APS

Palo Verde Nuclear
Generating Station

David Mauldin
Vice President
Nuclear Engineering
and Support

TEL (623) 393-5553
FAX (623) 393-6077

10 CFR 50.90
Mail Station 7605
P.O. Box 52034
Phoenix, AZ 85072-2034

102-04641-CDM/RAB
December 21, 2001

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

Dear Sirs:

**Subject: Palo Verde Nuclear Generating Station (PVNGS)
Unit 2
Docket No. STN 50-529
Request for a License Amendment to Support Replacement of Steam Generators and Uprated Power Operations**

Pursuant to 10 CFR 50.90, Arizona Public Service Company (APS) submits herewith a request to amend Facility Operating License NPF-51 and Appendix A, Technical Specifications for PVNGS Unit 2. The proposed changes support replacement of the steam generators and subsequent operation at an increased maximum power level of 3990 MWt, a 2.94% increase. Operating License and Technical Specification changes associated with this Power Uprate (PUR) amendment request are described in Attachment 2, "License Amendment Request Analysis." As noted in Attachment 2, some of the proposed changes are being made to accomplish the PUR and others are needed both to accomplish the PUR as well as the steam generator replacement.

The uprate program included a reanalysis or evaluation of the Large Break Loss of Coolant Accident (LBLOCA), Small Break Loss of Coolant Accidents (SBLOCA), non-LOCA accidents, and Nuclear Steam Supply System (NSSS) and Balance-of-Plant (BOP) Structures, Systems and Components (SSCs). Major NSSS components (e.g., reactor pressure vessel, pressurizer, reactor coolant pumps, and steam generators) and BOP components (e.g., safety injection, auxiliary feedwater, shutdown cooling, electrical distribution, emergency diesel generators, containment cooling and the ultimate heat sink) have been assessed with respect to the bounding conditions expected for operation at the uprated power level. Control systems (e.g., reactor regulating, pressurizer pressure and level, turbine control, feedwater control and steam bypass control) have been evaluated for operation at uprated power conditions. The results of the above analyses and evaluations have yielded acceptable results and demonstrate that applicable design basis acceptance criteria will continue to be met during uprated power operations.

w/CD A001

The Reactor Protective System and Engineered Safety Features Actuation System set points assessment determined that low steam generator pressure set points for reactor trip and main steam isolation need to be changed and, as such, are part of this amendment request as described in Attachment 2.

The analyses performed to support PUR assume that the Replacement Steam Generators (RSGs) have been installed. These analyses demonstrate that PVNGS Unit 2 continues to meet applicable licensing criteria. Analyses and evaluations that have been performed in support of PUR include the application of methods and assumptions not previously used for PVNGS. These changes are identified in the Executive Summary of Attachment 6.

In addition to the plant changes directly associated with the Operating License and Technical Specification revisions described in Attachment 2, the containment spray flow instrumentation will be changed to provide increased margin for surveillance testing, the spray pond temperature monitoring system will be improved, and the steam admission logic for the high pressure turbine will be changed from partial arc admission to full arc admission. Additionally, setpoints in the core operating limit supervisory system, pressurizer level control system, feedwater control system, and steam bypass control system will be adjusted. These changes are discussed in Attachment 6 and do not require changes to the Technical Specifications.

The proposed PUR would be implemented during the plant start-up after the steam generators are replaced in Unit 2 during refueling No. 11, scheduled for the fall 2003 outage. APS requests approval of these proposed amendments by September 1, 2002 so that approved values may be used in the core reload design scheduled to begin in September 2002. APS requests to implement these changes prior to the entry into Mode 4 after refueling outage No. 11, currently scheduled for December 2003.

In accordance with the PVNGS Quality Assurance Program, the Plant Review Board and the Offsite Safety Review Committee have reviewed and concurred with this proposed amendment. By copy of this letter, this submittal is being forwarded to the Arizona Radiation Regulatory Agency (ARRA) pursuant to 10 CFR 50.91(b)(1).

No commitments are being made to the NRC by this letter.

Should you have any questions, please contact Thomas N. Weber at (623) 393-5764.

Sincerely,

David Mauldin

CDM/RAB/kg

Attachments:

1. Notarized Affidavit
2. License Amendment Request Analysis
3. Markup of Technical Specification Pages
4. Retyped Technical Specification Pages
5. Associated Changes to the Technical Specification Bases (for information only)
6. Power Uprate Licensing Report

Enclosure:

CD-ROM (PDF Normal format) - Palo Verde Nuclear Generating Station, Unit 2, Request for a License Amendment to Support Replacement of Steam Generators and Uprated Power Operations

cc:	E. W. Merschoff	(NRC Region IV)	(w/attachments)
	L. R. Wharton	(NRR Project Manager)	(w/attachments & enclosure)
	J. H. Moorman	(NRC Resident Inspector)	(w/attachments)
	A. V. Godwin	(ARRA)	(w/attachments)

Attachment 1

STATE OF ARIZONA)
) ss.
COUNTY OF MARICOPA)

I, David Mauldin, represent that I am Vice President Nuclear Engineering and Support, Arizona Public Service Company (APS), that the foregoing document has been signed by me on behalf of APS with full authority to do so, and that to the best of my knowledge and belief, the statements made therein are true and correct.

David Mauldin

David Mauldin

Sworn To Before Me This 21st Day Of December, 2001.

Karen D Greiner

Notary Public

"OFFICIAL SEAL"
Karen D. Greiner
Notary Public-Arizona
Maricopa County
My Commission Expires 8/28/2004

Notary Commission Stamp

ATTACHMENT 2

LICENSE AMENDMENT REQUEST ANALYSIS

1.0 DESCRIPTION OF PROPOSED AMENDMENT

The proposed amendment would allow operation of PVNGS Unit 2 up to a maximum reactor core power level of 3990 Megawatts thermal (MWt), an increase of 2.94 percent above the current licensed power level of 3876 MWt. Specifically, the following Unit 2 Facility Operating License and Technical Specification changes are requested to support the increased power operation:

A. Revise paragraph 2.C.(1) of the Facility Operating License to increase the authorized 100% reactor core power (rated thermal power) from 3876 MWt to 3990 MWt, an increase of 2.94%. The new power level of 3990 MWt represents an increase of 5% above the originally licensed power level of 3800 MWt. The increase to 3876 MWt was authorized by the NRC in a letter dated May 23, 1996, Amendment No. 100 for Unit 2.

B. Revise Technical Specification Section 1.1, Definition of Rated Thermal Power, for Unit 2, from 3876 MWt to 3990 MWt.

C. Revise Table 3.3.1-1, Reactor Protective System Instrumentation (referenced in LCO 3.3.1), item 6, Steam Generator #1 Pressure - Low and item 7, Steam Generator #2 Pressure - Low, to increase the Allowable Value from 890 psia to 955 psia for Unit 2. This increase in the Allowable Value is proportional to the increase in steam generator pressure during normal operation and will ensure a comparable reactor protection system response. Both the power uprate and the Replacement Steam Generators (RSGs) affect this specification.

D. Revise Table 3.3.2-1, Reactor Protective System Instrumentation - Shutdown (referenced in LCO 3.3.2), item 2, Steam Generator #1 Pressure - Low and item 3, Steam Generator #2 Pressure - Low, to increase the Allowable Value from 890 psia to 955 psia for Unit 2. This increase in the Allowable Value is proportional to the increase in steam generator pressure during normal operation and will ensure a comparable reactor protection system response. Both the power uprate and the RSGs affect this specification.

E. Revise Table 3.3.5-1, Engineered Safety Features Actuation System Instrumentation (referenced in LCO 3.3.5), item 4.a, Steam Generator #1 Pressure - Low and item 4.b, Steam Generator #2 Pressure - Low, to increase the Allowable Value from 890 psia to 955 psia for Unit 2. This increase in the Allowable Value is proportional to the increase in steam generator pressure during normal operation and will ensure a comparable engineered safety features system response. Both the power uprate and the RSGs affect this specification.

F. Revise Figure 3.4.1-1, Reactor Coolant Cold Leg Temperature vs. Core Power Level, to change the upper limit in the area of acceptable operation for Unit 2. The new upper limit line would allow a cold leg temperature of 570 °F at 0% power, decreasing linearly to 564 °F at 100% power. The upper limit line of Figure 3.4.1-1 in the current Technical Specification decreases linearly from 570 °F at 0% power to 568 °F at 30% power. At 30% power the current figure then decreases linearly from 568 °F to 560 °F at 100% power. The increase in cold leg temperature at 100% power will allow a more optimum main steam pressure for turbine operation. Additionally, editorial changes will be made to Figure 3.4.1-1 for Units 1 and 3. Both the power uprate and RSGs affect this specification.

G. Revise Table 3.7.1-1, Variable Overpower Trip (VOPT) Setpoint versus Operable Main Steam Safety Valves for Unit 2, to decrease the Maximum Power and the Maximum Allowable VOPT Setpoint when the Minimum Number of Main Steam Safety Valves (MSSVs) per Steam Generator Required Operable is less than ten. The reduction in allowable power levels and VOPT setpoints for Unit 2 are required to offset the impacts of increased core power and increased cold leg temperature on the consequences of the UFSAR Chapter 15 design basis events. The power uprate affects this specification.

H. Revise Technical Specification 5.5.16, Containment Leakage Rate Testing Program, to increase the peak calculated containment internal pressure for the design basis loss of coolant accident (P_a) for Unit 2 from 52.0 psig to 58.0 psig. The proposed value for P_a has been rounded up from the actual calculated value of 57.85 psig. The calculated peak containment pressure remains below the containment internal design pressure of 60.0 psig. Both the power uprate and the RSGs affect this specification.

The bases for Technical Specifications 3.6.1, 3.6.2, 3.6.4, 3.6.6 and 3.7.1 would also be revised to reflect these changes and are included in Attachment 5 of this submittal.

2.0 BACKGROUND

This proposed amendment is requested to improve the economic performance of PVNGS Unit 2. Increasing the rated thermal power limit of PVNGS Unit 2 from 3876 MWt to 3990 MWt would result in an increase in electrical output of approximately 55 Megawatts electric (MWe).

The original full power operating license for Unit 2, issued in April 1986, authorized a Rated Thermal Power (RTP) of 3800 MWt. In May 1996, the NRC issued Amendment Nos. 108, 100 and 80 to Units 1, 2 and 3, respectively, to increase the authorized RTP to 3876 MWt. This amendment request to increase RTP to 3990 MWt would be a 2.94% increase above that authorized in Amendment 100, and represents a 5% increase from the original RTP.

Paragraph 2.C.(1) of the Facility Operating License specifies, as a license condition, the maximum reactor core thermal power level at which the unit is authorized to operate. The maximum authorized reactor thermal power level is specified as a license condition in order to limit thermal power to the value used in the safety analyses. The maximum reactor core thermal power specified in the operating license is also known as the Rated Thermal Power (RTP). Regulatory Guide 1.49 recommends a 2% uncertainty be included in the power level used in the safety analysis, as appropriate. Thus, the safety analysis supporting this amendment uses a reactor core thermal power of 4070 MWt, which is 102% of 3990 MWt, the proposed new RTP. The definition of Rated Thermal Power in Technical Specification 1.1 identifies the licensed limit of the total reactor core heat transfer rate to the reactor coolant.

LCO 3.3.1, Reactor Protective System Instrumentation - Operating and Table 3.3.1-1, which it references, specify the required number of channels operable for each reactor trip function, the applicable modes for each function, the surveillance requirements and the allowable value for the setpoint to ensure that the purpose of the function is satisfied. The Steam Generator Pressure - Low trip function (items 6 and 7 in Table 3.3.1-1) provides protection against an excessive rate of heat extraction from the steam generators and the resulting rapid, uncontrolled cooldown of the Reactor Coolant System (RCS). This trip is needed to shut down the reactor and assist the Engineered Safety Features (ESF) system in the event of a Main Steam Line Break (MSLB) or Main Feedwater Line Break (MFWLB) accident. A Main Steam Isolation Signal (MSIS) is initiated simultaneously.

LCO 3.3.2, Reactor Protective System Instrumentation - Shutdown and Table 3.3.2-1, which it references, specify the required number of channels operable for each reactor trip function, the applicable modes for each function, the surveillance requirements and the allowable value for the setpoint to ensure that the purpose of the function is satisfied. The Steam Generator Pressure - Low trip function (items 2 and 3 in Table 3.3.2-1) provides shutdown margin to prevent or minimize the return to power following a large MSLB in Mode 3.

LCO 3.3.5, Engineered Safety Features Actuation System Instrumentation and Table 3.3.5-1, which it references specify the required number of channels operable for each reactor trip function, the applicable modes for each function, and the allowable value for the setpoint to ensure that the purpose of the function is satisfied. The Steam Generator Pressure - Low signal actuates a MSIS to prevent an excessive rate of heat extraction and subsequent cooldown of the RCS in the event of a MSLB or MFWLB.

Figure 3.4.1-1, Reactor Coolant Cold Leg temperature vs. Core Power Level, referenced in LCO 3.4.1, RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits, provides parametric limits to ensure that the actual value of the reactor coolant cold leg temperature is maintained within the range of values used in the safety analysis. The safety analysis supporting this requested amendment uses the proposed new allowable cold leg temperature range (550 °F to 570 °F), and this proposed change maintains the basis for the cold leg temperature limits.

Table 3.7.1-1, Variable Overpower Trip Setpoint versus Operable Main Steam Safety Valves, referenced in LCO 3.7.1, Main Steam Safety Valves (MSSVs), specifies maximum power levels and overpower reactor trip setpoints for specified numbers of

OPERABLE MSSVs. Adherence to the values in the table will ensure that the available relieving capacity maintains secondary system pressure within allowable limits.

Technical Specification 5.5.16, Containment Leakage Rate Testing Program, provides the requirements for the Containment Leakage Rate Testing Program. The calculated peak containment internal pressure for the design basis LOCA (P_a) is the basis for the containment leakage rate in the testing program.

3.0. SAFETY ANALYSIS

Refer to Attachment 6 (Power Uprate Licensing Report).

4.0. NO SIGNIFICANT HAZARDS CONSIDERATION

According to 10CFR 50.92 (c), a proposed amendment to an operating license involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not:

1. involve a significant increase in the probability of occurrence or consequences of an accident previously evaluated,
2. create the possibility of a new or different kind of accident from any previously analyzed, or
3. involve a significant reduction in a margin of safety.

In support of this determination, an evaluation of each of the three criteria set forth in 10 CFR 50.92 is provided below regarding the proposed license amendment.

Overview

APS has completed a comprehensive reanalysis or evaluation of Large Break Loss of Coolant Accidents (LBLOCA), Small Break Loss of Coolant Accidents (SBLOCA), non-LOCA accidents, and Nuclear Steam Supply System (NSSS) and Balance-of-Plant (BOP) structures, systems, and components to demonstrate the acceptability of increasing the licensed reactor power from 3876 Megawatts-thermal (MWt) to 3990 MWt for Unit 2.

Major NSSS components (e.g., reactor pressure vessel, pressurizer, reactor coolant pumps, and steam generators), BOP components (e.g., main turbine, generator, and condensate and feedwater pumps), and major systems and sub-systems (e.g., safety injection, auxiliary feedwater, residual heat removal, electrical distribution, emergency diesel generators, containment spray, and the ultimate heat sink) have been assessed with respect to the bounding conditions expected for operation at the uprated power level. Control systems (e.g., reactor regulating system, pressurizer pressure and level control, turbine control, feedwater control, and steam bypass control) have been evaluated for operation at uprated power conditions. Reactor trip and ESF actuation setpoints have been assessed, and only the changes to low steam generator pressure, were identified as a result of uprated power operations or SG replacement. The analyses and evaluations have yielded acceptable results and demonstrate that all design basis acceptance criteria will continue to be met during uprated power operations. This detailed analysis is presented in the "Power Uprate Licensing Report," submitted as Attachment 6 to this license amendment request.

<u>Standard 1</u> - Do the proposed changes involve a significant increase in the probability or consequences of an accident previously evaluated?

No. The proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

a. Evaluation of the Probability of Previously Evaluated Accidents

Plant Structures, Systems and Components (SSCs) have been verified to be capable of performing their intended design functions at uprated power conditions. Where necessary, a small number of minor modifications will be made prior to implementation of uprated power operations so that surveillance test acceptance criteria continues to be met. The analysis has concluded that operation at uprated power conditions will not adversely affect the capability or reliability of plant equipment. Current technical specification surveillance requirements ensure frequent and adequate monitoring of system and component operability. All systems will continue to be operated within current operating requirements at uprated conditions. Therefore, no new structure, system or component interactions have been identified that could lead to an increase in the probability of any accident previously evaluated in the Updated Final Safety Analysis Report (UFSAR).

b. Evaluation of the Consequences of Previously Evaluated Accidents

The radiological consequences were reviewed for all design basis accidents (DBAs) (i.e., both LOCA and non-LOCA accidents) previously analyzed in the UFSAR. The analyses showed that the resultant radiological consequences for both LOCA and non-LOCA accidents remained within regulatory and Standard Review Plan (SRP) limits at uprated power conditions.

<u>Standard 2</u> - Do the proposed changes create the possibility of a new or different kind of accident from any accident previously evaluated?

No. The proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

The configuration, operation and accident response of the PVNGS Unit 2 structures, systems, and components are unchanged by operation at uprated power conditions or by the associated proposed Technical Specification changes. Analyses of transient events have confirmed that no transient event results in a new sequence of events that could lead to a new accident or different scenario.

The effect of operation at uprated power conditions on plant equipment has been evaluated. No new operating mode, safety-related equipment lineup, accident scenario, or equipment failure mode was identified as a result of operating at uprated conditions. In addition, operation at uprated power conditions does not create any new failure modes that could lead to a different kind of accident. Minor plant modifications, to support implementation of uprated power conditions, will be made as required to existing SSCs. The basic design function of all SSCs remains unchanged and no new equipment or systems have been installed that could potentially introduce new failure modes or accident sequences.

Based on these analyses, it is concluded that no new accident scenarios, failure mechanisms or limiting single failures are introduced as a result of the proposed changes. The proposed changes do not have an adverse effect on any safety-related system or design basis function. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

Standard 3 - Do the proposed changes involve a significant reduction in a margin of safety?

No. The proposed changes do not involve a significant reduction in a margin of safety.

A comprehensive analysis was performed to evaluate the effects of power uprate on PVNGS Unit 2. This analysis identified and defined the major input parameters to the NSSS, reviewed NSSS design transients, and reviewed the capabilities of the NSSS and BOP fluid systems, NSSS/BOP interfaces, NSSS and BOP control systems, and NSSS and BOP SSCs. NSSS accident analyses were re-performed or reviewed to confirm that acceptable results were maintained and that the radiological consequences remained within regulatory and Standard Review Plan (SRP) limits. The nuclear and thermal hydraulic performance of nuclear fuel was also reviewed to confirm acceptable results. The analyses confirmed that all NSSS and BOP SSCs are capable, some with minor modifications, to safely support operations at uprated power conditions.

The margin of safety of the reactor coolant pressure boundary is maintained under uprated power conditions. The design pressure of the reactor pressure vessel and reactor coolant system will not be challenged as the pressure mitigating systems were confirmed to be sufficiently sized to adequately control pressure under uprated power conditions.

Reanalysis of containment structural integrity under Design Basis Accident (DBA) conditions indicates that the calculated peak containment pressure (P_a) increases from 52.0 psig to 58.0 psig, but remains less than the containment internal design pressure of 60 psig. The proposed value for P_a has been rounded up from the actual calculated value of 57.85 psig.

Radiological consequences of the following accidents were reviewed: Main Steam Line Break, Locked Reactor Coolant Pump (RCP) Rotor, CEA Ejection, Small Steam Line Break Outside Containment, Steam Generator Tube Rupture, LBLOCA, SBLOCA, Waste Gas Decay Tank Rupture, Liquid Waste Tank Failure, and Fuel Handling Accident. The resultant radiological consequences for each of these accidents remained within regulatory and SRP limits at uprated power conditions.

The analyses supporting operation at power uprate conditions have demonstrated that all systems and components are capable of safely operating at uprated power conditions. All design basis accident acceptance criteria will continue to be met. Therefore, it is concluded that the proposed changes do not involve a significant reduction in the margin of safety.

Conclusion

Based upon the above analyses and evaluations, APS has concluded that the proposed changes to the Unit 2 operating license and Technical Specifications involve no significant hazards consideration.

5.0. ENVIRONMENTAL CONSIDERATION

APS has determined that the proposed amendment does not involve an unreviewed environmental question, in accordance with Section 3.1 of Appendix B of the Operating License. A proposed change, test or experiment shall be deemed to involve an unreviewed environmental question if it concerns (1) a matter which may result in a significant increase in any adverse environmental impact previously evaluated in the Final Environmental Statement (FES) as modified by the staff's testimony to the Atomic Safety and Licensing Board; or (2) a significant change in the effluents or power level; or (3) a matter not previously reviewed and evaluated in the documents specified in (1) above, which may have a significant adverse environmental impact. Based on the following, this amendment request does not constitute an unreviewed environmental question:

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

APS reviewed the FES and determined that this amendment request does not significantly increase any adverse environmental impact. The plant is not being modified in any way which would significantly increase or change the type of effluents currently produced. The 2.94% increase in RTP is within the 4100 MWt design stretch power evaluated in the FES - Construction Permit Stage (FES-CP). Thus the environmental effects as a result of the uprate are bounded by those previously evaluated during FES-CP phase.

Radiological releases are controlled in accordance with PVNGS Offsite Dose Calculation Manual and the results are reported annually to the NRC. Design Basis Event radiological releases have been demonstrated, in the safety analysis provided with this amendment request, to not significantly increase offsite exposure and remain within regulatory limits. The radiological exposure to plant workers is controlled under the PVNGS As Low As Reasonably Achievable (ALARA) Program and will not significantly change.

2) A significant change in the effluents or power level.

A 2.94% increase in RTP is not a significant increase in power level. The Final Environmental Statement (NUREG 0841) recognized in the Summary and Conclusions Section that the maximum design thermal output for each unit is 4100 MWt. The proposed increase is less than the FES-CP evaluated maximum design thermal output of the units. Thus the environmental effects previously evaluated for land and water usage are bounded by those previously evaluated. The increase in RTP does not change any of the conclusions of NUREG 0841.

Effluents as discussed above will not be significantly increased and are controlled by PVNGS programs and applicable regulations.

3) A matter not previously reviewed and evaluated in the documents specified in (1) above which may have a significant adverse environmental impact.

The increase in RTP does not change the processes, plant equipment, types of effluents, or significantly affect operation of the units. The changes are within the design basis of the NSSS and BOP SSCs at the increased RTP conditions. Safety analyses of design basis events affected by the increase have been reviewed or reanalyzed and the consequences found to be bounded by current UFSAR consequences or within regulatory requirements. The FES-CP, FES-OL, and NUREG-0841 all evaluated the environmental impact assuming the maximum design thermal output of 4100 MWt for each unit. Thus the proposed increase in rated thermal power is within the scope of the previous reviews performed to assess the environmental impact associated with the operation of each unit.

Based on the above, no unreviewed environmental question exists concerning this amendment request for increased RTP and associated Technical Specification changes.

6.0 REFERENCES

References used to develop this request are listed at the end of each section in the Power Uprate Licensing Report (Attachment 6).

7.0 PRECEDENT

A similar amendment request has been approved for the following facilities:

Facility	Amendment #	Approval Date	Accession #
Farley 1, 2	137, 139	April 29, 1998	Not Available
Byron 1, 2	119, 119	May 4, 2001	ML011420274
Braidwood 1, 2	113, 113	May 4, 2001	ML011420274

ATTACHMENT 3

MARKED-UP OPERATING LICENSE AND TECHNICAL SPECIFICATION PAGES

Marked-up Operating License and Technical Specification Pages

Unit 2 Operating License

Page 5

Technical Specifications

Page 1.1-6
Page 3.3.1.8
Page 3.3.2-5
Page 3.3.5-4
Page 3.4.1-3
Page 3.7.1-3
Page 5.5-23

(8)(a) Arizona Public Service Company is authorized to transfer all or a portion of its 29.1% ownership share in Palo Verde, Unit 2 to certain equity investors identified in its submissions of August 6, August 8 and December 5, 1986, and at the same time to lease back from such purchasers such interest sold in the Palo Verde, Unit 2 facility. The term of the lease is for approximately 29-½ years subject to a right of renewal. Additional sale and leaseback transactions of all or a portion of APS's remaining ownership share in Palo Verde, Unit 2 are hereby authorized until June 30, 1987. Any such sale and lease back transaction is subject to the representations and conditions set forth in the aforementioned application of May 2, 1986, and the subsequent submittals dated July 30, August 2, August 6, August 7, August 8, August 13, October 16 and December 5, 1986, as well as the letters of the Director of the Office of Nuclear Reactor Regulation dated August 15, and December 11, 1986, consenting to such transactions. Specifically, the lessor and anyone else who may acquire an interest under this transaction are prohibited from exercising directly or indirectly any control over the licensees of the Palo Verde Nuclear Generating Station, Unit 2. For purposes of this condition the limitations in 10 CFR 50.81, "Creditor Regulations," as now in effect and as they may be subsequently amended, are fully applicable to the lessor and any successor in interest to the lessor as long as the license for Palo Verde, Unit 2 remains in effect; this financial transaction shall have no effect on the license for the Palo Verde nuclear facility throughout the term of the license.

(b) Further, the licensees are also required to notify the NRC in writing prior to any change in: (i) the terms or conditions of any lease agreements executed as part of this transaction; (ii) the ANPP Participation Agreement, (iii) the existing property insurance coverage for the Palo Verde nuclear facility, Unit 2 as specified in licensee counsel's letter of November 26, 1985, and (iv) any action by the lessor or others that may have an adverse effect on the safe operation of the facility.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) <u>Maximum Power Level</u>

Arizona Public Service Company (APS) is authorized to operate the facility at reactor core power levels not in excess of ~~3876~~ megawatts thermal (100% power) in accordance with the conditions specified herein. 3990

1.1 Definitions (continued)

RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 3876 MWt FOR UNITS 1 AND 3, AND 3990 MWt FOR UNIT 2
REACTOR PROTECTIVE SYSTEM (RPS) RESPONSE TIME	The RPS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RPS trip setpoint at the channel sensor until electrical power to the CEAs drive mechanism is interrupted. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC.
SHUTDOWN MARGIN (SDM)	SDM shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming: a. All full length CEAs (shutdown and regulating) are fully inserted except for the single CEA of highest reactivity worth, which is assumed to be fully withdrawn. With any full length CEAs not capable of being fully inserted, the withdrawn reactivity worth of these CEAs must be accounted for in the determination of SDM and b. There is no change in part length CEA position.

(continued)

Table 3.3.1-1 (page 1 of 3)
Reactor Protective System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Variable Over Power	1,2	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.6 SR 3.3.1.7 SR 3.3.1.8 SR 3.3.1.9 SR 3.3.1.13	Ceiling ≤ 111.0% RTP Band ≤ 9.9% RTP Incr. Rate ≤ 11.0%/min RTP Decr. Rate > 5%/sec RTP
2. Logarithmic Power Level – High[a]	2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.12 SR 3.3.1.13	≤ 0.011% NRTP
3. Pressurizer Pressure – High	1,2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.13	≤ 2388 psia
4. Pressurizer Pressure – Low	1,2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.12 SR 3.3.1.13	≥ 1821 psia
5. Containment Pressure – High	1,2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.13	≤ 3.2 psig
6. Steam Generator #1 Pressure – Low	1,2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.13	UNITS 1&3: ≥ 890 psia UNIT 2: ≥ 955 PSIA
7. Steam Generator #2 Pressure – Low	1,2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.13	UNITS 1&3: ≥ 890 psia UNIT 2: ≥ 955 PSIA

(continued)

(a) Trip may be bypassed when logarithmic power is > 1E-4% NRTP. Bypass shall be automatically removed when logarithmic power is ≤ 1E-4% NRTP.

Table 3.3.2-1
Reactor Protective System Instrumentation - Shutdown

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALVE
1. Logarithmic Power Level-High[d]	$3^{(a)}$, $4^{(a)}$, $5^{(a)}$	SR 3.3.2.1 SR 3.3.2.2 SR 3.3.2.3 SR 3.3.2.4 SR 3.3.2.5	≤ 0.011% NRTP[c]
2. Steam Generator #1 Pressure-Low[b]	$3^{(a)}$	SR 3.3.2.1 SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.5	UNITS 1&3: ≥ 890 psia UNIT 2: ≥ 955 PSIA
3. Steam Generator #2 Pressure-Low[b]	$3^{(a)}$	SR 3.3.2.1 SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.5	UNIT 1&3: ≥ 890 psia UNIT 2: ≥ 955 PSIA

(a) With any Reactor Trip Circuit Breakers (RTCBs) closed and any control element assembly capable of being withdrawn.

(b) The setpoint may be decreased as steam pressure is reduced, provided the margin between steam pressure and the setpoint is maintained ≤ 200 psig. The setpoint shall be automatically increased to the normal setpoint as steam pressure is increased.

(c) The setpoint must be reduced to ≤ 1E-4% NRTP when less than 4 RCPs are running.

(d) Trip may be bypassed when logarithmic power is > 1E-4% NRTP. Bypass shall be automatically removed when logarithmic power is ≤ 1E-4% NRTP.

Table 3.3.5-1 (page 1 of 1)
Engineered Safety Features Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	ALLOWABLE VALUE
1. Safety Injection Actuation Signal		
a. Containment Pressure – High	1,2,3	≤ 3.2 psig
b. Pressurizer Pressure – Low[a]		≥ 1821 psia
2. Containment Spray Actuation Signal		
a. Containment Pressure – High High	1,2,3	≤ 8.9 psig
3. Containment Isolation Actuation Signal		
a. Containment Pressure – High	1,2,3	≤ 3.2 psig
b. Pressurizer Pressure – Low[a]		≥ 1821 psia
4. Main Steam Isolation Signal[c]		UNIT 2: ≥ 955 PSIA
a. Steam Generator #1 Pressure–Low[b]	1,2,3	UNITS 1 & 3: ≥ 890 psia
b. Steam Generator #2 Pressure–Low[b]		UNITS 1 & 3: ≥ 890 psia
c. Steam Generator #1 Level-High		≤ 91.5%
d. Steam Generator #2 Level-High		≤ 91.5%
e. Containment Pressure-High		≤ 3.2 psig
5. Recirculation Actuation Signal		
a. Refueling Water Storage Tank Level–Low	1,2,3	≥ 6.9 and ≤ 7.9%
6. Auxiliary Feedwater Actuation Signal SG #1 (AFAS-1)		
a. Steam Generator #1 Level–Low	1,2,3	≥ 25.3%
b. SG Pressure Difference–High		≤ 192 psid
7. Auxiliary Feedwater Actuation Signal SG #2 (AFAS-2)		
a. Steam Generator #2 Level–Low	1,2,3	≥ 25.3%
b. SG Pressure Difference–High		≤ 192 psid

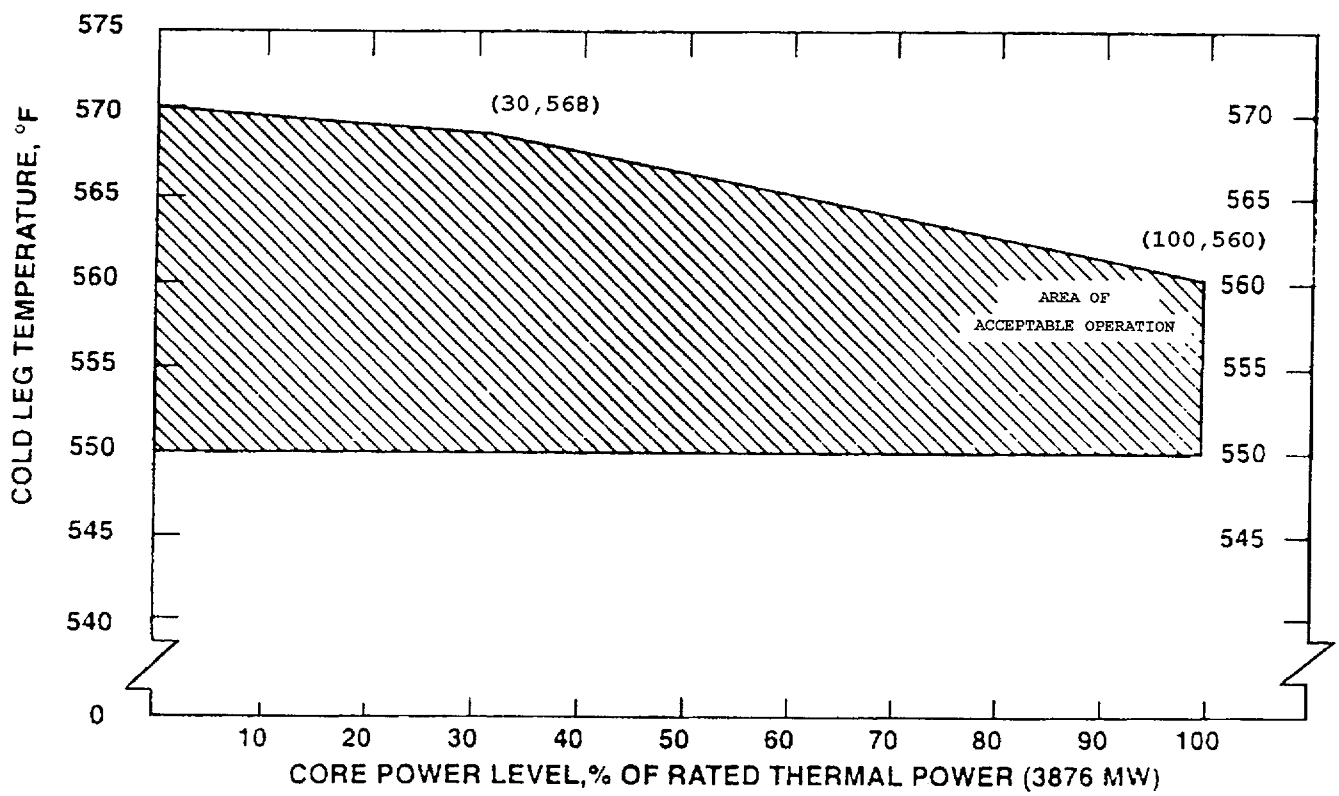
(a) The setpoint may be decreased to a minimum value of 100 psia, as pressurizer pressure is reduced, provided the margin between pressurizer pressure and the setpoint is maintained ≤ 400 psia or ≥ 140 psia greater than the saturation pressure of the RCS cold leg when the RCS cold leg temperature is ≥ 485°F. Trips may be bypassed when pressurizer pressure is < 400 psia. Bypass shall be automatically removed when pressurizer pressure is ≥ 500 psia. The setpoint shall be automatically increased to the normal setpoint as pressurizer pressure is increased.

(b) The setpoint may be decreased as steam pressure is reduced, provided the margin between steam pressure and the setpoint is maintained ≤ 200 psig. The setpoint shall be automatically increased to the normal setpoint as steam pressure is increased.

(c) The Main Steam Isolation Signal (MSIS) Function (Steam Generator Pressure – Low, Steam Generator Level-High and Containment Pressure – High signals) is not required to be OPERABLE when all associated valves isolated by the MSIS Function are closed.

Figure 3.4.1-1
Reactor Coolant Cold Leg Temperature vs. Core Power Level

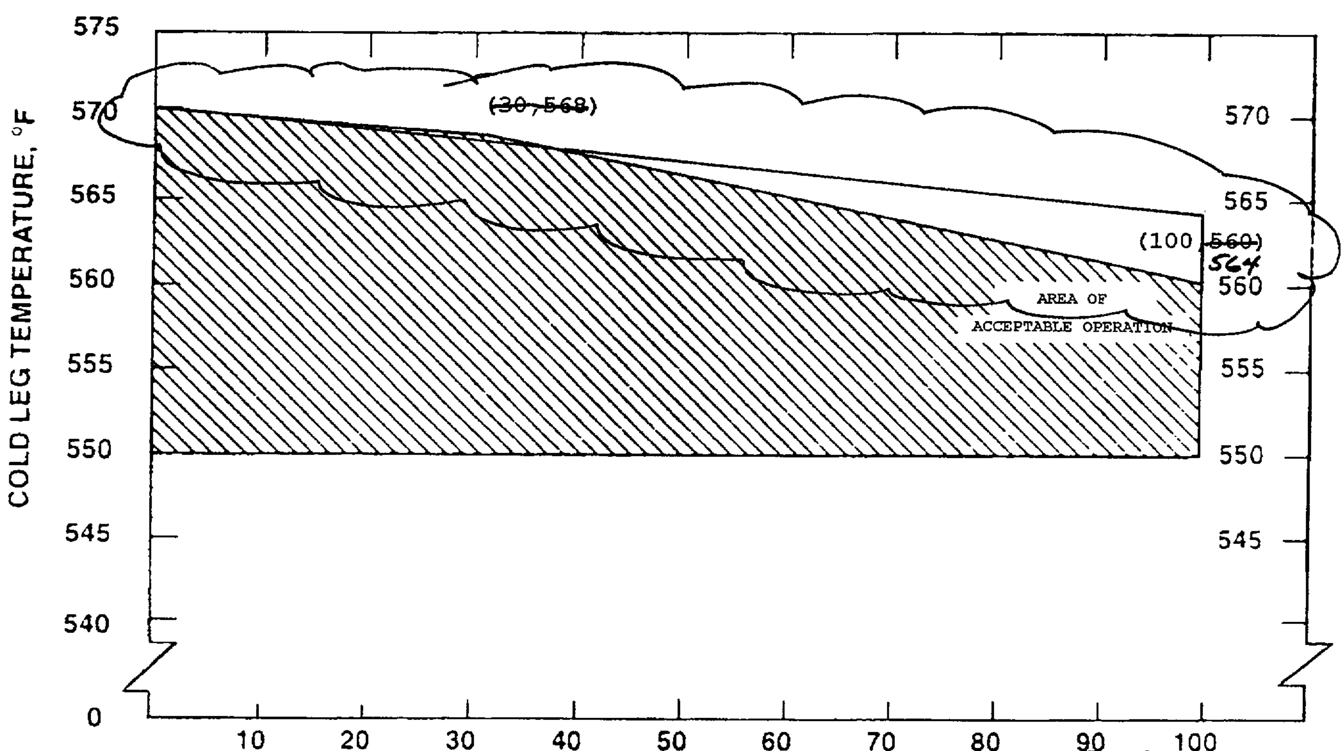
UNITS 1 AND 3



~~REACTOR COOLANT COLD LEG TEMPERATURE vs. CORE POWER LEVEL~~

Figure 3.4.1-1
Reactor Coolant Cold Leg Temperature vs. Core Power Level

UNIT 2



~~REACTOR COOLANT COLD LEG TEMPERATURE vs. CORE POWER LEVEL~~

Table 3.7.1-1 (page 1 of 1)
Variable Overpower Trip Setpoint versus
OPERABLE Main Steam Safety Valves

MINIMUM NUMBER OF MSSVs PER STEAM GENERATOR REQUIRED OPERABLE	MAXIMUM POWER (% RTP)		MAXIMUM ALLOWABLE VARIABLE OVERPOWER TRIP SETPOINT (% RTP)	
	UNITS 1 & 3	UNIT 2	UNITS 1 & 3	UNIT 2
10	100.0	100.0	111.0	111.0
9	98.2	90.0	108.0	99.7
8	87.3	80.0	97.1	89.7
7	76.4	68.0	86.2	77.7
6	65.5	56.0	75.3	65.7

5.5 Programs and Manuals (continued)

5.5.15 Safety Function Determination Program (continued)

b. A required system redundant to system(s) in turn supported by the inoperable supported system is also inoperable; or

c. A required system redundant to support system(s) for the supported systems (a) and (b) above is also inoperable.

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

When a loss of safety function is caused by the inoperability of a single Technical Specification support system, the appropriate Conditions and Required Actions to enter are those of the support system.

5.5.16 Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September, 1995, as modified by the following exceptions:

FOR UNITS 1 AND 3, AND 58.0 PSIG FOR UNIT 2

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 52.0 psig. The containment ~~design pressure is 60 psig.~~

The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.1 % of containment air weight per day.

Leakage Rate acceptance criteria are:

a. Containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance are $< 0.60\ L_a$ for the Type B and C tests and $\leq 0.75\ L_a$ for Type A tests.

(continued)

ATTACHMENT 4

RETYPED OPERATING LICENSE AND TECHNICAL SPECIFICATION PAGES

Attachment 4

Retyped Operating License and Technical Specification Pages

Unit 2 Operating License

Page 5

Technical Specifications

Page 1.1-6
Page 3.3.1-8
Page 3.3.2-5
Page 3.3.5-4
Page 3.4.1-3
Page 3.4.1-4
Page 3.7.1-3
Page 5.5-23

(8)(a) Arizona Public Service Company is authorized to transfer all or a portion of its 29.1% ownership share in Palo Verde, Unit 2 to certain equity investors identified in its submissions of August 6, August 8 and December 5, 1986, and at the same time to lease back from such purchasers such interest sold in the Palo Verde, Unit 2 facility. The term of the lease is for approximately 29-½ years subject to a right of renewal. Additional sale and leaseback transactions of all or a portion of APS's remaining ownership share in Palo Verde, Unit 2 are hereby authorized until June 30, 1987. Any such sale and lease back transaction is subject to the representations and conditions set forth in the aforementioned application of May 2, 1986, and the subsequent submittals dated July 30, August 2, August 6, August 7, August 8, August 13, October 16 and December 5, 1986, as well as the letters of the Director of the Office of Nuclear Reactor Regulation dated August 15, and December 11, 1986, consenting to such transactions. Specifically, the lessor and anyone else who may acquire an interest under this transaction are prohibited from exercising directly or indirectly any control over the licensees of the Palo Verde Nuclear Generating Station, Unit 2. For purposes of this condition the limitations in 10 CFR 50.81, "Creditor Regulations," as now in effect and as they may be subsequently amended, are fully applicable to the lessor and any successor in interest to the lessor as long as the license for Palo Verde, Unit 2 remains in effect; this financial transaction shall have no effect on the license for the Palo Verde nuclear facility throughout the term of the license.

(b) Further, the licensees are also required to notify the NRC in writing prior to any change in: (i) the terms or conditions of any lease agreements executed as part of this transaction; (ii) the ANPP Participation Agreement, (iii) the existing property insurance coverage for the Palo Verde nuclear facility, Unit 2 as specified in licensee counsel's letter of November 26, 1985, and (iv) any action by the lessor or others that may have an adverse effect on the safe operation of the facility.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) <u>Maximum Power Level</u>

Arizona Public Service Company (APS) is authorized to operate the facility at reactor core power levels not in excess of 3990 megawatts thermal (100% power) in accordance with the conditions specified herein. |

1.1 Definitions (continued)

RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 3876 MWt for Units 1 and 3, and 3990 MWt for Unit 2.
REACTOR PROTECTIVE SYSTEM (RPS) RESPONSE TIME	The RPS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RPS trip setpoint at the channel sensor until electrical power to the CEAs drive mechanism is interrupted. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC.
SHUTDOWN MARGIN (SDM)	SDM shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming: a. All full length CEAs (shutdown and regulating) are fully inserted except for the single CEA of highest reactivity worth, which is assumed to be fully withdrawn. With any full length CEAs not capable of being fully inserted, the withdrawn reactivity worth of these CEAs must be accounted for in the determination of SDM and b. There is no change in part length CEA position.

(continued)

Table 3.3.1-1 (page 1 of 3)
Reactor Protective System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Variable Over Power	1,2	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.6 SR 3.3.1.7 SR 3.3.1.8 SR 3.3.1.9 SR 3.3.1.13	Ceiling ≤ 111.0% RTP Band ≤ 9.9% RTP Incr. Rate ≤ 11.0%/min RTP Decr. Rate > 5%/sec RTP
2. Logarithmic Power Level – High[a]	2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.12 SR 3.3.1.13	≤ 0.011% NRTP
3. Pressurizer Pressure – High	1,2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.13	≤ 2388 psia
4. Pressurizer Pressure – Low	1,2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.12 SR 3.3.1.13	≥ 1821 psia
5. Containment Pressure – High	1,2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.13	≤ 3.2 psig
6. Steam Generator #1 Pressure – Low	1,2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.13	Units 1 and 3: ≥ 890 psia Unit 2: ≥ 955 psia
7. Steam Generator #2 Pressure – Low	1,2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.13	Units 1 and 3: ≥ 890 psia Unit 2: ≥ 955 psia

(continued)

(a) Trip may be bypassed when logarithmic power is > 1E-4% NRTP. Bypass shall be automatically removed when logarithmic power is ≤ 1E-4% NRTP.

Table 3.3.2-1
Reactor Protective System Instrumentation - Shutdown

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALVE
1. Logarithmic Power Level-High[d]	$3^{(a)}$, $4^{(a)}$, $5^{(a)}$	SR 3.3.2.1 SR 3.3.2.2 SR 3.3.2.3 SR 3.3.2.4 SR 3.3.2.5	≤ 0.011% NRTP[c]
2. Steam Generator #1 Pressure-Low[b]	$3^{(a)}$	SR 3.3.2.1 SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.5	Units 1 and 3: ≥ 890 psia Unit 2: ≥ 955 psia
3. Steam Generator #2 Pressure-Low[b]	$3^{(a)}$	SR 3.3.2.1 SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.5	Units 1 and 3: ≥ 890 psia Unit 2: ≥ 955 psia

(a) With any Reactor Trip Circuit Breakers (RTCBs) closed and any control element assembly capable of being withdrawn.

(b) The setpoint may be decreased as steam pressure is reduced, provided the margin between steam pressure and the setpoint is maintained ≤ 200 psig. The setpoint shall be automatically increased to the normal setpoint as steam pressure is increased.

(c) The setpoint must be reduced to ≤ 1E-4% NRTP when less than 4 RCPs are running.

(d) Trip may be bypassed when logarithmic power is > 1E-4% NRTP. Bypass shall be automatically removed when logarithmic power is ≤ 1E-4% NRTP.

Table 3.3.5-1 (page 1 of 1)
Engineered Safety Features Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	ALLOWABLE VALUE
1. Safety Injection Actuation Signal		
a. Containment Pressure – High	1,2,3	≤ 3.2 psig
b. Pressurizer Pressure – Low(a)		≥ 1821 psia
2. Containment Spray Actuation Signal		
a. Containment Pressure – High High	1,2,3	≤ 8.9 psig
3. Containment Isolation Actuation Signal		
a. Containment Pressure – High	1,2,3	≤ 3.2 psig
b. Pressurizer Pressure – Low(a)		≥ 1821 psia
4. Main Steam Isolation Signal(c)		
a. Steam Generator #1 Pressure–Low(b)	1,2,3	Units 1 and 3: ≥ 890 psia Unit 2: ≥ 955 psia
b. Steam Generator #2 Pressure–Low(b)		Units 1 and 3: ≥ 890 psia Unit 2: ≥ 955 psia
c. Steam Generator #1 Level-High		≤ 91.5%
d. Steam Generator #2 Level-High		≤ 91.5%
e. Containment Pressure-High		≤ 3.2 psig
5. Recirculation Actuation Signal		
a. Refueling Water Storage Tank Level–Low	1,2,3	≥ 6.9 and ≤ 7.9%
6. Auxiliary Feedwater Actuation Signal SG #1 (AFAS-1)		
a. Steam Generator #1 Level–Low	1,2,3	≥ 25.3%
b. SG Pressure Difference–High		≤ 192 psid
7. Auxiliary Feedwater Actuation Signal SG #2 (AFAS-2)		
a. Steam Generator #2 Level–Low	1,2,3	≥ 25.3%
b. SG Pressure Difference–High		≤ 192 psid

(a) The setpoint may be decreased to a minimum value of 100 psia, as pressurizer pressure is reduced, provided the margin between pressurizer pressure and the setpoint is maintained ≤ 400 psia or ≥ 140 psia greater than the saturation pressure of the RCS cold leg when the RCS cold leg temperature is ≥ 485°F. Trips may be bypassed when pressurizer pressure is < 400 psia. Bypass shall be automatically removed when pressurizer pressure is ≥ 500 psia. The setpoint shall be automatically increased to the normal setpoint as pressurizer pressure is increased.

(b) The setpoint may be decreased as steam pressure is reduced, provided the margin between steam pressure and the setpoint is maintained ≤ 200 psig. The setpoint shall be automatically increased to the normal setpoint as steam pressure is increased.

(c) The Main Steam Isolation Signal (MSIS) Function (Steam Generator Pressure – Low, Steam Generator Level-High and Containment Pressure – High signals) is not required to be OPERABLE when all associated valves isolated by the MSIS Function are closed.

Figure 3.4.1-1, (Page 1 of 2)
Reactor Coolant Cold Leg Temperature vs. Core Power Level

Units 1 and 3

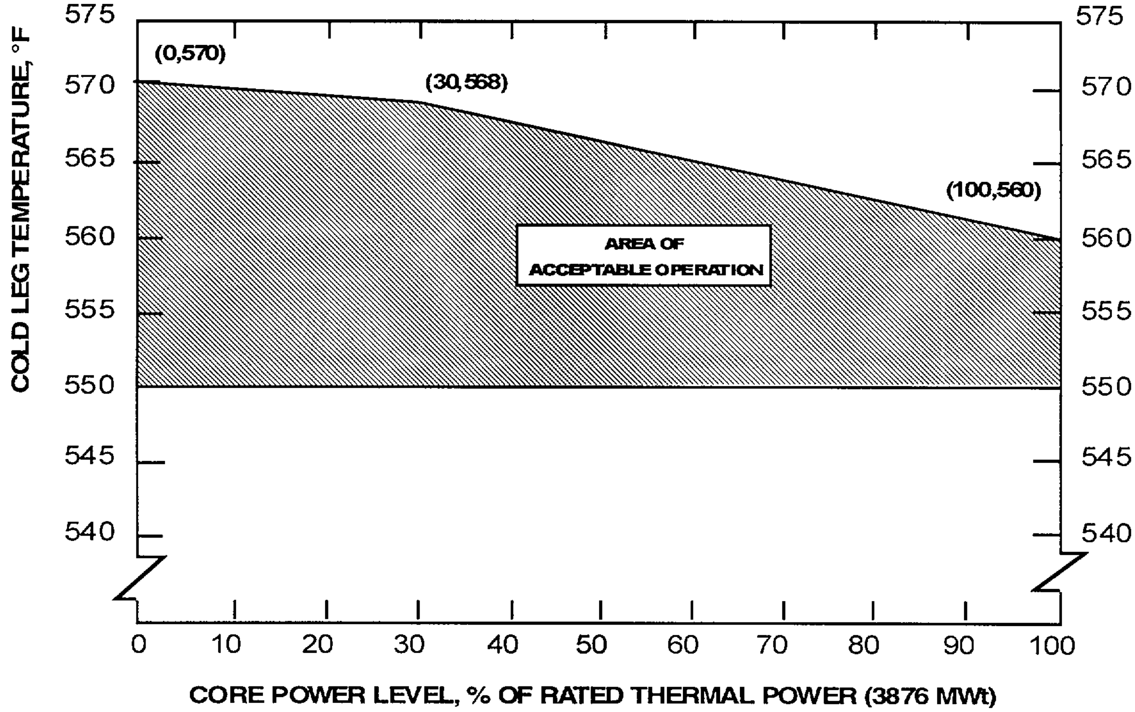


Figure 3.4.1-1, (Page 2 of 2)
Reactor Coolant Cold Leg Temperature vs. Core Power Level

Unit 2

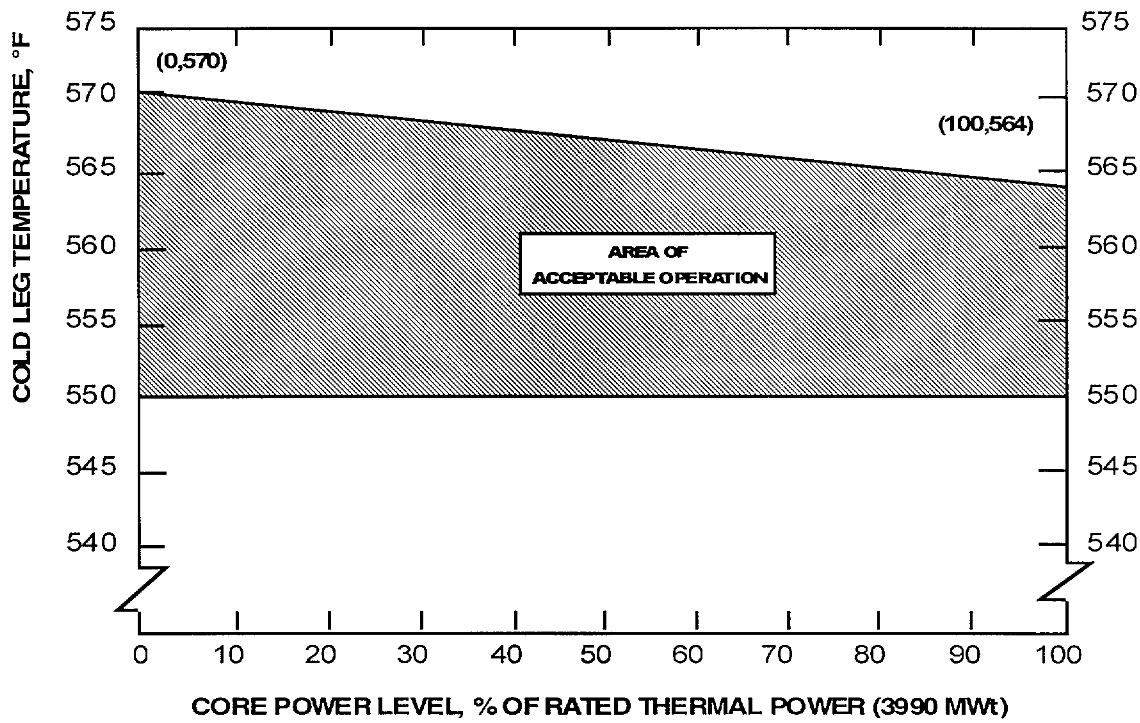


Table 3.7.1-1 (page 1 of 1)
Variable Overpower Trip Setpoint versus
OPERABLE Main Steam Safety Valves

MINIMUM NUMBER OF MSSVs PER STEAM GENERATOR REQUIRED OPERABLE	MAXIMUM POWER (% RTP)		MAXIMUM ALLOWABLE VARIABLE OVERPOWER TRIP SETPOINT (% RTP)	
	Units 1 and 3	Unit 2	Units 1 and 3	Unit 2
10	100.0	100.0	111.0	111.0
9	98.2	90.0	108.0	99.7
8	87.3	80.0	97.1	89.7
7	76.4	68.0	86.2	77.7
6	65.5	56.0	75.3	65.7

5.5 Programs and Manuals (continued)

5.5.15 Safety Function Determination Program (continued)

b. A required system redundant to system(s) in turn supported by the inoperable supported system is also inoperable; or

c. A required system redundant to support system(s) for the supported systems (a) and (b) above is also inoperable.

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

When a loss of safety function is caused by the inoperability of a single Technical Specification support system, the appropriate Conditions and Required Actions to enter are those of the support system.

5.5.16 Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September, 1995, as modified by the following exceptions:

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 52.0 psig for Units 1 and 3, and 58.0 psig for Unit 2. The containment design pressure is 60 psig.

The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.1 % of containment air weight per day.

Leakage Rate acceptance criteria are:

a. Containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance are $< 0.60\ L_a$ for the Type B and C tests and $\leq 0.75\ L_a$ for Type A tests.

(continued)

ATTACHMENT 5

ASSOCIATED CHANGES TO PVNGS TECHNICAL SPECIFICATION BASES

(Information Only)

Associated Changes to Technical Specification Bases

Bases

Page B 3.6.1-2
Page B 3.6.2-2
Page B 3.6.4-1
Page B 3.6.6-3
Page B3.7.1-1
Page B 3.7.1-2
Page B 3.7.1-3
Page B 3.7.1-4
Page B 3.7.1-3 (modified for Unit 2 only)
Page B 3.7.1-4 (modified for Unit 2 only)
Page B 3.7.1-5
Page B 3.7.1-6

BASES (continued)

BACKGROUND (continued)

2. closed by manual valves, blind flanges, or de-activated automatic valves secured in their closed positions, except as provided in LCO 3.6.3, "Containment Isolation Valves";

b. Each air lock is OPERABLE, except as provided in LCO 3.6.2, "Containment Air Locks"; and

c. All equipment hatches are closed.

APPLICABLE SAFETY ANALYSES

The safety design basis for the containment is that the containment must withstand the pressures and temperatures of the limiting DBA without exceeding the design leakage rate.

The DBAs that result in a release of radioactive material within containment are a Loss Of Coolant Accident (LOCA), a Main Steam Line Break (MSLB), a feedwater line break, and a control element assembly ejection accident (Ref. 2). In the analysis of each of these accidents, it is assumed that containment is OPERABLE such that release of fission products to the environment is controlled by the rate of containment leakage. The containment was designed with an allowable leakage rate of 0.1% of containment air mass per day (Ref. 3). This leakage rate is defined in 10 CFR 50, Appendix J, Option B (Ref. 1), as L_a: the maximum allowable containment leakage rate at the calculated maximum peak containment pressure (P_a) of 52.0 psig FOR UNITS 1 AND 3, AND 58.0 PSIG FOR UNIT 2, which results from the limiting design basis LOCA.

Satisfactory leakage rate test results are a requirement for the establishment of containment OPERABILITY.

The containment satisfies Criterion 3 of 10 CFR 50.36 (c)(2)(ii).

LCO

Containment OPERABILITY is maintained by limiting leakage to ≤ 1.0 L_a, except prior to the first startup after performing a required Containment Leakage Rate Testing Program leakage test. At this time, the applicable leakage limits must be met.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The DBAs that result in a release of radioactive material within containment are a Loss Of Coolant Accident (LOCA), a Main Steam Line Break (MSLB), a feedwater line break, and a control element assembly (CEA) ejection accident (Ref. 2). In the analysis of each of these accidents, it is assumed that containment is OPERABLE such that release of fission products to the environment is controlled by the rate of containment leakage. The containment was designed with an allowable leakage rate of 0.1% of containment air mass per day (Ref. 3). This leakage rate is defined in 10 CFR 50, Appendix J, Option B, as the maximum allowable containment leakage rate at the calculated peak containment internal pressure P_a [52.0 psig], following a design basis LOCA. This allowable leakage rate forms the basis for the acceptance criteria imposed on the SRs ~~associated with the~~ air lock.

FOR UNITS 1 AND 3, AND 58.0 PSIG FOR UNIT 2

The containment air locks satisfy Criterion ~~3 of~~ 10 CFR 50.36 (c)(2)(ii).

LCO

Each containment air lock forms part of the containment pressure boundary. As part of the containment pressure boundary, the air lock safety function is related to control of the containment leakage rate resulting from a DBA. Thus, each air lock's structural integrity and leak tightness are essential to the successful mitigation of such an event.

Each air lock is required to be OPERABLE. For the air lock to be considered OPERABLE, the air lock interlock mechanism must be OPERABLE, the air lock must be in compliance with the Type B air lock leakage test, and both air lock doors must be OPERABLE. The interlock allows only one air lock door of an air lock to be opened at one time. This provision ensures that a gross breach of containment does not exist when containment is required to be OPERABLE. Closure of a single door in each air lock is sufficient to provide a leak tight barrier following postulated events. Nevertheless, both doors are kept closed when the air lock is not being used for normal entry into or exit from containment.

(continued)

B 3.6 CONTAINMENT SYSTEMS

B 3.6.4 Containment Pressure

BASES

BACKGROUND

The containment pressure is limited during normal operation to preserve the initial conditions assumed in the accident analyses for a Loss Of Coolant Accident (LOCA) or Main Steam Line Break (MSLB). These limits also prevent the containment pressure from exceeding the containment design negative pressure differential with respect to the outside atmosphere in the event of inadvertent actuation of the Containment Spray System.

Containment pressure is a process variable that is monitored and controlled. The containment pressure limits are derived from the input conditions used in the containment functional analyses and the containment structure external pressure analysis. Should operation occur outside these limits coincident with a Design Basis Accident (DBA), post accident containment pressures could exceed calculated values.

APPLICABLE SAFETY ANALYSES

Containment internal pressure is an initial condition used in the DBA analyses to establish the maximum peak containment internal pressure. The limiting DBAs considered for determining the maximum containment internal pressure (P_a) are the LOCA and MSLB. A double ended discharge line break LOCA with maximum ECCS results in the highest calculated internal containment pressure of 52.0 psig, which is below the internal design pressure of 60 psig. The postulated DBAs are analyzed assuming degraded containment Engineered Safety Feature (ESF) Systems (i.e., assuming the loss of one ESF bus, which is the worst case single active failure, resulting in one train of the Containment Spray System being rendered inoperable). It is this maximum containment pressure that is used to ensure that the licensing basis dose limitations are met.

FOR UNITS 1 AND 3, AND 58.0 PSIG FOR UNIT 2

The initial pressure condition used in the containment analysis bounds the containment pressure allowed during normal operation. The LCO limit of 2.5 psig ensures that, in the event of an accident, the maximum peak containment internal pressure, 52.0 psig, and the maximum accident design pressure for containment, 60 psig, are not exceeded.

(continued)

BASES

BACKGROUND (continued)

The Containment Spray System accelerates the air mixing process between the upper dome space of the containment atmosphere during LOCA operations. It also prevents any hot spot air pockets during the containment cooling mode and avoids any hydrogen concentration in pocket areas.

APPLICABLE SAFETY ANALYSES

The Containment Spray System limits the temperature and pressure that could be experienced following a DBA. The Containment Spray System is required to be capable of reducing containment pressure to 1/2 the peak pressure within 24 hours following a DBA. The limiting DBAs considered relative to containment temperature and pressure are the Loss Of Coolant Accident (LOCA) and the Main Steam Line Break (MSLB). The DBA LOCA and MSLB are analyzed using computer codes designed to predict the resultant containment pressure and temperature transients. No DBAs are assumed to occur simultaneously or consecutively. The postulated DBAs are analyzed with regard to containment ESF systems, assuming the loss of one ESF bus, which is the worst case single active failure, resulting in one train of the Containment Spray System being rendered inoperable.

FOR UNITS 1 AND 3, AND 58.0 PSIG FOR UNIT 2

The analysis and evaluation show that under the worst case scenario, the highest peak containment pressure is 52.0 psig (experienced during a LOCA). The analysis shows that the peak containment vapor temperature is 405.65°F (experienced during a MSLB). Both results are within the design. (See the Bases for Specifications 3.6.4, "Containment Pressure," and 3.6.5, "Containment Air Temperature," for a detailed discussion.) The analyses and evaluations assume a power level of 102% RTP, one containment spray train operating, and initial (pre-accident) conditions of 120°F and 16.7 psia (LOCA) and 13.22 psia (MSLB). The analyses also assume a response time delayed initiation in order to provide a conservative calculation of peak containment pressure and temperature responses.

The effect of an inadvertent containment spray actuation has been analyzed. An inadvertent spray actuation reduces the containment pressure to -2.6 psig due to the sudden cooling effect in the interior of the air tight containment. Additional discussion is provided in the Bases for Specification 3.6.4.

(continued)

NO CHANGES ON THIS PAGE
INCLUDED FOR CONTINUITY

B 3.7 PLANT SYSTEMS

B 3.7.1 Main Steam Safety Valves (MSSVs)

BASES

BACKGROUND

The primary purpose of the MSSVs is to provide overpressure protection for the secondary system. The MSSVs also provide protection against overpressurizing the Reactor Coolant Pressure Boundary (RCPB) by providing a heat sink for the removal of energy from the Reactor Coolant System (RCS) if the preferred heat sink, provided by the Condenser and Circulating Water System, is not available.

Five MSSVs are located on each of the four main steam lines, outside containment, upstream of the main steam isolation valves, as described in the UFSAR, Section 5.2 (Ref. 1). The MSSV rated capacity passes the full steam flow at 102% RTP (100% + 2% for instrument error) with the valves full open. This meets the requirements of the ASME Code, Section III (Ref. 2). The MSSV design includes staggered setpoints, according to Table 3.7.1-2, in the accompanying LCO, so that only the number of valves needed will actuate. Staggered setpoints reduce the potential for valve chattering if there is insufficient steam pressure to fully open all valves.

APPLICABLE SAFETY ANALYSES

The design basis for the MSSVs comes from Reference 2; its purpose is to limit secondary system pressure to ≤ 110% of design pressure when passing 100% of design steam flow. This design basis is sufficient to cope with any Anticipated Operational Occurrence (AOO) or accident considered in the Design Basis Accident (DBA) and transient analysis.

The events that challenge the MSSV relieving capacity, and thus RCS pressure, are those characterized as decreased heat removal events, and are presented in the FSAR, Section 15.2 (Ref. 3). Of these, the full power Loss Of Condenser Vacuum (LOCV) event is the limiting AOO. An LOCV isolates the turbine and condenser, and terminates normal feedwater flow to the steam generators. Before delivery of auxiliary feedwater to the steam generators, RCS pressure reaches ≤ 2742 psia. This peak pressure is < 110% of the design pressure of 2500 psia, but high enough to actuate the pressurizer safety valves.

(continued)

NO CHANGES ON THIS PAGE
INCLUDED FOR CONTINUITY

BASES

APPLICABLE SAFETY ANALYSES (continued)

The limiting accident for peak RCS pressure is the full power feedwater line break (FWLB), inside containment, with the failure of the backflow check valve in the feedwater line from the affected steam generator. Water from the affected steam generator is assumed to be lost through the break with minimal additional heat transfer from the RCS. With heat removal limited to the unaffected steam generator, the reduced heat transfer causes an increase in RCS temperature, and the resulting RCS fluid expansion causes an increase in pressure. The RCS pressure increases to ≤ 2843 psia, with the pressurizer safety valves providing relief capacity. These results were found acceptable by the NRC based on the low probability of the event.

The MSSVs satisfy Criterion 3 of 10CFR 50.36 (c)(2)(ii).

LCO

This LCO requires all MSSVs to be OPERABLE in compliance with Reference 2, even though this is not a requirement of the DBA analysis. This is because operation with less than the full number of MSSVs requires limitations on allowable THERMAL POWER (to meet Reference 2 requirements), and adjustment to the Reactor Protection System trip setpoints. These limitations are according to those shown in Table 3.7.1-1 and Required Action A.2 in the accompanying LCO. An MSSV is considered inoperable if it fails to open upon demand.

The OPERABILITY of the MSSVs is defined as the ability to open within the setpoint tolerances, relieve steam generator overpressure, and reseat when pressure has been reduced. The OPERABILITY of the MSSVs is determined by periodic surveillance testing in accordance with the Inservice Testing Program.

The lift settings, according to Table 3.7.1-2 in the accompanying LCO, correspond to ambient conditions of the valve at nominal operating temperature and pressure.

This LCO provides assurance that the MSSVs will perform their designed safety function to mitigate the consequences of accidents that could result in a challenge to the RCPB.

(continued)

BASES

APPLICABILITY

In MODES 1, 2 and 3, a minimum of six MSSVs per steam generator are required to be OPERABLE, according to Table 3.7.1-1 in the accompanying LCO, which is limiting and bounds all lower MODES.

In MODES 4 and 5, there are no credible transients requiring the MSSVs.

The steam generators are not normally used for heat removal in MODES 5 and 6, and thus cannot be overpressurized; there is no requirement for the MSSVs to be OPERABLE in these MODES.

ACTIONS

The ACTIONS table is modified by a Note indicating that separate Condition entry is allowed for each MSSV.

A.1 and A.2 — UNITS 1 AND 3 ONLY

When 10 MSSVs are OPERABLE per steam generator, THERMAL POWER is limited to 100% RTP per the Operating Licenses, and the VOPT allowable trip setpoint is limited to 111.0% RTP per TS Table 3.3.1-1.

An alternative to restoring inoperable MSSV(s) to OPERABLE status is to reduce power so that the available MSSV relieving capacity meets Code requirements for the power level. Operation may continue provided the allowable THERMAL POWER is equal to the product of: 1) the ratio of the number of MSSVs available per steam generator to the total number of MSSVs per steam generator, and 2) the ratio of the available relieving capacity to total steam flow, multiplied by 100%.

(continued)

BASES

ACTIONS

A.1 and A.2 (continued) - UNITS 1 AND 3 ONLY

Allowable THERMAL POWER = $\left(\frac{10 - N}{10}\right) \times 109.2$

With one or more MSSVs inoperable, the ceiling on the variable overpower trip is reduced to an amount over the allowable THERMAL POWER equal to the band given for this trip, according to Table 3.7.1-1 in the accompanying LCO.

SP = Allowable THERMAL POWER + 9.8

where:

SP = Reduced reactor trip setpoint in percent RTP. This is a ratio of the available relieving capacity over the total steam flow at rated power.

10 = Total number of MSSVs per steam generator.

N = Number of inoperable MSSVs on the steam generator with the greatest number of inoperable valves.

109.2 = Ratio of MSSV relieving capacity at 110% steam generator design pressure to calculated steam flow rate at 100% RTP + 2% instrument uncertainty expressed as a percentage (see text above).

9.8 = Band between the maximum THERMAL POWER and the variable overpower trip setpoint ceiling (Table 3.7.1-1).

The operator should limit the maximum steady state power level to the value determined from Table 3.7.1-1 to avoid an inadvertent overpower trip.

The Completion Time of 12 hours for Required Action A.2 is based on operating experience in resetting all channels of a protective function and on the low probability of the occurrence of a transient that could result in steam generator overpressure during this period.

(continued)

BASES

~~APPLICABILITY~~

~~In MODES 1, 2 and 3, a minimum of six MSSVs per steam generator are required to be OPERABLE, according to Table 3.7.1-1 in the accompanying LCO, which is limiting and bounds all lower MODES.~~

~~In MODES 4 and 5, there are no credible transients requiring the MSSVs.~~

~~The steam generators are not normally used for heat removal in MODES 5 and 6, and thus cannot be overpressurized; there is no requirement for the MSSVs to be OPERABLE in these MODES.~~

~~ACTIONS~~

~~The ACTIONS table is modified by a Note indicating that separate Condition entry is allowed for each MSSV.~~

A.1 and A.2 — UNIT 2 ONLY

When 10 MSSVs are OPERABLE per steam generator, THERMAL POWER is limited to 100% RTP per the Operating Licenses, and the VOPT allowable trip setpoint is limited to 111.0% RTP per TS Table 3.3.1-1.

An alternative to restoring inoperable MSSV(s) to OPERABLE status is to reduce power ~~so that the available MSSV relieving capacity meets Code requirements for the power level. Operation may continue provided the allowable THERMAL POWER is equal to the product of: 1) the ratio of the number of MSSVs available per steam generator to the total number of MSSVs per steam generator, and 2) the ratio of the available relieving capacity to total steam flow, multiplied by 100%.~~

IN ACCORDANCE WITH TABLE 3.7.1-1. THESE REDUCED POWER LEVELS, DERIVED FROM THE TRANSIENT ANALYSIS, COMPENSATE FOR DEGRADED RELIEVING CAPACITY AND ENSURE THAT THE RESULTS OF THE TRANSIENT ANALYSIS ARE ACCEPTABLE.

(continued)

BASES

ACTIONS

A.1 and A.2 (continued) — UNIT 2 ONLY

~~Allowable THERMAL POWER = (10 - N) x 109.2 / 10~~

~~With one or more MSSVs inoperable, the ceiling on the variable overpower trip is reduced to an amount over the allowable THERMAL POWER equal to the band given for this trip, according to Table 3.7.1-1 in the accompanying LCO.~~

~~SP = Allowable THERMAL POWER + 9.8~~

~~where:~~

~~SP = Reduced reactor trip setpoint in percent RTP. This is a ratio of the available relieving capacity over the total steam flow at rated power.~~

~~10 = Total number of MSSVs per steam generator.~~

~~N = Number of inoperable MSSVs on the steam generator with the greatest number of inoperable valves.~~

~~109.2 = Ratio of MSSV relieving capacity at 110% steam generator design pressure to calculated steam flow rate at 100% RTP + 2% instrument uncertainty expressed as a percentage (see text above).~~

~~9.8 = Band between the maximum THERMAL POWER and the variable overpower trip setpoint ceiling (Table 3.7.1-1).~~

The operator should limit the maximum steady state power level to the value determined from Table 3.7.1-1 to avoid an inadvertent overpower trip.

The Completion Time of 12 hours for Required Action A.2 is based on operating experience in resetting all channels of a protective function and on the low probability of the occurrence of a transient that could result in steam generator overpressure during this period.

(continued)

BASES

ACTIONS (continued)

B.1 and B.2

If the MSSVs cannot be restored to OPERABLE status in the associated Completion Time, or if one or more steam generators have less than six MSSVs OPERABLE, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.1.1

This SR verifies the OPERABILITY of the MSSVs by the verification of each MSSV lift setpoints in accordance with the Inservice Testing Program. The ASME Code, Section XI (Ref. 4), requires that safety and relief valve tests be performed in accordance with ANSI/ASME OM-1-1987 (Ref. 5). According to Reference 5, the following tests are required for MSSVs:

a. Visual examination;

b. Seat tightness determination;

c. Setpoint pressure determination (lift setting);

d. Compliance with owner's seat tightness criteria; and

e. Verification of the balancing device integrity on balanced valves.

The ASME Standard requires that all valves be tested every 5 years, and a minimum of 20% of the valves tested every 24 months. The ASME Code specifies the activities and frequencies necessary to satisfy the requirements. Table 3.7.1-2 allows a ± 3% setpoint tolerance for OPERABILITY; however, the valves are reset to ± 1% during the Surveillance to allow for drift.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.1.1 (continued)

This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR. This is to allow testing of the MSSVs at hot conditions. The MSSVs may be either bench tested or tested in situ at hot conditions using an assist device to simulate lift pressure. If the MSSVs are not tested at hot conditions, the lift setting pressure shall be corrected to ambient conditions of the valve at operating temperature and pressure.

REFERENCES

1. UFSAR, Section 5.2.
2. ASME, Boiler and Pressure Vessel Code, Section III, Article NC-7000, Class 2 Components.
3. UFSAR, Section 15.2.
4. ASME, Boiler and Pressure Vessel Code, Section XI, Subsection IWV.
5. ANSI/ASME OM-1-1987.

Attachment 6

ATTACHMENT 6

POWER UPRATE LICENSING REPORT