July 25, 1990

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

See correction letter of 8/10/90

Mr. William F. Conway
Executive Vice President
Arizona Public Service Company
Post Office Box 52034
Phoenix, Arizona 85072-2034

Dear Mr. Conway:

SUBJECT: ISSUANCE OF AMENDMENT NO. 52 TO FACILITY OPERATING LICENSE NO. NPF-41, AMENDMENT NO. 38 TO FACILITY OPERATING LICENSE NO. NPF-51 AND AMENDMENT NO. 24 TO FACILITY OPERATING LICENSE NO. NPF-74 FOR THE PALO VERDE NUCLEAR GENERATING STATION, UNITS 1, 2, AND 3 (TAC NOS. 71527, 71528 and 71529)

The Commission has issued the subject Amendments, which are enclosed, to the Facility Operating Licenses for Palo Verde Nuclear Generating Station, Units 1, 2, and 3. The Amendments consist of changes to the Technical Specifications (Appendix A to each license) in response to your application transmitted by letter dated March 13, 1990.

The Amendments revise those portions of the Technical Specifications regarding 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 2 of Regulatory Guide 1.99, "Radiation Embrittlement of Reactor Vessel Materials."

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

Sincerely,
ORIGINAL SIGNED BY SHERI R. PETERSON
Sheri R. Peterson, Project Manager
Project Directorate V
Division of Reactor Projects III, IV, V and Special Projects
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 52 to NPF-41
2. Amendment No. 38 to NPF-51
3. Amendment No. 24 to NPF-74
4. Safety Evaluation

cc: See next page

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(10)

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) Technical Specifications and Environmental Protection Plan

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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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.

#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.

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

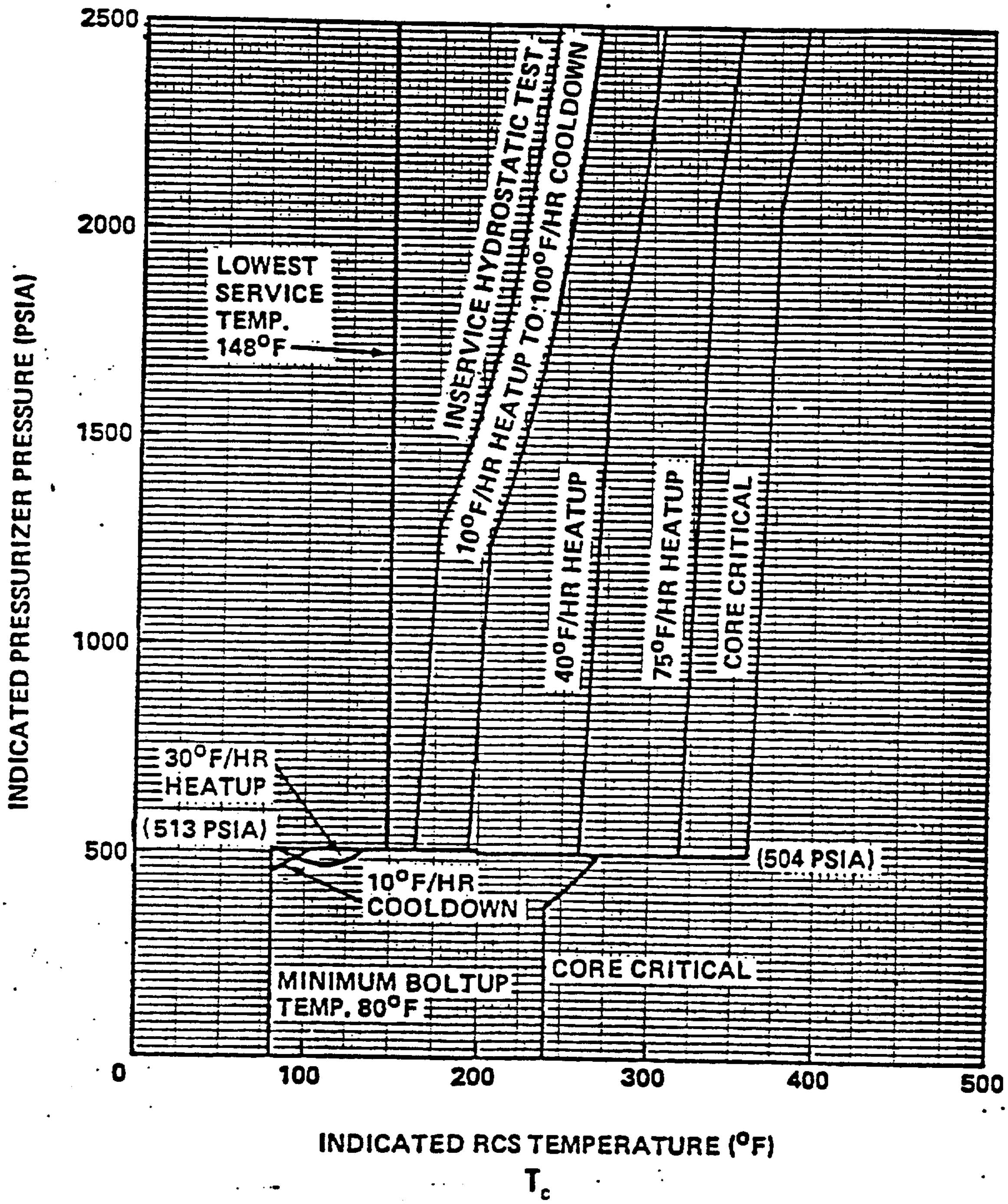
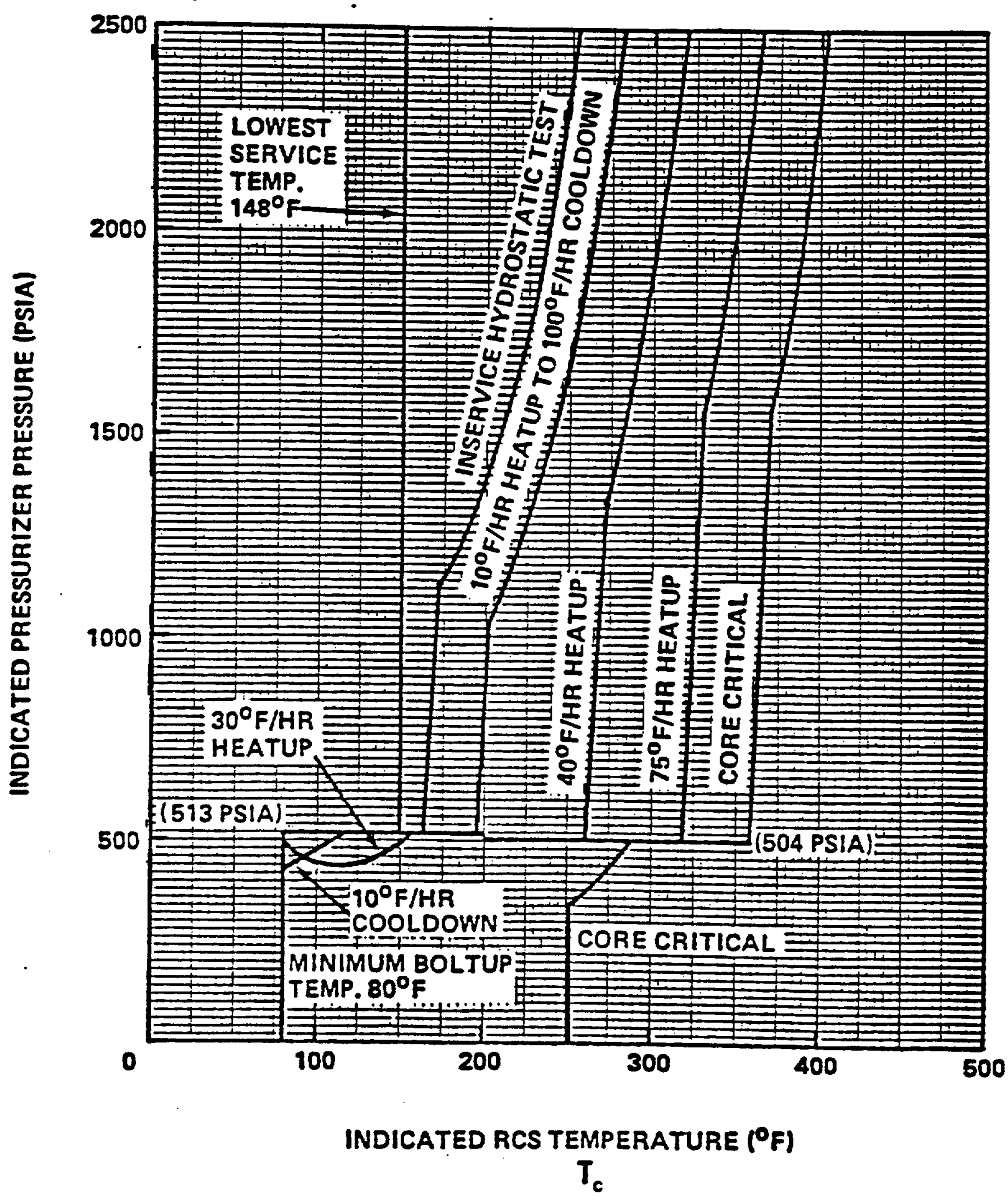


FIGURE 3.4-2B
REACTOR COOLANT SYSTEM PRESSURE/TEMPERATURE LIMITATIONS FOR 8 TO 32 EFFECTIVE FULL POWER YEARS OF OPERATION



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.

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

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

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 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 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

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,

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
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LIST OF TABLES

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

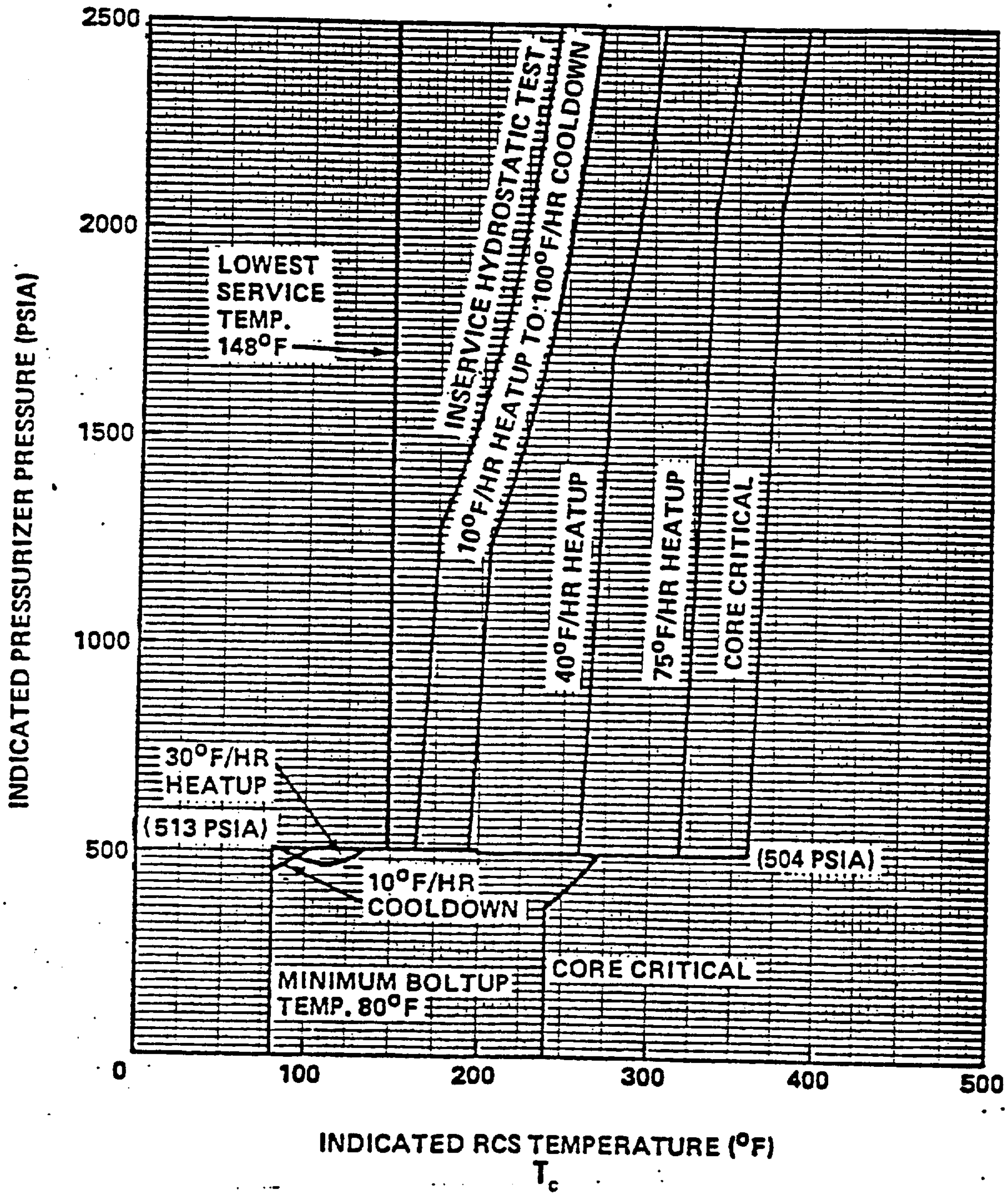
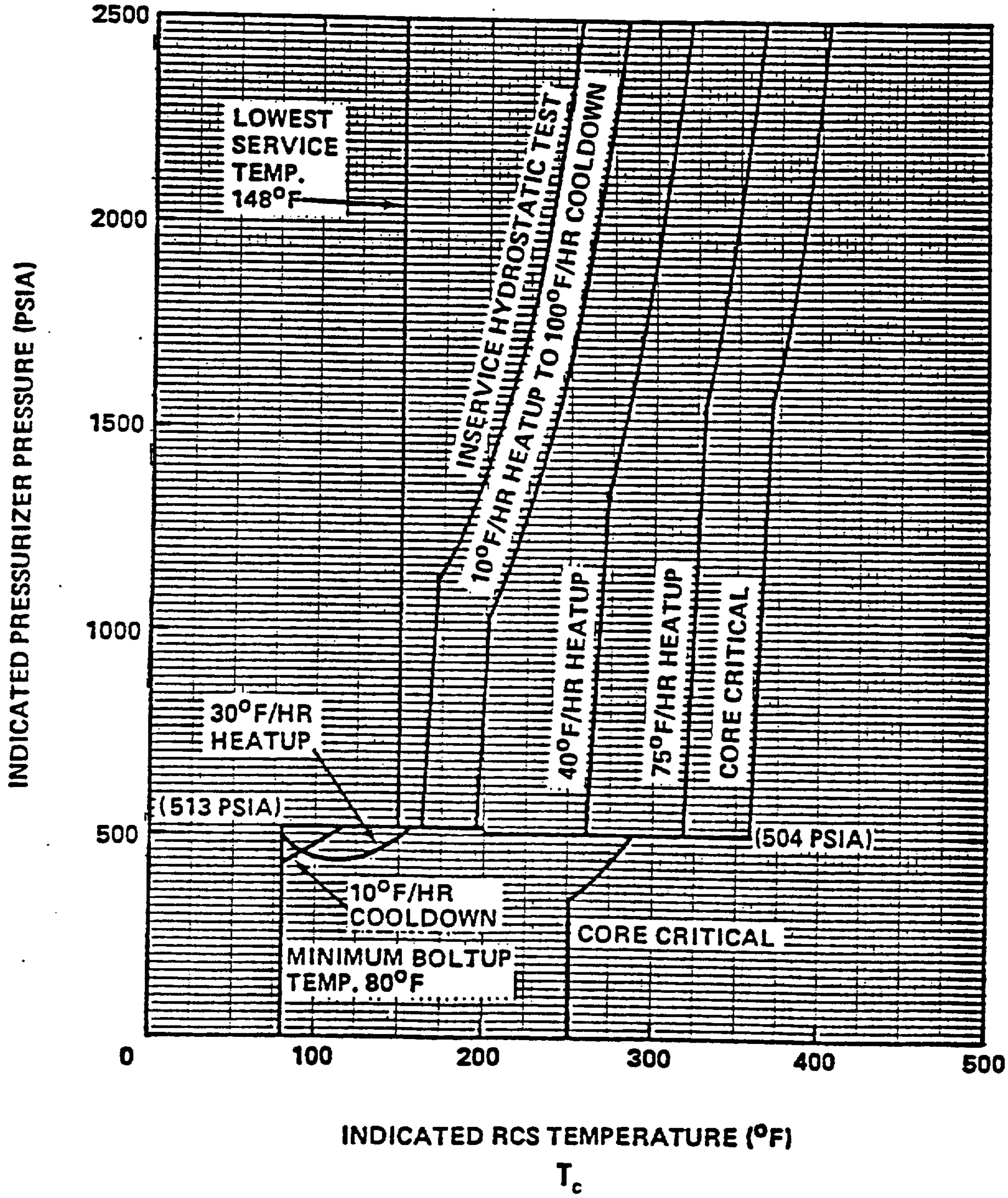


FIGURE 3.4-2b
REACTOR COOLANT SYSTEM PRESSURE/TEMPERATURE LIMITATIONS FOR 8 TO 32 EFFECTIVE FULL POWER YEARS OF OPERATION



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.

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

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

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."

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.

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XIX	XIX
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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.

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

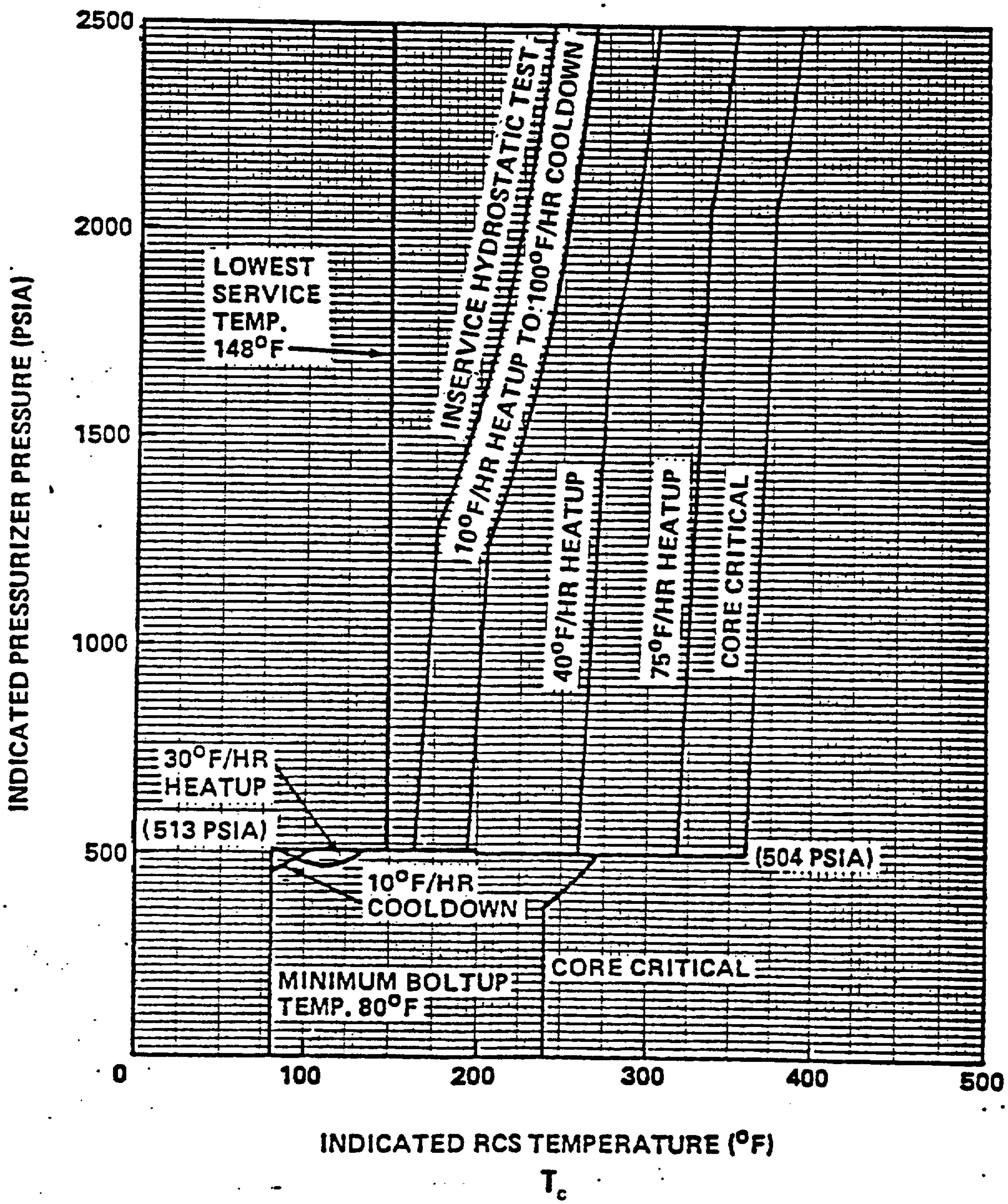
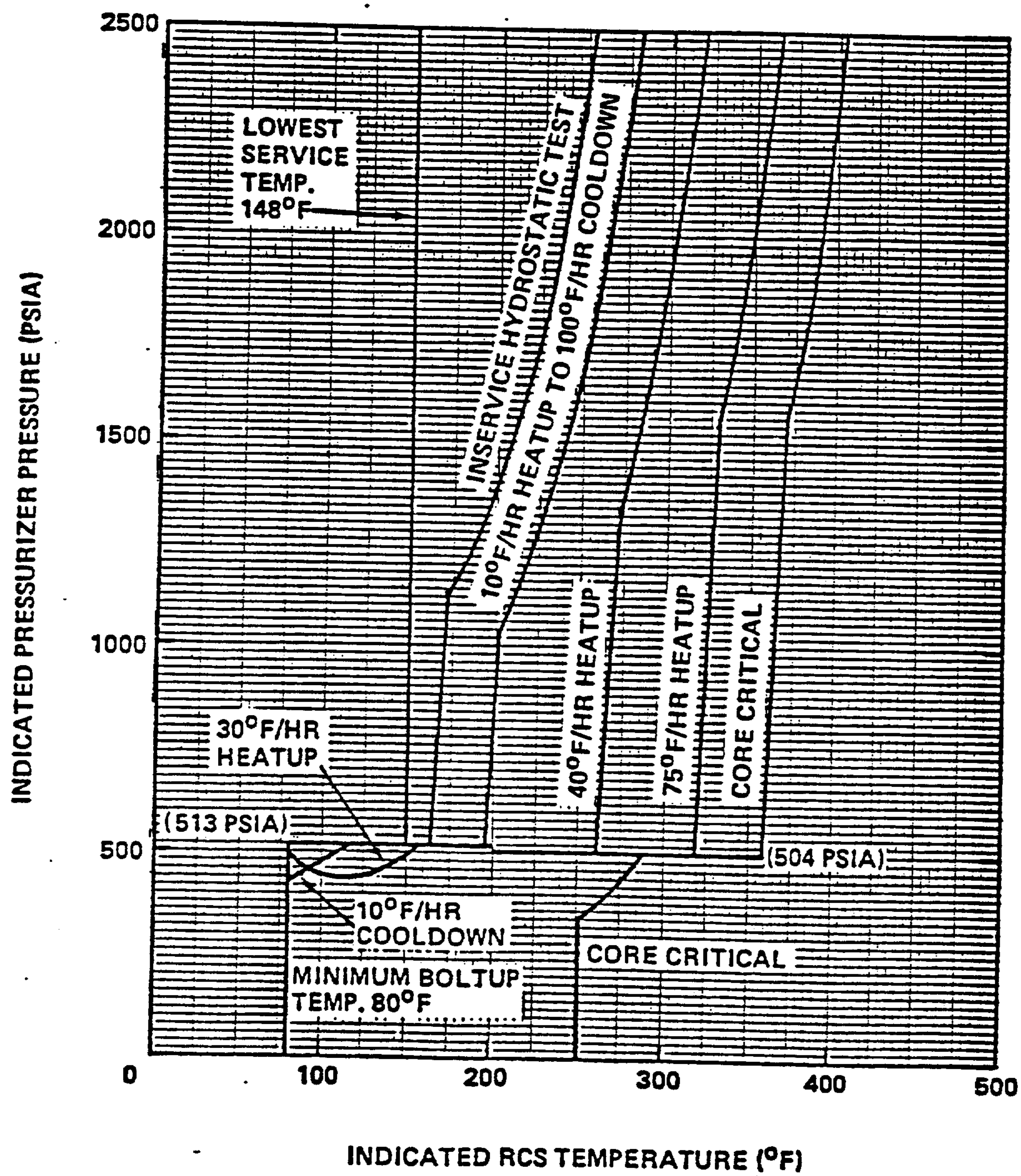


FIGURE 3.4-2b
REACTOR COOLANT SYSTEM PRESSURE/TEMPERATURE LIMITATIONS FOR 8 TO 32 EFFECTIVE FULL POWER YEARS OF OPERATION



T_c

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.

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

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."

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.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

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

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

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

ARIZONA PUBLIC SERVICE COMPANY, ET AL.

PALO VERDE NUCLEAR GENERATING STATION, UNITS 1, 2, AND 3

DOCKET NOS. STN 50-528, STN 50-529, AND STN 50-530

1.0 INTRODUCTION

By letter dated March 13, 1990, the Arizona Public Service Company (APS or the licensee) on behalf of itself, the Salt River Project Agricultural Improvement and Power District, Southern California Edison Company, El Paso Electric Company, Public Service Company of New Mexico, Los Angeles Department of Water and Power, and Southern California Public Power Authority (licensees), requested changes to the Technical Specifications for the Palo Verde Nuclear Generating Station, Units 1, 2, and 3 (Appendix A to Facility Operating License Nos. NPF-41, NPF-51, and NPF-74, respectively). The proposed changes 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 2 of Regulatory Guide 1.99 "Radiation Embrittlement of Reactor Vessel Materials," May 1988.

The Reactor Coolant System (RCS) pressure-temperature limits during plant heatup and cooldown are specified in Technical Specification 3.4.8.1 for the Palo Verde Units. The pressure-temperature curves in the current Technical Specifications are based on an assumed design basis neutron fluence through 10 effective full power years (EFPY). The proposed amendments change the effectiveness of the P/T limits of 8 and 32 effective full power years (EFPY). The licensee proposed to use one set of P/T limits for all three units. The proposed P/T limits were developed based on Section 1 of Regulatory Guide (RG) 1.99, Revision 2. The proposed revision provides up-to-date P/T limits for the operation of the reactor coolant system during heatup, cooldown, criticality, and hydrotest.

APS provided its updated pressure-temperature curves in proposed Technical Specification Figure 3.4-2a (for less than 8 EFPY) and Figure 3.4-2b (for 8 to 32 EFPY), changes in the values of the RCS cold leg temperature at which LTOP should be enabled, and the justification for the changes. New heatup and cooldown rates as a function of indicated reactor coolant temperature are also proposed in an updated Table 3.4-3.

2.0 EVALUATION OF THE P/T LIMITS

To evaluate the P/T limits, the staff used the following NRC regulations and guidance: Appendices G and H of 10 CFR Part 50; the ASTM Standards and the ASME Code, which are referenced in Appendices G and H; 10 CFR 50.36(c)(2); RG 1.99, Rev. 2; Standard Review Plan (SRP) Section 5.3.2; and Generic Letter 88-11.

Each licensee authorized to operate a nuclear power reactor is required by 10 CFR 50.36 to provide Technical Specifications for the operation of the plant. In particular, 10 CFR 50.36(c)(2) requires that limiting conditions of operation be included in the Technical Specifications. The P/T limits are among the limiting conditions of operation in the Technical Specifications for all commercial nuclear plants in the U.S. Appendices G and H of 10 CFR Part 50 describe specific requirements for fracture toughness and reactor vessel material surveillance that must be considered in setting P/T limits. An acceptable method for constructing the P/T limits is described in SRP Section 5.3.2.

Appendix G of 10 CFR Part 50 specifies fracture toughness and testing requirements for reactor vessel materials in accordance with the ASME Code and, in particular, that the beltline materials in the surveillance capsules be tested in accordance with Appendix H of 10 CFR Part 50. Appendix H, in turn, refers to ASTM Standards. These tests define the extent of vessel embrittlement at the time of capsule withdrawal in terms of the increase in reference temperature. Appendix G also requires the licensee to predict the effects of neutron irradiation on vessel embrittlement by calculating the adjusted reference temperature (ART) and Charpy upper shelf energy (USE). Generic Letter 88-11 requested that licensees and permittees use the methods in RG 1.99, Rev. 2, to predict the effect of neutron irradiation on reactor vessel materials. This guide defines the ART as the sum of unirradiated reference temperature, the increase in reference temperature resulting from neutron irradiation, and a margin to account for uncertainties in the prediction method.

Appendix H of 10 CFR Part 50 requires the licensee to establish a surveillance program to periodically withdraw surveillance capsules from the reactor vessel. Appendix H refers to the ASTM Standards which, in turn, require that the capsules be installed in the vessel before startup and that they contain test specimens made from plate, weld, and heat-affected-zone (HAZ) materials of the reactor beltline.

The staff evaluated the effect of neutron irradiation embrittlement on each beltline material in PVNGS 1, 2, and 3 reactor vessels. The amount of irradiation embrittlement was calculated in accordance with RG 1.99, Rev. 2. The staff has determined that the limiting materials at 8 EFPY and 32 EFPY for all three units were Unit 1 intermediate shell plates M-6701-1 with 0.07% copper (Cu), 0.66% nickel (Ni), and an initial RT_{ndt} of 30°F; and plate M-6701-2 with 0.06% Cu, 0.61% Ni, and an initial RT_{ndt} of 40°F.

The licensee has not removed any surveillance capsules from PVNGS 1, 2, and 3 because none of the units has reached the removal date in the capsule withdrawal schedule. The staff has ascertained that all surveillance capsules contained Charpy impact specimens and tensile specimens made from base metal, weld metal, and HAZ metal.

At 8 EFPY, the staff calculated the highest ART to be 96.8°F and 72.2°F at the 1/4T (T= reactor vessel beltline thickness) and at 3/4T locations, respectively. At 32 EFPY, the staff calculated the ART to be 116°F and 97.8°F at the 1/4T and 3/4T locations. The staff used a neutron fluence of 4.2E18 n/cm^2 at 1/4T and 1.09E18 n/cm^2 at 3/4T at 8 EFPY. The staff used a neutron fluence of 1.68E19 n/cm^2 at 1/4T and 4.38E18 at 3/4T at 32 EFPY. The ART was determined by Section 1 of RG 1.99, Rev. 2 because no surveillance capsules have been removed.

The licensee used the method in RG 1.99, Rev. 2, to calculate an ART of 102°F and 90°F for the 1/4T and 3/4T locations at 8 EFPY, and 116°F and 103°F for the 1/4T and 3/4T locations at 32 EFPY. The licensee identified plates M-6701-2 and M-6701-3 as the limiting materials. The difference between the staff and licensee's limiting materials selection and ARTs is because the licensee used a different safety margin in calculating ART. The staff considers the licensee's limiting materials and ARTs acceptable. Substituting the licensee's ARTs into equations in SRP 5.3.2, the staff verified that the proposed P/T limits for heatup, cooldown, and hydrotest meet the beltline material requirements in Appendix G of 10 CFR Part 50.

In addition to beltline materials, Appendix G of 10 CFR Part 50 also imposes P/T limits based on the reference temperature for the reactor vessel closure flange materials. Section IV.2 of Appendix G states that when the pressure exceeds 20% of the preservice system hydrostatic test pressure, the temperature of the closure flange regions highly stressed by the bolt preload must exceed the reference temperature of the material in those regions by at least 120°F for normal operation and by 90°F for hydrostatic pressure tests and leak tests. Based on the highest flange reference temperature of -10°F for Unit 3, the staff has determined that the proposed P/T limits satisfy Section IV.2 of Appendix G.

Section IV.B of Appendix G requires that the predicted Charpy USE at end of life be above 50 ft-lb. Using the method in RG 1.99, Rev. 2, the lowest predicted Charpy USE of all beltline materials is plate M-6701-1 from Unit 1 with 62.2 ft-lb. This is above 50 ft-lb and, therefore, is acceptable.

3.0 EVALUATION OF LTOP

LTOP is provided by relief valves on the Shutdown Cooling System (SCS) lines. These relief valves are set at a pressure low enough to prevent violation of the Appendix G heatup and cooldown curves should a RCS pressure transient occur during low temperature operations. The licensee, in its March 13, 1990 submittal, identified the most limiting overpressure transients analyzed to determine the relief valve setpoint for LTOP. The relief valve setpoint limit has been previously set by analysis of the limiting transients for mass addition and energy addition. Technical Specification 3.4.8.3 currently requires that two relief valves shall be OPERABLE with the setpoint selected for the low temperature mode of operation. The modified Technical Specification 3.4.8.3 maintains the same pressure setpoint and revises the values of the applicable temperatures for LTOP based on a reanalysis of the limiting transients.

The most limiting mass addition transient was analyzed assuming two High Pressure Safety Injection (HPSI) pumps injecting into a water solid RCS with full charging capacity and with the letdown isolated. The transient analysis is typically performed to determine the pressure overshoot past the LTOP setpoint such that the Appendix G curves are not exceeded during the transient.

The energy input transient was analyzed assuming a 100°F temperature difference between the steam generator and the RCS cold leg. A reactor coolant pump startup in one loop was assumed in order to maximize the heat transfer effect. As was the case for the mass addition transient, the pressure overshoot is calculated such that the Appendix G pressure-temperature curves for each Unit are not exceeded.

The licensee's analyses were performed using the same methodology as the prior application for ten EFPY. For the revised analyses the LTOP enable temperatures were determined by following the guidance that for LTOP, the enable temperature is the water temperature corresponding to a metal temperature at the vessel beltline that is controlling in the Appendix G calculations. The resulting enable temperatures were calculated by the licensee to be 291°F during heatup and 214°F during cooldown. The results indicated that a change in the present SCS relief valve setpoint of 467 psig is not required. A footnote is added to Technical Specifications 3.4.1.3 and 3.4.1.4.1 which states:

> 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.

This note is added to maintain the analysis assumptions of the flow induced pressure correction factors due to Reactor Coolant Pump operation.

The licensee-proposed changes in Technical Specifications 3.4.1.3, 3.4.1.4.1, 3.4.8.1, 3.4.8.3 and 4.4.8.3 and the associated bases sections reflect the above discussed LTOP alignment temperatures and the heatup and cooldown rates identified by the updated Figures 3.4-2a and 3.4-2b, and Table 3.4-3 in Technical Specification 3.4.8.1. The staff finds that they are reasonably conservative and acceptable.

4.0 CONCLUSION

Based on the staff evaluation in Section 2.0, the staff concludes that the proposed P/T limits for the reactor coolant system for heatup, cooldown, leak test, and criticality are valid through 8 EFPY and 32 EFPY because the limits conform to the requirements of Appendices G and H of 10 CFR Part 50. The licensee's submittal also satisfies Generic Letter 88-11 because the licensee used the method in RG 1.99, Rev. 2 to calculate the ART. Hence, the proposed P/T limits may be incorporated into the PVNGS 1, 2, and 3 Technical Specifications.

Based on the staff evaluation in Section 3.0, the staff concludes that the proposed Technical Specifications 3.4.1.3, 3.4.1.4.1, 3.4.8.1, 3.4.8.3, and 4.4.8.3.1 and their associated bases are acceptable to support the updated pressure-temperature limits identified in Technical Specification Figures 3.4-2a and 3.4-2b applicable for a period up to 32 EFPY.

5.0 CONTACT WITH STATE OFFICIAL

The Arizona Radiation Regulatory Agency has been advised of the proposed determination of no significant hazards consideration with regard to these changes. No comments were received.

6.0 ENVIRONMENTAL CONSIDERATION

The amendments involve changes in a requirement with respect to the installation or use of facility components located within the restricted area as defined in 10 CFR Part 20. The staff has determined that the amendments involve no significant increase in the amount, and no significant change in the type, of any effluent that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued proposed findings that the amendments involves no significant hazard consideration, and there has been no public comment on such finding. Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need to be prepared in connection with the issuance of the amendments.

7.0 CONCLUSION

The staff has concluded, based on the considerations discussed above, that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public. We, therefore, conclude that the proposed changes are acceptable.

8.0 REFERENCES

1. Regulatory Guide 1.99, Radiation Embrittlement of Reactor Vessel Materials, Revision 2, May 1988

2. NUREG-0800, Standard Review Plan, Section 5.3.2, Pressure-Temperature Limits

3. Palo Verde Nuclear Generating Station Updated Final Safety Analysis Report

4. January 31, 1989, Letter from D. B. Karner (APS) to USNRC Document Control Desk, Subject: Palo Verde Nuclear Generating Station, Units 1, 2, and 3; Generic Letter 88-11

5. March 13, 1990, Letter from W. F. Conway (APS) to USNRC Document Control Desk, Subject: Palo Verde Nuclear Generating Station, Units 1, 2, and 3; Proposed Technical Specification to Incorporate the Requirements of Generic Letter 88-11

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