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

ARIZONA PUBLIC SERVICE COMPANY, ET AL.

DOCKET NO. STN 50-528

PALO VERDE NUCLEAR GENERATING STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 52
License No. NPF-41

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment, dated March 13, 1990 by the Arizona Public Service Company (APS) on behalf of itself and the Salt River Project Agricultural Improvement and Power District, El Paso Electric Company, Southern California Edison Company, Public Service Company of New Mexico, Los Angeles Department of Water and Power, and Southern California Public Power Authority (licensees), complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of Act, and the regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public;

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the enclosure to this license amendment, and paragraph 2.C(2) of Facility Operating License No. NPF-41 is hereby amended to read as follows:

(2) <u>Technical Specifications and Environmental Protection Plan</u>

The Technical Specifications contained in Appendix A, as revised through Amendment No. 52, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated into this license. APS shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective 45 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

John T. Larkins

John T. Larkins, Acting Director
Project Directorate V
Division of Reactor Projects III,
IV, V and Special Projects
Office of Nuclear Reactor Regulation

Enclosure:
Changes to the Technical
Specifications

Date of Issuance: July 25, 1990

ENCLOSURE TO LICENSE AMENDMENT

AMENDMENT NO. 52 TO FACILITY OPERATING LICENSE NO. NPF-41

DOCKET NO. STN 50-528

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain vertical lines indicating the areas of change.

Remove Pages	Insert Pages
XIX	XIX
XXI	XXI
3/4 4-3	3/4 4-3
3/4 4-5	3/4 4-5
3/4 4-28	3/4 4-28
--	3/4 4-28a
3/4 4-29	3/4 4-29
--	3/4 4-29a
3/4 4-32	3/4 4-32
3/4 4-33	3/4 4-33
B 3/4 4-6	B 3/4 4-6
B 3/4 4-7	B 3/4 4-7
B 3/4 4-10	B 3/4 4-10
B 3/4 4-11	B 3/4 4-11 thru B 3/4 4-13

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REACTOR COOLANT SYSTEM

HOT SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.1.3 At least two of the loop(s)/train(s) listed below shall be OPERABLE and at least one reactor coolant and/or shutdown cooling loops shall be in operation*.

a. Reactor Coolant Loop 1 and its associated steam generator and at least one associated reactor coolant pump**,

b. Reactor Coolant Loop 2 and its associated steam generator and at least one associated reactor coolant pump**,

c. Shutdown Cooling Train A,

d. Shutdown Cooling Train B.

APPLICABILITY: MODE 4#.

ACTION:

a. With less than the above required reactor coolant and/or shutdown cooling loops OPERABLE, immediately initiate corrective action to return the required loops to OPERABLE status as soon as possible; if the remaining OPERABLE loop is a shutdown cooling loop, be in COLD SHUTDOWN within 24 hours.

b. With no reactor coolant or shutdown cooling loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required coolant loop to operation.

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*All reactor coolant pumps and shutdown cooling pumps may be deenergized for up to 1 hour provided (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

**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.

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.

#See Special Test Exception 3.10.9.

REACTOR COOLANT SYSTEM

COLD SHUTDOWN - LOOPS FILLED

LIMITING CONDITION FOR OPERATION

3.4.1.4.1 At least one shutdown cooling loop shall be OPERABLE and in operation*, and either:

a. One additional shutdown cooling loop shall be OPERABLE#, or

b. The secondary side water level of at least two steam generators shall be greater than 25% indicated wide range level.

APPLICABILITY: MODE 5 with reactor coolant loops filled##.

ACTION:

a. With less than the above required loops OPERABLE or with less than the required steam generator level, immediately initiate corrective action to return the required loops to OPERABLE status or to restore the required level as soon as possible.

b. With no shutdown cooling loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required shutdown cooling loop to operation.

SURVEILLANCE REQUIREMENTS

4.4.1.4.1.1 The secondary side water level of both steam generators when required shall be determined to be within limits at least once per 12 hours.

4.4.1.4.1.2 At least one shutdown cooling loop shall be determined to be in operation and circulating reactor coolant at a flow rate of greater than or equal to 4000 gpm at least once per 12 hours.

---

*The shutdown cooling pump may be deenergized for up to 1 hour provided (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

#One shutdown cooling loop may be inoperable for up to 2 hours for surveillance testing provided the other shutdown cooling loop is OPERABLE and in operation.

##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.

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.

[illegible]

3/4.4.8 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.8.1 The Reactor Coolant System (except the pressurizer) temperature and pressure shall be limited in accordance with the limit lines shown on 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.

APPLICABILITY: At all times*.

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limit within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operations or be in at least HOT STANDBY within the next 6 hours and reduce the RCS T_{cold} and pressure to less than 210°F and 500 psia, respectively, within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.8.1.1 The Reactor Coolant System temperature and pressure shall be determined to be within the limits at least once per 30 minutes during system heatup, cooldown, and inservice leak and hydrostatic testing operations.

4.4.8.1.2 The reactor vessel material irradiation surveillance specimens shall be removed and examined, to determine changes in material properties, at the intervals required by 10 CFR Part 50 Appendix H in accordance with the schedule in Table 4.4-5. The results of these examinations shall be used to update Figure 3.4-2.

*See Special Test Exception 3.10.5.

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TABLE 3.4-3

Maximum Allowable Heatup and Cooldown Rates

≤8 Effective Full Power Years

Heatup

T_c* (°F)	Rate (°F/HR)
< 128°F	20°F/HR
128° - 180°F	30°F/HR
181° - 230°F	50°F/HR
> 230°F	75°F/HR

Cooldown

T_c* (°F)	Rate (°F/HR)
≤ 93°F	0°F/HR
94° - 114°F	10°F/HR
115° - 148°F	20°F/HR
> 148°F	100°F/HR

8-32 Effective Full Power Years

Heatup

T_c* (°F)	Rate (°F/HR)
< 116°F	10°F/HR
117° - 150°F	20°F/HR
151° - 199°F	30°F/HR
200° - 246°F	50°F/HR
> 246°F	75°F/HR

Cooldown

T_c* (°F)	Rate (°F/HR)
≤ 108°F	0°F/HR
109° - 126°F	10°F/HR
127° - 147°F	20°F/HR
148° - 162°F	40°F/HR
> 162°F	100°F/HR

* Indicated Cold Leg Temperature

FIGURE 3.4-2a
REACTOR COOLANT SYSTEM PRESSURE/TEMPERATURE LIMITATIONS FOR LESS THAN 8 EFFECTIVE FULL POWER YEARS OF OPERATION

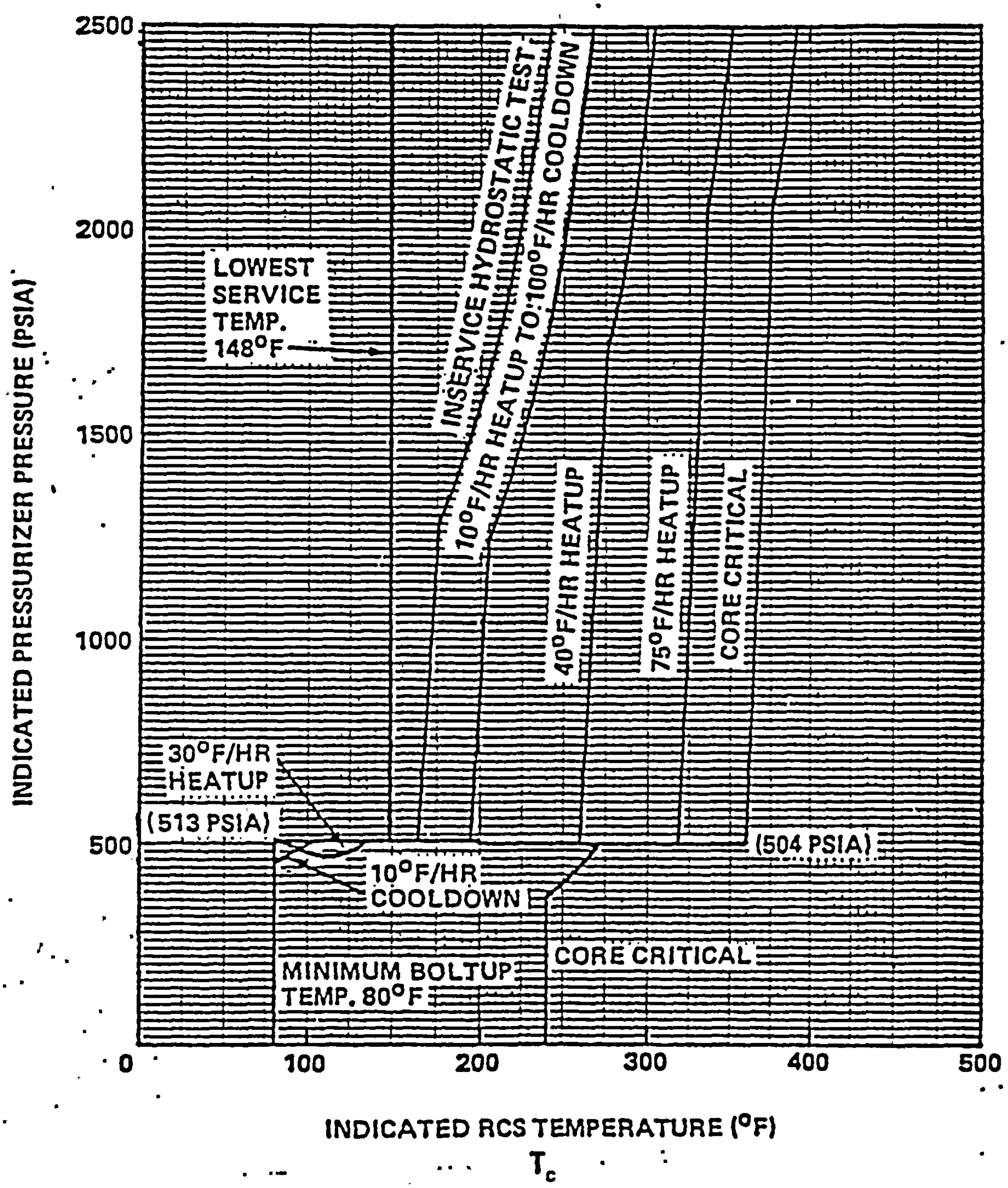

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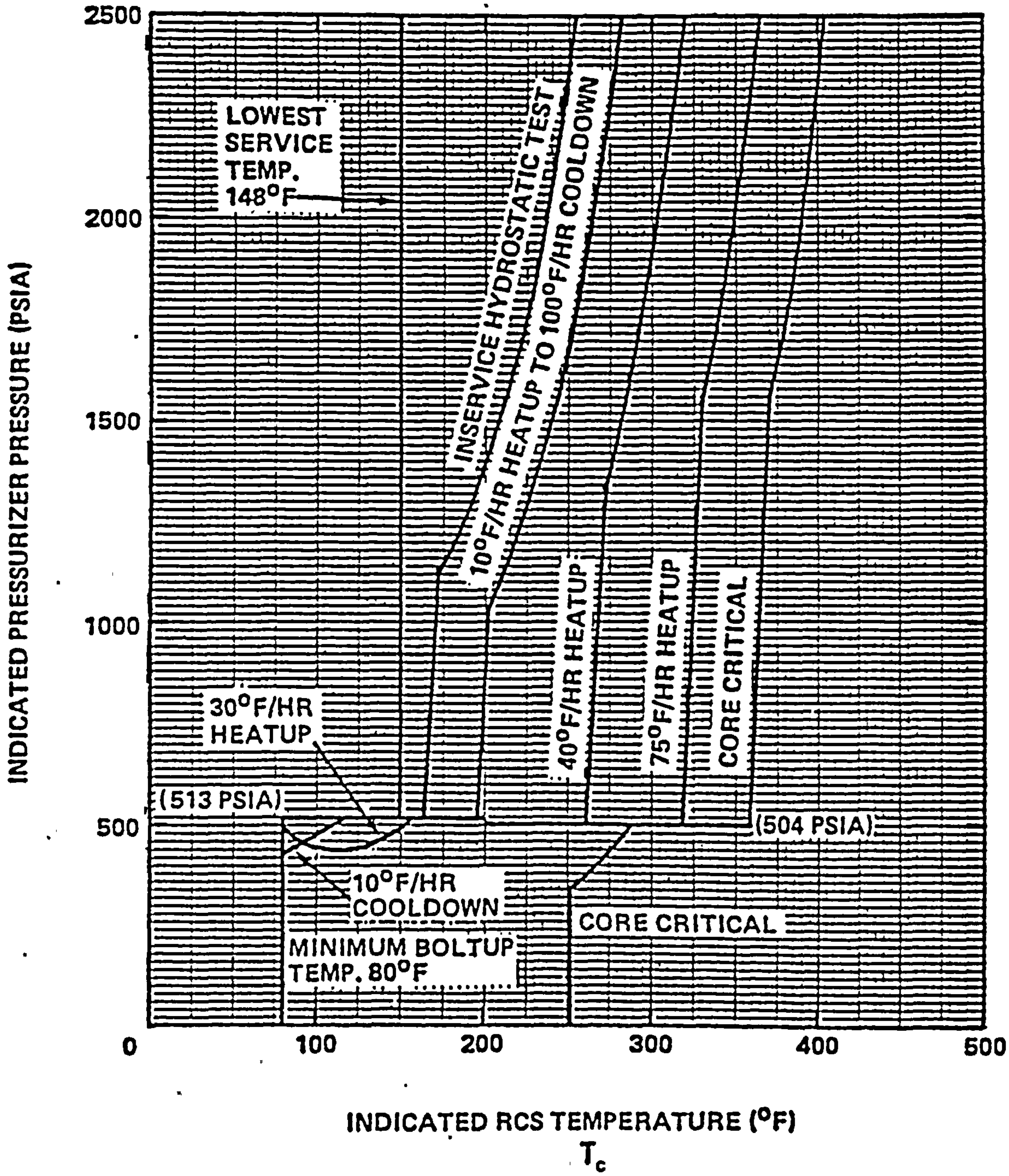
FIGURE 3.4-2B
REACTOR COOLANT SYSTEM PRESSURE/TEMPERATURE LIMITATIONS FOR 8 TO 32 EFFECTIVE FULL POWER YEARS OF OPERATION



LIMITING CONDITION FOR OPERATION

3.4.8.3 Both shutdown cooling system (SCS) suction line relief valves with lift settings of less than or equal to 467 psig shall be OPERABLE and aligned to provide overpressure protection for the Reactor Coolant System.

APPLICABILITY: When the reactor vessel head is installed and the temperature of one or more of the RCS cold legs is less than or equal to:

a. 214°F during cooldown

b. 291°F during heatup

ACTION:

a. With one SCS relief valve inoperable, restore the inoperable valve to OPERABLE status within seven days or reduce T_{cold} to less than 200°F and, depressurize and vent the RCS through a greater than or equal to 16 square inch vent(s) within the next eight hours. Do not start a reactor coolant pump if the steam generator secondary water temperature is greater than 100°F above any RCS cold leg temperature.

b. With both SCS relief valves inoperable, reduce T_{cold} to less than 200°F and, depressurize and vent the RCS through a greater than or equal to 16 square inch vent(s) within eight hours. Do not start a reactor coolant pump if the steam generator secondary water temperature is greater than 100°F above any RCS cold leg temperature.

c. In the event either the SCS suction line relief valves or an RCS vent(s) are used to mitigate an RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the SCS suction line relief valves or RCS vent(s) on the transient and any corrective action necessary to prevent recurrence.

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.8.3.1 Each SCS suction line relief valve shall be verified to be aligned to provide overpressure protection for the RCS once every 8 hours during

a. Cooldown with the RCS temperature less than or equal to 214°F.

b. Heatup with the RCS temperature less than or equal to 291°F.

4.4.8.3.2 The SCS suction line relief valves shall be verified OPERABLE with the required setpoint at least once per 18 months.

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SPECIFIC ACTIVITY (Continued)

Reducing T_{cold} to less than 500°F prevents the release of activity should a steam generator tube rupture, since the saturation pressure of the primary coolant is below the lift pressure of the atmospheric steam relief valves. The surveillance requirements provide adequate assurance that excessive specific activity levels in the primary coolant will be detected in sufficient time to take corrective action. Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomena. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained.

3/4.4.8 PRESSURE/TEMPERATURE LIMITS

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

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PRESSURE/TEMPERATURE LIMITS (Continued)

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 $T_c < 200°F$, 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 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

[illegible]

PRESSURE/TEMPERATURE LIMITS (Continued)

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 pressurė 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 irradation 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 M-6701-2 and M-4311-1 and weld 101-142 as determined by impact testing, or

[illegible]

PRESSURE/TEMPERATURE LIMITS (Continued)

(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 systems'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,

[illegible]

PRESSURE/TEMPERATURE LIMITS (Continued)

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.

3/4.4.9 STRUCTURAL INTEGRITY

The inservice inspection and testing programs for ASME Code Class 1, 2, and 3 components ensure that the structural integrity and operational readiness of these components will be maintained at an acceptable level throughout the life of the plant. These programs are in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50.55a(g) except where specific written relief has been granted by the Commission pursuant to 10 CFR 50.55a (g) (6) (i).

Components of the Reactor Coolant System were designed to provide access to permit inservice inspections in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, 1974 Edition and Addenda through Summer 1975.

3/4.4.10 REACTOR COOLANT SYSTEM VENTS

Reactor Coolant System vents are provided to exhaust noncondensible gases and/or steam from the primary system that could inhibit natural circulation core cooling. The OPERABILITY of at least one Reactor Coolant System vent path from the reactor vessel head ensures the capability exists to perform this function.

The valve redundancy of the Reactor Coolant System vent paths serves to minimize the probability of inadvertent or irreversible actuation while ensuring that a single failure of a vent valve, power supply, or control system does not prevent isolation of the vent path.

The function, capabilities, and testing requirements of the Reactor Coolant System vent systems are consistent with the requirements of Item II.B.1 of NUREG-0737.

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

ARIZONA PUBLIC SERVICE COMPANY, ET AL.

DOCKET NO. STN 50-529

PALO VERDE NUCLEAR GENERATING STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 38
License No. NPF-51

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment, dated March 13, 1990 by the Arizona Public Service Company (APS) on behalf of itself and the Salt River Project Agricultural Improvement and Power District, El Paso Electric Company, Southern California Edison Company, Public Service Company of New Mexico, Los Angeles Department of Water and Power, and Southern California Public Power Authority (licensees), complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's regulations set forth in 10 CFR Part I;

   B. The facility will operate in conformity with the application, the provisions of Act, and the regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public;

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the enclosure to this license amendment, and paragraph 2.C(2) of Facility Operating License No. NPF-51 is hereby amended to read as follows:

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 38, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated into this license. APS shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of 45 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

John T. Larkins

John T. Larkins, Acting Director
Project Directorate V
Division of Reactor Projects III,
IV, V and Special Projects
Office of Nuclear Reactor Regulation

Enclosure:
Changes to the Technical
Specifications

Date of Issuance: July 25, 1990

ENCLOSURE TO LICENSE AMENDMENT

AMENDMENT NO. 38 TO FACILITY OPERATING LICENSE NO. NPF-51

DOCKET NO. STN 50-529

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain vertical lines indicating the areas of change.

Remove Pages	Insert Pages
XIX	XIX
XXI	XXI
3/4 4-3	3/4 4-3
3/4 4-5	3/4 4-5
3/4 4-28	3/4 4-28
--	3/4 4-28a
3/4 4-29	3/4 4-29
--	3/4 4-29a
3/4 4-32	3/4 4-32
3/4 4-33	3/4 4-33
B 3/4 4-6	B 3/4 4-6
B 3/4 4-7	B 3/4 4-7
B 3/4 4-10	B 3/4 4-10
B 3/4 4-11	B 3/4 4-11 thru B 3/4 4-13

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[illegible]

REACTOR COOLANT SYSTEM

HOT SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.1.3 At least two of the loop(s)/train(s) listed below shall be OPERABLE and at least one reactor coolant and/or shutdown cooling loops shall be in operation*.

a. Reactor Coolant Loop 1 and its associated steam generator and at least one associated reactor coolant pump**,

b. Reactor Coolant Loop 2 and its associated steam generator and at least one associated reactor coolant pump**,

c. Shutdown Cooling Train A,

d. Shutdown Cooling Train B.

APPLICABILITY: MODE 4.

ACTION:

a. With less than the above required reactor coolant and/or shutdown cooling loops OPERABLE, immediately initiate corrective action to return the required loops to OPERABLE status as soon as possible; if the remaining OPERABLE loop is a shutdown cooling loop, be in COLD SHUTDOWN within 24 hours.

b. With no reactor coolant or shutdown cooling loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required coolant loop to operation.

---

*All reactor coolant pumps and shutdown cooling pumps may be deenergized for up to 1 hour provided (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

**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.

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.

COLD SHUTDOWN - LOOPS FILLED

LIMITING CONDITION FOR OPERATION

3.4.1.4.1 At least one shutdown cooling loop shall be OPERABLE and in operation*, and either:

a. One additional shutdown cooling loop shall be OPERABLE#, or

b. The secondary side water level of at least two steam generators shall be greater than 25% indicated wide range level.

APPLICABILITY: MODE 5 with reactor coolant loops filled##.

ACTION:

a. With less than the above required loops OPERABLE or with less than the required steam generator level, immediately initiate corrective action to return the required loops to OPERABLE status or to restore the required level as soon as possible.

b. With no shutdown cooling loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required shutdown cooling loop to operation.

SURVEILLANCE REQUIREMENTS

4.4.1.4.1.1 The secondary side water level of both steam generators when required shall be determined to be within limits at least once per 12 hours.

4.4.1.4.1.2 At least one shutdown cooling loop shall be determined to be in operation and circulating reactor coolant at a flow rate of greater than or equal to 4000 gpm at least once per 12 hours.

---

*The shutdown cooling pump may be deenergized for up to 1 hour provided (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

#One shutdown cooling loop may be inoperable for up to 2 hours for surveillance testing provided the other shutdown cooling loop is OPERABLE and in operation.

##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.

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.

[illegible]

3/4.4.8 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.8.1 The Reactor Coolant System (except the pressurizer) temperature and pressure shall be limited in accordance with the limit lines shown on 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.

APPLICABILITY: At all times*.

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limit within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operations or be in at least HOT STANDBY within the next 6 hours and reduce the RCS T_{cold} and pressure to less than 210°F and 500 psia, respectively, within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.8.1.1 The Reactor Coolant System temperature and pressure shall be determined to be within the limits at least once per 30 minutes during system heatup, cooldown, and inservice leak and hydrostatic testing operations.

4.4.8.1.2 The reactor vessel material irradiation surveillance specimens shall be removed and examined, to determine changes in material properties, at the intervals required by 10 CFR Part 50 Appendix H in accordance with the schedule in Table 4.4-5. The results of these examinations shall be used to update Figure 3.4-2.

*See Special Test Exception 3.10.5.

TABLE 3.4-3

Maximum Allowable Heatup and Cooldown Rates

≤8 Effective Full Power Years

Heatup

T_c* (°F)	Rate (°F/HR)
< 128°F	20°F/HR
128° - 180°F	30°F/HR
181° - 230°F	50°F/HR
> 230°F	75°F/HR

Cooldown

T_c* (°F)	Rate (°F/HR)
≤ 93°F	0°F/HR
94° - 114°F	10°F/HR
115° - 148°F	20°F/HR
> 148°F	100°F/HR

8-32 Effective Full Power Years

Heatup

T_c* (°F)	Rate (°F/HR)
< 116°F	10°F/HR
117° - 150°F	20°F/HR
151° - 199°F	30°F/HR
200° - 246°F	50°F/HR
> 246°F	75°F/HR

Cooldown

T_c* (°F)	Rate (°F/HR)
≤ 108°F	0°F/HR
109° - 126°F	10°F/HR
127° - 147°F	20°F/HR
148° - 162°F	40°F/HR
> 162°F	100°F/HR

* Indicated Cold Leg Temperature

FIGURE 3.4-2a
REACTOR COOLANT SYSTEM PRESSURE/TEMPERATURE LIMITATIONS FOR LESS THAN 8 EFFECTIVE FULL POWER YEARS OF OPERATION

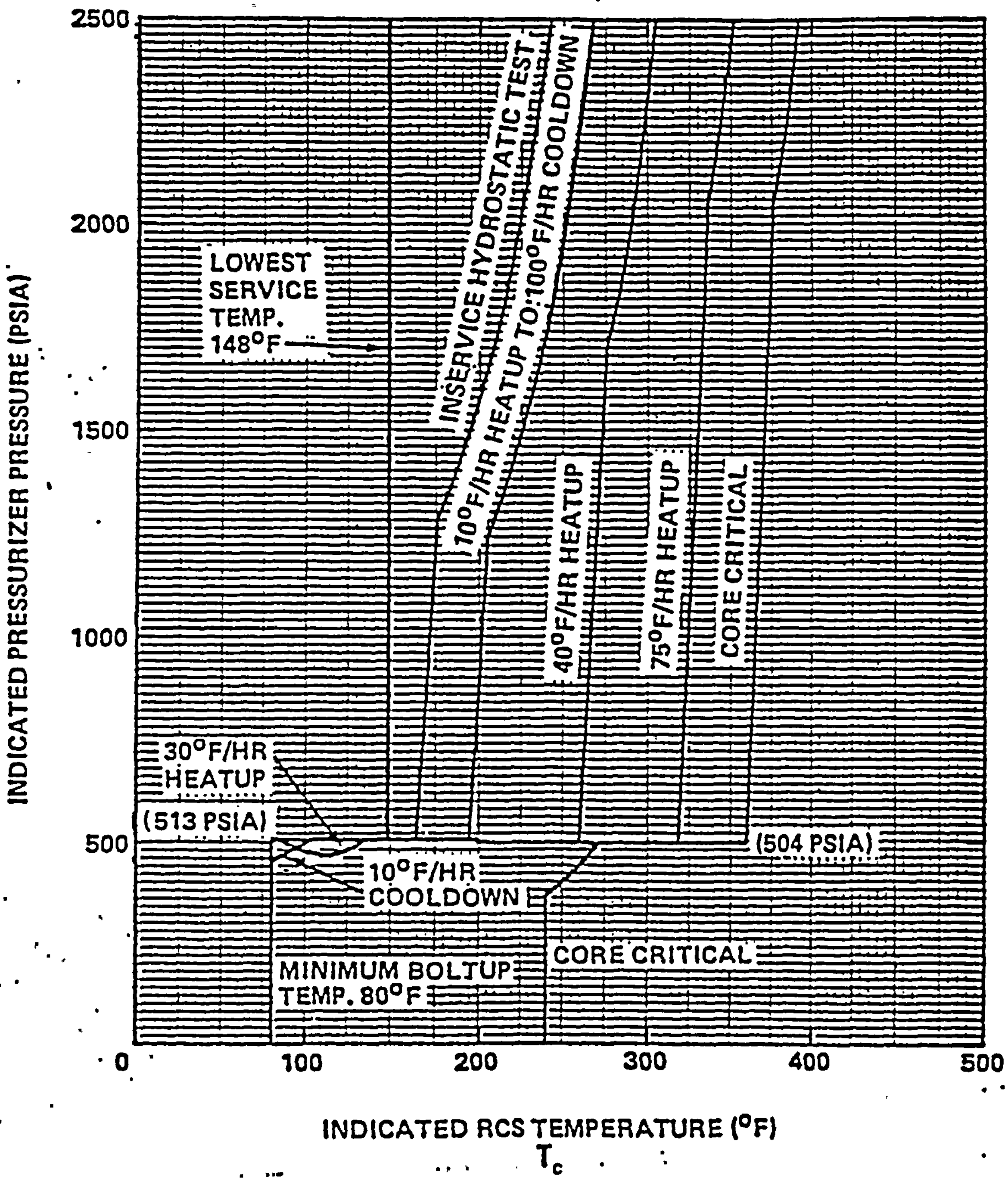
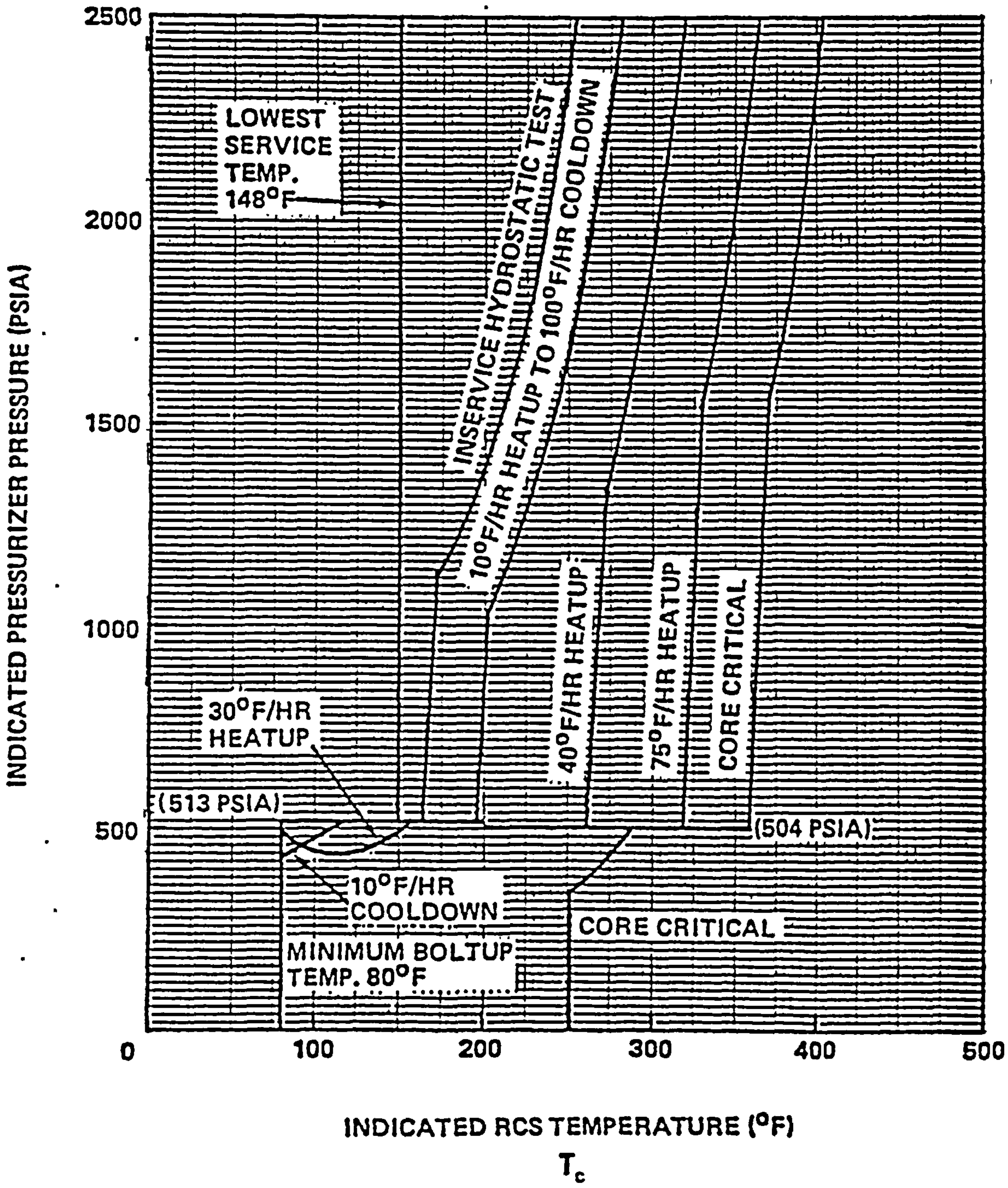


FIGURE 3.4-2b
REACTOR COOLANT SYSTEM PRESSURE/TEMPERATURE LIMITATIONS FOR 8 TO 32 EFFECTIVE FULL POWER YEARS OF OPERATION



REACTOR COOLANT SYSTEM

OVERPRESSURE PROTECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.8.3 Both shutdown cooling system (SCS) suction line relief valves with lift settings of less than or equal to 467 psig shall be OPERABLE and aligned to provide overpressure protection for the Reactor Coolant System.

APPLICABILITY: When the reactor vessel head is installed and the temperature of one or more of the RCS cold legs is less than or equal to:

a. 214°F during cooldown

b. 291°F during heatup

ACTION:

a. With one SCS relief valve inoperable, restore the inoperable valve to OPERABLE status within seven days or reduce T_{cold} to less than 200°F and, depressurize and vent the RCS through a greater than or equal to 16 square inch vent(s) within the next eight hours. Do not start a reactor coolant pump if the steam generator secondary water temperature is greater than 100°F above any RCS cold leg temperature.

b. With both SCS relief valves inoperable, reduce T_{cold} to less than 200°F and, depressurize and vent the RCS through a greater than or equal to 16 square inch vent(s) within eight hours. Do not start a reactor coolant pump if the steam generator secondary water temperature is greater than 100°F above any RCS cold leg temperature.

c. In the event either the SCS suction line relief valves or an RCS vent(s) are used to mitigate an RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the SCS suction line relief valves or RCS vent(s) on the transient and any corrective action necessary to prevent recurrence.

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.8.3.1 Each SCS suction line relief valve shall be verified to be aligned to provide overpressure protection for the RCS once every 8 hours during

a. Cooldown with the RCS temperature less than or equal to 214°F.
b. Heatup with the RCS temperature less than or equal to 291°F.

4.4.8.3.2 The SCS suction line relief valves shall be verified OPERABLE with the required setpoint at least once per 18 months.

[illegible]

SPECIFIC ACTIVITY (Continued)

Reducing T_{cold} to less than 500°F prevents the release of activity should a steam generator tube rupture since the saturation pressure of the primary coolant is below the lift pressure of the atmospheric steam relief valves. The surveillance requirements provide adequate assurance that excessive specific activity levels in the primary coolant will be detected in sufficient time to take corrective action. Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomena. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained.

3/4.4.8 PRESSURE/TEMPERATURE LIMITS

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

[illegible]

PRESSURE/TEMPERATURE LIMITS (Continued)

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 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 postcore 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

[illegible]

PRESSURE/TEMPERATURE LIMITS (Continued)

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-773-1 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."

[illegible]

PRESSURE/TEMPERATURE LIMITS (Continued)

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 Part 50 when one or more of the RCS cold legs are less than or equal to 214°F during cooldown and 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 the 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

PRESSURE/TEMPERATURE LIMITS (Continued)

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 Chapter 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.

3/4.4.9 STRUCTURAL INTEGRITY

The inservice inspection and testing programs for ASME Code Class 1, 2, and 3 components ensure that the structural integrity and operational readiness of these components will be maintained at an acceptable level throughout the life of the plant. These programs are in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50.55a(g) except where specific written relief has been granted by the Commission pursuant to 10 CFR 50.55a (g) (6) (i).

Components of the Reactor Coolant System were designed to provide access to permit inservice inspections in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, 1974 Edition and Addenda through Summer 1975.

3/4.4.10 REACTOR COOLANT SYSTEM VENTS

Reactor Coolant System vents are provided to exhaust noncondensible gases and/or steam from the primary system that could inhibit natural circulation core cooling. The OPERABILITY of at least one Reactor Coolant System vent path from the reactor vessel head ensures the capability exists to perform this function.

The valve redundancy of the Reactor Coolant System vent paths serves to minimize the probability of inadvertent or irreversible actuation while ensuring that a single failure of a vent valve, power supply, or control system does not prevent isolation of the vent path.

The function, capabilities, and testing requirements of the Reactor Coolant System vent systems are consistent with the requirements of Item II.B.1 of NUREG-0737.

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

ARIZONA PUBLIC SERVICE COMPANY, ET AL.

DOCKET NO. STN 50-530

PALO VERDE NUCLEAR GENERATING STATION, UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 24
License No. NPF-74

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment, dated March 13, 1990 by the Arizona Public Service Company (APS) on behalf of itself and the Salt River Project Agricultural Improvement and Power District, El Paso Electric Company, Southern California Edison Company, Public Service Company of New Mexico, Los Angeles Department of Water and Power, and Southern California Public Power Authority (licensees), complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of Act, and the regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public;

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the enclosure to this license amendment, and paragraph 2.C(2) of Facility Operating License No. NPF-74 is hereby amended to read as follows:

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 24 , and the Environmental Protection Plan contained in Appendix B, are hereby incorporated into this license. APS shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of 45 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

John T. Larkins

John T. Larkins, Acting Director
Project Directorate V
Division of Reactor Projects III,
IV, V and Special Projects
Office of Nuclear Reactor Regulation

Enclosure:
Changes to the Technical
Specifications

Date of Issuance: July 25, 1990

ENCLOSURE TO LICENSE AMENDMENT

AMENDMENT NO. 24 TO FACILITY OPERATING LICENSE NO. NPF-74

DOCKET NO. STN 50-530

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain vertical lines indicating the areas of change.

Remove Pages	Insert Pages
XIX	XIX
XXI	XXI
3/4 4-3	3/4 4-3
3/4 4-5	3/4 4-5
3/4 4-28	3/4 4-28
--	3/4 4-28a
3/4 4-29	3/4 4-29
--	3/4 4-29a
3/4 4-32	3/4 4-32
3/4 4-33	3/4 4-33
B 3/4 4-6	B 3/4 4-6
B 3/4 4-7	B 3/4 4-7
B 3/4 4-10	B 3/4 4-10
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REACTOR COOLANT SYSTEM

HOT SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.1.3 At least two of the loop(s)/train(s) listed below shall be OPERABLE and at least one reactor coolant and/or shutdown cooling loops shall be in operation*.

a. Reactor Coolant Loop 1 and its associated steam generator and at least one associated reactor coolant pump**,

b. Reactor Coolant Loop 2 and its associated steam generator and at least one associated reactor coolant pump**,

c. Shutdown Cooling Train A,

d. Shutdown Cooling Train B.

APPLICABILITY: MODE 4.

ACTION:

a. With less than the above required reactor coolant and/or shutdown cooling loops OPERABLE, immediately initiate corrective action to return the required loops to OPERABLE status as soon as possible; if the remaining OPERABLE loop is a shutdown cooling loop, be in COLD SHUTDOWN within 24 hours.

b. With no reactor coolant or shutdown cooling loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required coolant loop to operation.

---

*All reactor coolant pumps and shutdown cooling pumps may be deenergized for up to 1 hour provided (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

**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.

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.

REACTOR COOLANT SYSTEM

COLD SHUTDOWN - LOOPS FILLED

LIMITING CONDITION FOR OPERATION

3.4.1.4.1 At least one shutdown cooling loop shall be OPERABLE and in operation*, and either:

a. One additional shutdown cooling loop shall be OPERABLE#, or

b. The secondary side water level of at least two steam generators shall be greater than 25% indicated wide range level.

APPLICABILITY: MODE 5 with reactor coolant loops filled##.

ACTION:

a. With less than the above required loops OPERABLE or with less than the required steam generator level, immediately initiate corrective action to return the required loops to OPERABLE status or to restore the required level as soon as possible.

b. With no shutdown cooling loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required shutdown cooling loop to operation.

SURVEILLANCE REQUIREMENTS

4.4.1.4.1.1 The secondary side water level of both steam generators when required shall be determined to be within limits at least once per 12 hours.

4.4.1.4.1.2 At least one shutdown cooling loop shall be determined to be in operation and circulating reactor coolant at a flow rate of greater than or equal to 4000 gpm at least once per 12 hours.

---

*The shutdown cooling pump may be deenergized for up to 1 hour provided (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

#One shutdown cooling loop may be inoperable for up to 2 hours for surveillance testing provided the other shutdown cooling loop is OPERABLE and in operation.

##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.

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.

3/4.4.8 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.8.1 The Reactor Coolant System (except the pressurizer) temperature and pressure shall be limited in accordance with the limit lines shown on 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.

APPLICABILITY: At all times*.

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limit within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operations or be in at least HOT STANDBY within the next 6 hours and reduce the RCS T_{cold} and pressure to less than 210°F and 500 psia, respectively, within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.8.1.1 The Reactor Coolant System temperature and pressure shall be determined to be within the limits at least once per 30 minutes during system heatup, cooldown, and inservice leak and hydrostatic testing operations.

4.4.8.1.2 The reactor vessel material irradiation surveillance specimens shall be removed and examined, to determine changes in material properties, at the intervals required by 10 CFR Part 50 Appendix H in accordance with the schedule in Table 4.4-5. The results of these examinations shall be used to update Figure 3.4-2.

*See Special Test Exception 3.10.5.

TABLE 3.4-3

Maximum Allowable Heatup and Cooldown Rates

<8 Effective Full Power Years

Heatup

T_c^* (°F)	Rate (°F/HR)
< 128°F	20°F/HR
128° - 180°F	30°F/HR
181° - 230°F	50°F/HR
> 230°F	75°F/HR

Cooldown

T_c^* (°F)	Rate (°F/HR)
≤ 93°F	0°F/HR
94° - 114°F	10°F/HR
115° - 148°F	20°F/HR
> 148°F	100°F/HR

8-32 Effective Full Power Years

Heatup

T_c^* (°F)	Rate (°F/HR)
< 116°F	10°F/HR
117° - 150°F	20°F/HR
151° - 199°F	30°F/HR
200° - 246°F	50°F/HR
> 246°F	75°F/HR

Cooldown

T_c^* (°F)	Rate (°F/HR)
≤ 108°F	0°F/HR
109° - 126°F	10°F/HR
127° - 147°F	20°F/HR
148° - 162°F	40°F/HR
> 162°F	100°F/HR

* Indicated Cold Leg Temperature

FIGURE 3.4-2a
REACTOR COOLANT SYSTEM PRESSURE/TEMPERATURE LIMITATIONS FOR LESS THAN 8 EFFECTIVE FULL POWER YEARS OF OPERATION

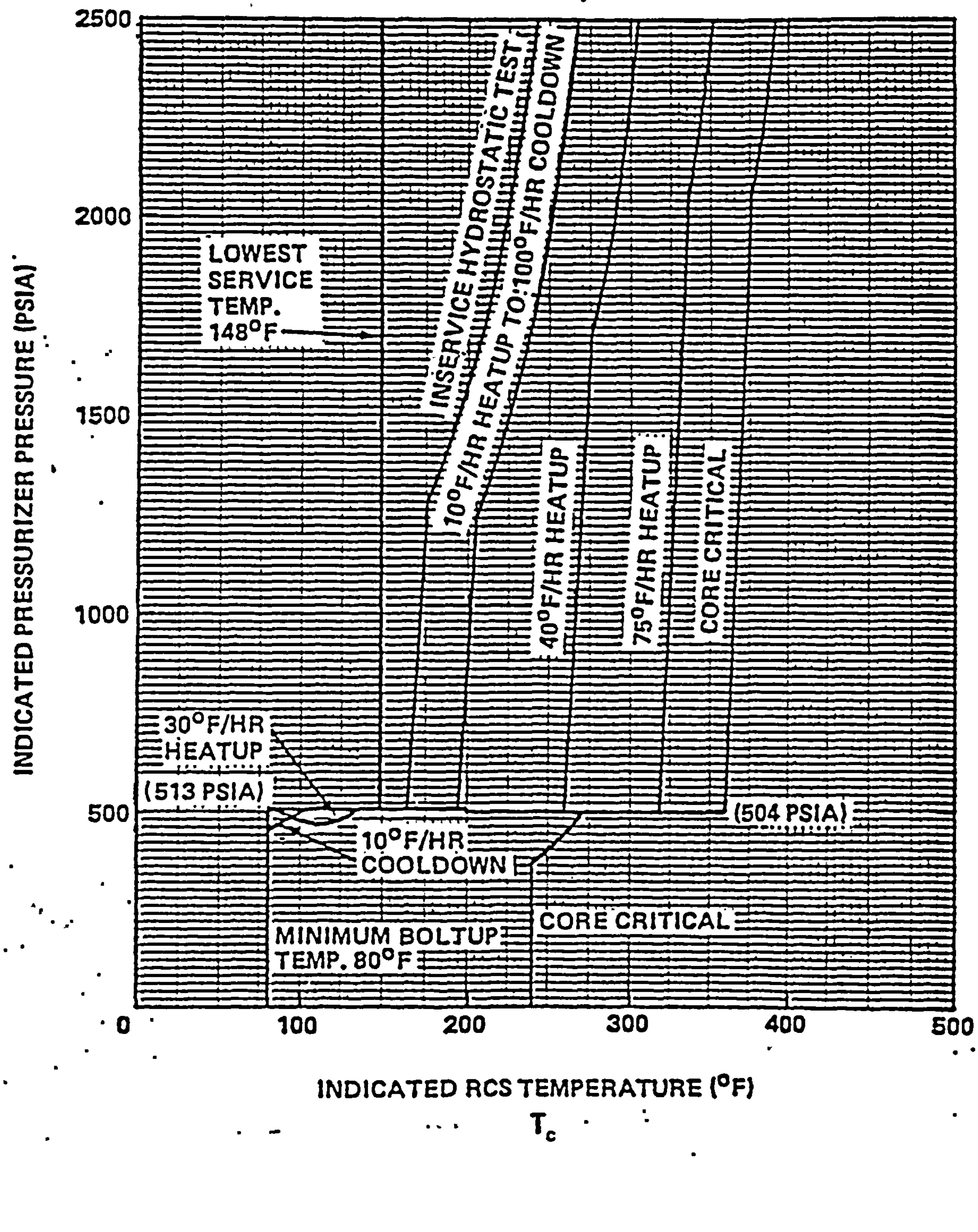
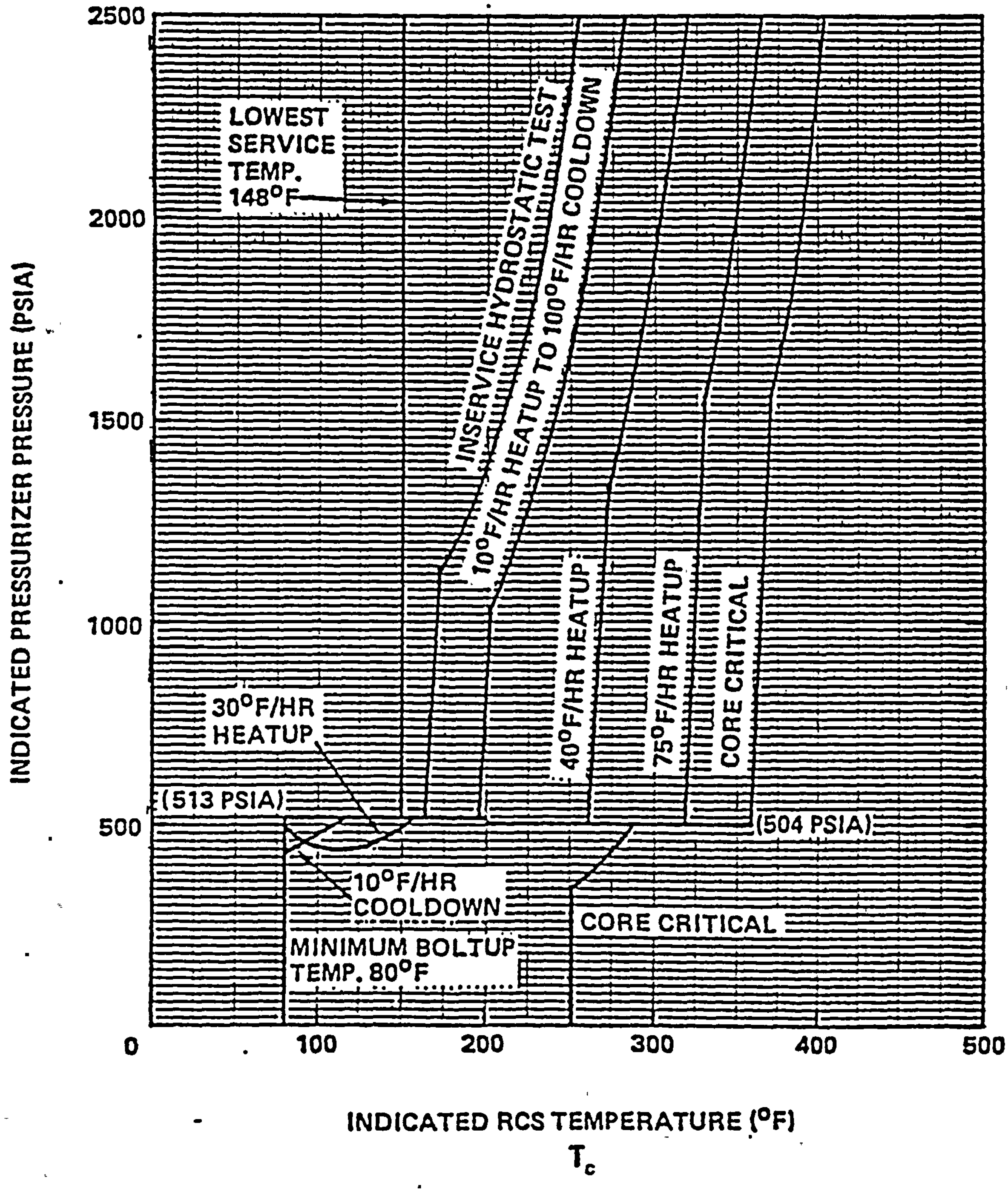


FIGURE 3.4-2b
REACTOR COOLANT SYSTEM PRESSURE/TEMPERATURE LIMITATIONS FOR 8 TO 32 EFFECTIVE FULL POWER YEARS OF OPERATION



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REACTOR COOLANT SYSTEM

OVERPRESSURE PROTECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.8.3 Both shutdown cooling system (SCS) suction line relief valves with lift settings of less than or equal to 467 psig shall be OPERABLE and aligned to provide overpressure protection for the Reactor Coolant System.

APPLICABILITY: When the reactor vessel head is installed and the temperature of one or more of the RCS cold legs is less than or equal to:

a. 214°F during cooldown

b. 291°F during heatup

ACTION:

a. With one SCS relief valve inoperable, restore the inoperable valve to OPERABLE status within seven days or reduce T_{cold} to less than 200°F and, depressurize and vent the RCS through a greater than or equal to 16 square inch vent(s) within the next eight hours. Do not start a reactor coolant pump if the steam generator secondary water temperature is greater than 100°F above any RCS cold leg temperature.

b. With both SCS relief valves inoperable, reduce T_{cold} to less than 200°F and, depressurize and vent the RCS through a greater than or equal to 16 square inch vent(s) within eight hours. Do not start a reactor coolant pump if the steam generator secondary water temperature is greater than 100°F above any RCS cold leg temperature.

c. In the event either the SCS suction line relief valves or an RCS vent(s) are used to mitigate an RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the SCS suction line relief valves or RCS vent(s) on the transient and any corrective action necessary to prevent recurrence.

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.8.3.1 Each SCS suction line relief valve shall be verified to be aligned to provide overpressure protection for the RCS once every 8 hours during

a. Cooldown with the RCS temperature less than or equal to 214°F.

b. Heatup with the RCS temperature less than or equal to 291°F.

4.4.8.3.2 The SCS suction line relief valves shall be verified OPERABLE with the required setpoint at least once per 18 months.

BASES

SPECIFIC ACTIVITY (Continued)

Reducing T_{cold} to less than 500°F prevents the release of activity should a steam generator tube rupture since the saturation pressure of the primary coolant is below the lift pressure of the atmospheric steam relief valves. The surveillance requirements provide adequate assurance that excessive specific activity levels in the primary coolant will be detected in sufficient time to take corrective action. Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomena. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained.

3/4.4.8 PRESSURE/TEMPERATURE LIMITS

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

[illegible]

PRESSURE/TEMPERATURE LIMITS (Continued)

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 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 postcore 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

PRESSURE/TEMPERATURE LIMITS (Continued)

curves Figures 3.4-2(a) and 3.4-2(b) include 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 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-6411-2 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."

[illegible]

PRESSURE/TEMPERATURE LIMITS (Continued)

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 Part 50 when one or more of the RCS cold legs are less than or equal to 214°F during cooldown and 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 cool-down rates to assure that the PT 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.

PRESSURE/TEMPERATURE LIMITS (Continued)

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 temperature 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.

[illegible]

3/4.4.9 STRUCTURAL INTEGRITY

The inservice inspection and testing programs for ASME Code Class 1, 2, and 3 components ensure that the structural integrity and operational readiness of these components will be maintained at an acceptable level throughout the life of the plant. These programs are in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50.55a(g) except where specific written relief has been granted by the Commission pursuant to 10 CFR 50.55a (g) (6) (i).

Components of the Reactor Coolant System were designed to provide access to permit inservice inspections in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, 1974 Edition and Addenda through Summer 1975.

3/4.4.10 REACTOR COOLANT SYSTEM VENTS

Reactor Coolant System vents are provided to exhaust noncondensible gases and/or steam from the primary system that could inhibit natural circulation core cooling. The OPERABILITY of at least one Reactor Coolant System vent path from the reactor vessel head ensures the capability exists to perform this function.

The valve redundancy of the Reactor Coolant System vent paths serves to minimize the probability of inadvertent or irreversible actuation while ensuring that a single failure of a vent valve, power supply, or control system does not prevent isolation of the vent path.

The function, capabilities, and testing requirements of the Reactor Coolant System vent systems are consistent with the requirements of Item II.B.1 of NUREG-0737.