor vessel materials have been tested to determine their initial RT_{NDT}; the results of these tests are shown in Table B 3/4.4-1. Reactor operation and resultant fast neutron (E greater than 1 MeV) irradiation will cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence and residual element content, can be predicted using Regulatory Guide 1.99, Revision 2, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials." The heatup and cooldown limit curves Figures 3.4-2(a) and 3.4-2(b) includes predicted adjustments for this shift in RT_{NDT} at the end of the applicable service period, as well as adjustments for possible errors in the pressure and temperature sensing instruments.

The controlling material for all three Palo Verde Units is the Palo Verde Unit 1 shell plates M-6701-2 and M-6701-3. In all three Palo Verde Units, the welds always showed lower reference temperatures than the base metal, i.e. lower initial RT_{NDT} and lower ART after irradiation. Therefore, only the base metal and not the weldments is predicted to be controlling during design life. The limiting ART values based upon the Palo Verde Unit 1 intermediate shell plates are 102°F and 90°F for the 1/4T and the 3/4T locations for 10 years of operation, and 116°F and 103°F for the 1/4T and 3/4T locations for 40 years of operation.

Note that two different sets of chemical content data were available for the reactor vessel beltline welds; one set being the weld metal certification tests, and the other being vessel weld seam sample analyses.

The former set tended to be more limiting (i.e., produced a slightly higher chemistry factor) and, therefore, was used in calculations of adjusted reference temperature. Even with the more conservative weld chemistry factors, the plates remained as the controlling vessel beltline materials in each of the three Palo Verde units.

The actual shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating, in accordance with ASTM E185-73 and Appendix H of 10 CFR 50, reactor vessel material irradiation surveillance specimens installed near the inside wall of the reactor vessel in the core area. Since the neutron spectra at the irradiation samples and vessel inside radius are essentially identical, the measured transition shift for a sample can be applied with confidence to the adjacent section of the reactor vessel. The heatup and cooldown curve must be recalculated when the delta RT_{NDT} determined from the surveillance capsule is different from the calculated delta RT_{NDT} for the equivalent capsule radiation exposure.

The pressure-temperature limit lines shown on Figures 3.4-2a and 3.4-2b for criticality and for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR Part 50. The reactor vessel material irradiation surveillance specimens are removed and examined to determine changes in material properties. The results of these examinations shall be used to update Figures 3.4-2a and 3.4-2b based on the greater of the following:

(1) the actual shift in reference temperature for plate F-377-1 and weld 101-142 as determined by impact testing, or

(2) the predicted shift in reference temperature for the limiting weld and plate as determined by RG 1.99, Rev. 2, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials."

The maximum RT_{NDT} for all Reactor Coolant System pressure-retaining materials, with the exception of the reactor pressure vessel, has been determined to be 40°F. The Lowest Service Temperature limit is based upon this RT_{NDT} since Article NB-2332 (Summer Addenda of 1972) of Section III of the ASME Boiler and Pressure Vessel Code requires the Lowest Service Temperature to be RT_{NDT} + 100°F for piping, pumps, and valves. Below this temperature, the system pressure must be limited to a maximum of 20% of the system's hydrostatic test pressure of 3125 psia. However, based upon the 10 CFR 50 Appendix G analysis, the isothermal condition for the reactor vessel is more restrictive than the Lowest Service Temperature line.
The number of reactor vessel irradiation surveillance capsules and the frequencies for removing and testing these capsules are provided in Table 4.4-5 to assure compliance with the requirements of Appendix H to 10 CFR 50.

The limitations imposed on the pressurizer heatup and cooldown rates and spray water temperature differential are provided to assure that the pressurizer is operated within the design criteria assumed for the fatigue analysis performed in accordance with the ASME Code Requirements.

The OPERABILITY of two shutdown cooling suction line relief valves, one located in each shutdown cooling suction line, while maintaining the limits imposed on the RCS heatup and cooldown rates, ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR 50 when one or more of the RCS cold legs are less than or equal to 214°F during cooldown, 291°F during heatup. Either one of the two SCS suction line relief valves provides relieving capability to protect the RCS from overpressurization when the transient is limited to either (1) the start of an idle RCP with the secondary water temperature of the steam generator less than or equal to 100°F above the RCS cold leg temperatures or (2) the inadvertent safety injection actuation with two HPSI pumps injecting into a water solid RCS with full charging capacity and with letdown isolated. These events are the most limiting energy and mass addition transients, respectively, when the RCS is at low temperatures.

The limitations imposed on the RCS heatup and cooldown rates are provided to assure low temperature overpressure protection (LTOP) with the two shutdown cooling suction line relief valves operable. At low temperatures with the relief valves aligned to the RCS, it is necessary to restrict heatup and cooldown rates to assure that P-T limits are not exceeded. The primary objective of the LTOP systems is to preclude violation of applicable Technical Specification P-T limits during startup and shutdown conditions. These P-T limits are usually applicable to a finite time period such as one cycle, 5 EFPY, etc. and are based upon the irradiation damage prediction by the end of the period. Accordingly, each time P-T limits change, the LTOP system needs to be re-analyzed and modified, if necessary, to continue its function.

A typical LTOP system includes pressure relieving devices and a number of administrative and operational controls. Each of the Palo Verde Units has a similar LTOP system that includes two Shutdown Cooling System suction line relief valves for transient mitigation. Each relief valve has an opening setpoint of 467 psig which, in combination with certain other limiting conditions for operation contained in Technical Specifications, comprises the LTOP system.

Previously, the LTOP enable temperatures during heatup and cooldown have been determined at the intersections between a horizontal line corresponding to the safety valve setpoint (2500 psia) and the most limiting P-T limit curves for heatup and cooldown, respectively. Note that the enable temperature generally identifies the upper temperature limit below which the LTOP system has to be operable.

In this analysis, the LTOP enable temperatures were determined in accordance with a definition contained in the latest revision of the Standard Review Plan 5.2.2. According to SRP 5.2.2 the LTOP enable temperature is "the water temperature corresponding to a metal temperature of at least RT_{NDT} + 90°F at the beltline location (1/4T or 3/4T) that is controlling in the Appendix G limit calculations." The heatup and cooldown rate limitations assure the limits of Appendix G to 10 CFR 50 will not be

exceeded with overpressure protection provided by the primary safety valves.

B. Purpose of the Technical Specification

Reactor vessel pressure-temperature limitations and Low Temperature Overpressure Protection requirements for the Palo Verde Nuclear Generating Station are calculated to meet the regulations of 10 CFR Part 50 Appendix A, Design Criterion 14 and Design Criteria 31. These design criteria require that the reactor coolant pressure boundary be designed, fabricated, erected, and tested in order to have an extremely low probability of abnormal leakage, of rapid failure, and of gross rupture. The criteria also require that the reactor coolant pressure boundary be designed with sufficient margin to assure that when stressed under operating, maintenance, and testing the boundary behave in a non-brittle manner and the probability of a rapidly propagating fracture is minimized.

The pressure-temperature limits are developed using the requirements of 10 CFR 50 Appendix G. This appendix describes the requirements for developing the pressure-temperature limits and provides the general basis for these limitations. The margins of safety against fracture provided by the pressure-temperature limits using the requirements of 10 CFR Part 50 Appendix G are equivalent to those recommended in the ASME Boiler and Pressure Vessel Code Section III, Appendix G, "Protection Against Nonductile Failure." The general guidance provided in those procedures has been utilized to develop the Palo Verde pressure-temperature limits with the requisite margins of safety for heatup and cooldown conditions.

The limitations imposed on the RCS heatup and cooldown rates are provided to assure low temperature overpressure protection (LTOP) with the two shutdown cooling suction line relief valves operable. At low temperatures with the relief valves aligned to the RCS, it is necessary to restrict heatup and cooldown rates to assure that P-T limits are not exceeded. The primary objective of the LTOP systems is to preclude violation of applicable Technical Specification P-T limits during startup and shutdown conditions. These P-T limits are usually applicable to a finite time period such as one cycle, 5 EFPY, etc. and are based upon the irradiation damage prediction by the end of the period.

C. Need for the Technical Specification Change

This amendment request is required by Generic Letter 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and its Impact on Plant Operations." The Generic Letter required reanalysis of the reactor vessel P-T limits using the methodology of Revision 2 to Regulatory Guide 1.99, "Radiation Embrittlement of Reactor Vessel Materials." This amendment submits the results of that analysis in the form of revised Technical Specifications for LTOP enable setpoints and temperatures and Reactor Coolant System (RCS) P-T limit curves.

D. Basis for No Significant Hazards Consideration

1. 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 the amendment request follows:

Standard 1: Involve a significant increase in the probability or consequences of an accident previously evaluated.

The probability and consequences of a Loss of Coolant Accident previously analyzed in the Updated Final Safety Analysis Report (UFSAR) Chapter 6 are not increased. Revision 2 to Regulatory Guide 1.99 provides a more accurate methodology for predicting neutron radiation effects on reactor vessel materials. The corresponding reactor vessel P-T limitations and LTOP requirements have been calculated to meet the requirements of 10 CFR 50 Appendix A, Design Criteria 14 and 31. These criteria require that the reactor coolant pressure boundary be designed to have an extremely low probability of abnormal leakage, rapid failure, and gross rupture.

These criteria also require that the reactor coolant pressure boundary be designed with sufficient margin to assure that when stressed under operating, maintenance, and testing the boundary behaves in a non-brittle manner and the probability of a rapidly propagating fracture is minimized. Linear elastic fracture mechanics analyses as described in the Revision 2 analysis performed by Combustion Engineering for Arizona Public Service demonstrate margins of safety equivalent to those recommended in the ASME Code Section III, Appendix G, "Protection Against Nonductile Failure."

Standard 2: Create the possibility of a new or different kind of accident from any accident previously evaluated.

New P-T limits and LTOP requirements have been calculated as a result of the new methodology for determining the effects of neutron radiation embrittlement on reactor vessel material as described in Regulatory Guide 1.99, Revision 2, Linear elastic fracture mechanics analyses, per ASME Section III Appendix G have been performed thereby ensuring the integrity of the reactor coolant pressure boundary is maintained. Therefore no new accidents or malfunctions are introduced.

Standard 3: Involve a significant reduction in a margin of safety.

The margin of safety is increased using Revision 2 to Regulatory Guide 1.99. In general this revision results in more conservative P-T and LTOP limits. The safety margin recommended in ASME Section III, Appendix G, and the requirements of Appendix G of 10 CFR 50 are maintained with the changes proposed in this amendment.

E. Safety Analysis of the Proposed Amendment

This amendment changes the P-T and LTOP enable temperatures to be consistent with the requirements of Revision 2 of Regulatory Guide 1.99, "Radiation Embrittlement of Reactor Vessel Materials." This new methodology results in different P-T curves and LTOP enable temperatures which are presented in this amendment. The use of updated methodology to predict neutron radiation embrittlement of reactor vessel materials does not change the margin of safety as provided for in 10 CFR 50 Appendix G and the ASME Code Section III, Appendix G. The LTOP requirements were established based upon the guidance provided in USNRC Standard Review Plan (SRP), Revision 2, Section 5.2.2, and the maximum transient pressure of 488 psia that resulted from the existing mass and energy addition transient analysis. The refinement to the methodology provided by Revision 2 to Regulatory Guide 1.99 only enhances the safety of continued operation by predicting a more accurate effect of neutron radiation embrittlement and thereby decreasing the probability of a non-ductile fracture of the reactor vessel.

F. Environmental Impact Consideration Determination

The proposed change request does not involve an unreviewed environmental question because operation of PVNGS Units 1, 2, and 3 in accordance with this change, would not:

1. Result in a significant increase in any adverse environmental impact previously evaluated in the Final Environmental Statement (FES) as modified by the staff's testimony to the Atomic Safety and Licensing Board; or

2. Result in a significant change in effluents or power levels; or

3. Result in matters not previously reviewed in the licensing basis for PVNGS which may have a significant environmental impact.

As discussed in the Section D of this amendment no reduction in safety and no new accidents are introduced by this change. This amendment does not affect effluents or power levels and has no environmental impact.

G. Marked-up Technical Specification Change Pages

XIX	3/4 4-28a	B 3/4 4-6
XXI	3/4 4-29	B 3/4 4-7
3/4 4-3	3/4 4-29a	B 3/4 4-10
3/4 4-5	3/4 4-32	B 3/4 4-11
3/4 4-28	3/4 4-33	

INSERTS FOR CHANGE PAGES

INSERT 1 DELETE REFERENCE TO FIGURES 3.4-2, B 3/4.4-1, AND ADD THE FOLLOWING ON PAGE XIX

INSERT 2 ADD THE FOLLOWING ON PAGE XXI

INSERT 3 CHANGE LTOP ENABLE TEMPERATURES AS MARKED ON PAGE 3/4 4-3 AND ADD THE FOLLOWING NOTE

Reactor Coolant Pump operation is limited to no more than, 2 Reactor Coolant Pumps with RCS cold leg temperature less than or equal to 200°F, 3 Reactor Coolant Pumps with RCS cold leg temperature greater than 200°F but less than or equal to 500°F.

INSERT 4 CHANGE LTOP ENABLE TEMPERATURES AS MARKED ON PAGE 3/4 4-5 AND ADD THE FOLLOWING NOTE

Reactor Coolant Pump operation is limited to no more than, 2 Reactor Coolant Pumps with RCS cold leg temperature less than or equal to 200°F, 3 Reactor Coolant Pumps with RCS cold leg temperature greater than 200°F but less than or equal to 500°F.

INSERT 5 DELETE SUBPARAGRAPHS a-c AND ADD THE FOLLOWING ON PAGE 3/4 4-28

Figures 3.4-2a or 3.4-2b during heatup, cooldown, criticality, and inservice leak and hydrostatic testing with:

a. Maximum heatup and cooldown rates as specified in Table 3.4-3.

b. A maximum temperature change of 10°F in any 1-hour period during inservice hydrostatic testing operations.

INSERT 6 NEW PAGE 3/4 4-28a

INSERT 7 DELETE P-T CURVE 3/4 3.4-2 AND ADD NEW P-T LIMIT CURVE 3/4 3.4-2a (LOCATED BEHIND CURRENT P-T CURVE) ON PAGE 3/4 4-29

INSERT 8 NEW PAGE 3/4 4-29a

CHANGE LTOP ENABLE TEMPERATURES AS MARKED ON COPY OF PAGES 3/4 4-32 AND 3/4 4-33 AND DELETE * AND NOTE AT BOTTOM OF PAGE 3/4 4-32

INSERT 9 REPLACE CURRENT BASES WITH THE FOLLOWING ON PAGES B 3/4 4-6, B 3/4 4-7, and B 3/4 4-11

The various categories of load cycles used for design purposes are provided in Chapters 3 and 5 of the FSAR. During startup and shutdown, the rates of temperature and pressure changes are limited so as not to exceed the limit lines of Figures 3.4-2a and 3.4-2b. This ensures that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation.

Reactor vessel pressure-temperature limitations and Low Temperature Overpressure Protection requirements for the Palo Verde Nuclear Generating Station are calculated to meet the regulations of 10 CFR Part 50 Appendix A, Design Criterion 14 and Design Criteria 31. These design criteria require that the reactor coolant pressure boundary be designed, fabricated, erected, and tested in order to have an extremely low probability of abnormal leakage, of rapid failure, and of gross rupture. The criteria also require that the reactor coolant pressure boundary be designed with sufficient margin to assure that when stressed under operation, maintenance, and testing the boundary; behaves in a non-brittle manner and the probability of a rapidly propagating fracture is minimized.

The pressure-temperature limits are developed using the requirements of 10 CFR 50 Appendix G. This appendix describes the requirements for developing the pressure-temperature limits and provides the general basis for these limitations. The margins of safety against fracture provided by the pressure-temperature limits using the requirements of 10 CFR Part 50 Appendix G are equivalent to those recommended in the ASME Boiler and Pressure Vessel Code Section III, Appendix G, "Protection Against Nonductile Failure." The general guidance provided in those procedures has been utilized to develop the Palo Verde pressure-temperature limits with the requisite margins of safety for heatup and cooldown conditions.

The pressure-temperature limits account for the temperature differential between the reactor vessel base metal and the reactor coolant bulk fluid temperature. Correction for elevation and RCS flow induced pressure differences between the reactor vessel beltline and pressurizer, are included in the development of the pressure-temperature limits as are instrumentation uncertainties for pressure and temperature measurement. Consequently the P-T limits are provided on coordinates of indicated pressurizer pressure versus indicated RCS temperature.

The pressure correction factors are based upon the differential pressure due to the elevation difference between the reactor vessel wall adjacent to the active core region and the pressurizer pressure instrument nozzle. This term of the pressure correction factor is equal to 29.62 psi. The pressure correction factors are also based upon flow induced pressure drops across the reactor core through the hot leg pipe up to the surge line nozzle. This term of the pressure correction factor has two values which are

dependent upon the Reactor Coolant Pump (RCP) combination utilized during operation. At temperatures of $T_c < 200^{\circ}F$, the flow induced pressure drop is based upon RCS flow rates resulting from two operating RCPs and is equal to 55.02 psi using post-core hot functional test data. At temperatures of $T_c \geq 200^{\circ}F$, the flow induced pressure drop is based upon the RCS flow rates resulting from three operating RCPs and is equal to 64.39 psi using post-core hot functional test data. The pressure correction factors also account for pressurizer pressure measurement uncertainty.

The Reactor Pressure Vessel beltline pressure-temperature limits are based upon the irradiation damage prediction method of Regulatory Guide 1.99 Revision 02. This methodology has been used to calculate the limiting material Adjusted Reference Temperatures (ART) for Palo Verde Units 1, 2, and 3. The adjusted reference temperatures of reactor vessel beltline materials for Palo Verde Units 1, 2, and 3 have been calculated at the 1/4T and 3/4T locations after 10 and 40 calendar years operation. By comparing the ART data for each material, the controlling materials for all three Palo Verde units, have been determined.

The analytical procedure for developing reactor vessel pressure-temperature limits utilizes the methods of Linear Elastic Fracture Mechanics (LEFM) found in the ASME Boiler and Pressure Vessel Code Section III, Appendix G in accordance with the requirements of 10 CFR Part 50 Appendix G. For these analyses, the Mode I (opening mode) stress intensity factors are used for the solution basis. The general method utilizes LEFM procedures. LEFM relates the size of a flaw with the allowable loading which precludes crack initiation. This relation is based upon a mathematical stress analysis of the beltline material fracture toughness properties as prescribed in Appendix G to Section III of the ASME code.

The reactor vessel materials have been tested to determine their initial RT_{NDT}; the results of these tests are shown in Table B 3/4.4-1. Reactor operation and resultant fast neutron (E greater than 1 MeV) irradiation will cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence and residual element content, can be predicted using Regulatory Guide 1.99, Revision 2, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials." The heatup and cooldown limit curves Figures 3.4-2(a) and 3.4-2(b) includes predicted adjustments for this shift in RT_{NDT} at the end of the applicable service period, as well as adjustments for possible errors in the pressure and temperature sensing instruments.

The controlling material for all three Palo Verde Units is the Palo Verde Unit 1 shell plates M-6701-2 and M-6701-3. In all three Palo Verde Units, the welds always showed lower reference temperatures than the base metal, i.e. lower initial RT_{NDT} and lower ART after irradiation. Therefore, only the base metal and not the weldments is predicted to be controlling during design life. The limiting ART values based upon the Palo Verde Unit 1 intermediate shell plates are $102^{\circ}F$ and $90^{\circ}F$ for the 1/4T and the 3/4T locations for

10 years of operation, and 116°F and 103°F for the 1/4T and 3/4T locations for 40 years of operation.

Note that two different sets of chemical content data were available for the reactor vessel beltline welds; one set being the weld metal certification tests, and the other being vessel weld seam sample analyses. The former set tended to be more limiting (i.e., produced a slightly higher chemistry factor) and, therefore, was used in calculations of adjusted reference temperature. Even with the more conservative weld chemistry factors, the plates remained as the controlling vessel beltline materials in each of the three Palo Verde units.

The actual shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating, in accordance with ASTM E185-73 and Appendix H of 10 CFR 50, reactor vessel material irradiation surveillance specimens installed near the inside wall of the reactor vessel in the core area. Since the neutron spectra at the irradiation samples and vessel inside radius are essentially identical, the measured transition shift for a sample can be applied with confidence to the adjacent section of the reactor vessel. The heatup and cooldown curve must be recalculated when the delta RT_{NDT} determined from the surveillance capsule is different from the calculated delta RT_{NDT} for the equivalent capsule radiation exposure.

The pressure-temperature limit lines shown on Figures 3.4-2a and 3.4-2b for criticality and for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR Part 50. The reactor vessel material irradiation surveillance specimens are removed and examined to determine changes in material properties. The results of these examinations shall be used to update Figures 3.4-2a and 3.4-2b based on the greater of the following:

(1) the actual shift in reference temperature for plate F-377-1 and weld 101-142 as determined by impact testing, or

(2) the predicted shift in reference temperature for the limiting weld and plate as determined by RG 1.99, Rev. 2, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials."

The maximum RT_{NDT} for all Reactor Coolant System pressure-retaining materials, with the exception of the reactor pressure vessel, has been determined to be 40°F. The Lowest Service Temperature limit is based upon this RT_{NDT} since Article NB-2332 (Summer Addenda of 1972) of Section III of the ASME Boiler and Pressure Vessel Code requires the Lowest Service Temperature to be RT_{NDT} + 100°F for piping, pumps, and valves. Below this temperature, the system pressure must be limited to a maximum of 20% of the system's hydrostatic test pressure of 3125 psia. However, based upon the 10 CFR 50 Appendix G analysis, the isothermal condition for the reactor vessel is more restrictive than the Lowest Service

Temperature line.

The number of reactor vessel irradiation surveillance capsules and the frequencies for removing and testing these capsules are provided in Table 4.4-5 to assure compliance with the requirements of Appendix H to 10 CFR 50.

The limitations imposed on the pressurizer heatup and cooldown rates and spray water temperature differential are provided to assure that the pressurizer is operated within the design criteria assumed for the fatigue analysis performed in accordance with the ASME Code Requirements.

The OPERABILITY of two shutdown cooling suction line relief valves, one located in each shutdown cooling suction line, while maintaining the limits imposed on the RCS heatup and cooldown rates, ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR 50 when one or more of the RCS cold legs are less than or equal to 214°F during cooldown, 291°F during heatup. Either one of the two SCS suction line relief valves provides relieving capability to protect the RCS from overpressurization when the transient is limited to either (1) the start of an idle RCP with the secondary water temperature of the steam generator less than or equal to 100°F above the RCS cold leg temperatures or (2) the inadvertent safety injection actuation with two HPSI pumps injecting into a water solid RCS with full charging capacity and with letdown isolated. These events are the most limiting energy and mass addition transients, respectively, when the RCS is at low temperatures.

The limitations imposed on the RCS heatup and cooldown rates are provided to assure low temperature overpressure protection (LTOP) with the two shutdown cooling suction line relief valves operable. At low temperatures with the relief valves aligned to the RCS, it is necessary to restrict heatup and cooldown rates to assure that P-T limits are not exceeded. The primary objective of the LTOP systems is to preclude violation of applicable Technical Specification P-T limits during startup and shutdown conditions. These P-T limits are usually applicable to a finite time period such as one cycle, 5 EFPY, etc. and are based upon the irradiation damage prediction by the end of the period. Accordingly, each time P-T limits change, the LTOP system needs to be re-analyzed and modified, if necessary, to continue its function.

A typical LTOP system includes pressure relieving devices and a number of administrative and operational controls. Each of the Palo Verde Units has a similar LTOP system that includes two Shutdown Cooling System suction line relief valves for transient mitigation. Each relief valve has an opening setpoint of 467 psig

which, in combination with certain other limiting conditions for operation contained in Technical Specifications, comprises the LTOP system.

Previously, the LTOP enable temperatures during heatup and cooldown have been determined at the intersections between a horizontal line corresponding to the safety valve setpoint (2500 psia) and the most limiting P-T limit curves for heatup and cooldown, respectively. Note that the enable temperature generally identifies the upper temperature limit below which the LTOP system has to be operable.

In this analysis, the LTOP enable temperatures were determined in accordance with a definition contained in the latest revision of the Standard Review Plan 5.2.2. According to SRP 5.2.2 the LTOP enable temperature is "the water temperature corresponding to a metal temperature of at least RT_{NDT} + 90°F at the beltline location (1/4T or 3/4T) that is controlling in the Appendix G limit calculations." The heatup and cooldown rate limitations assure the limits of Appendix G to 10 CFR 50 will not be exceeded with overpressure protection provided by the primary safety valves. The various categories of load cycles used for design purposes are provided in Chapters 3 and 5 of the FSAR. During startup and shutdown, the rates of temperature and pressure changes are limited so as not to exceed the limit lines of Figures 3.4-2a and 3.4-2b. This ensures that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation.

<u>DELETE FIGURE B 3/4.4-1 ON PAGE B 3/4 4-10</u>