September 1, 1995

Mr. William L. Stewart
Executive Vice President, Nuclear
Arizona Public Service Company
Post Office Box 53999
Phoenix, Arizona 85072-3999

SUBJECT: ISSUANCE OF AMENDMENTS FOR THE PALO VERDE NUCLEAR GENERATING STATION UNIT NO. 1 (TAC NO. M92019), UNIT NO. 2 (TAC NO. M92020), AND UNIT NO. 3 (TAC NO. M92021)

Dear Mr. Stewart:

The Commission has issued the enclosed Amendment No. 98 to Facility Operating License No. NPF-41, Amendment No. 86 to Facility Operating License No. NPF-51, and Amendment No. 69 to Facility Operating License No. NPF-74 for the Palo Verde Nuclear Generating Station, Unit Nos. 1, 2, and 3, respectively. The amendments consist of changes to the Technical Specifications in response to your application dated March 31, 1995.

These amendments would clarify the shutdown margin definition, change the shutdown margin applicability and surveillance requirements to comply with safety analysis assumptions for subcritical inadvertent control element assembly withdrawal (UFSAR Section 15.4), and expand the applicability for core protection calculator (CPC) operability. In addition, the proposed amendment would add a reference to the Core Operating Limits Report for the MODE 6 refueling boron concentration limit. The amendment would also change the power calibration requirements for the linear power level, the CPC delta T power, and CPC nuclear power signals to allow more conservative settings than previously requested.

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

Sincerely,
Original Signed By
Charles R. Thomas, Project Manager
Project Directorate IV-2
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

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

Enclosures:
1. Amendment No. 98 to NPF-41
2. Amendment No. 86 to NPF-51
3. Amendment No. 69 to NPF-74
4. Safety Evaluation

cc w/encls: See next page

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CGrimes, O11E22	ACRS (4), T2E26
Region IV	BHolian
CThomas	

9509080322 950901
PDR ADOCK 05000528
P PDR

DOCUMENT NAME: PV92019.AMD

OFC	PDIV-2/LA	PDIV-2/PM	PDIV-2/PM	SRXB	OGC
NAME	EPeyton	CThomas:pk	BHolian	RJones	
DATE	7/13/95	7/13/95	7/21/95	7/17/95	8/11/95

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Mr. William L. Stewart
Executive Vice President, Nuclear
Arizona Public Service Company
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LHurley, RIV	

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OFC	PDIV-2/LA	PDIV-2/PM	PDIV-2/PM	SRXB	OGC
NAME	EPeyton	CThomas:pk	BHolian	RJones	
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

September 1, 1995

Mr. William L. Stewart
Executive Vice President, Nuclear
Arizona Public Service Company
Post Office Box 53999
Phoenix, Arizona 85072-3999

SUBJECT: ISSUANCE OF AMENDMENTS FOR THE PALO VERDE NUCLEAR GENERATING STATION UNIT NO. 1 (TAC NO. M92019), UNIT NO. 2 (TAC NO. M92020), AND UNIT NO. 3 (TAC NO. M92021)

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

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cc w/encls: See next page

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

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. 98
License No. NPF-41

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

   A. The application for amendment by the Arizona Public Service Company (APS or the licensee) 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 dated March 31, 1995, 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 the Act, and the rules and 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; and

   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 attachment to this license amendment, and paragraph 2.C(2) of Facility Operating License No. NPF-41 is hereby amended to read as follows:

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

The Technical Specifications contained in Appendix A, as revised through Amendment No. 98, 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, except where otherwise stated in specific license conditions.

3. The license amendment is effective as of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Charles R. Thomas

Charles R. Thomas, Project Manager
Project Directorate IV-2
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: September 1, 1995

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 98 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. The corresponding overleaf pages are also provided to maintain document completeness.

REMOVE	INSERT
III*	III*
IV	IV
1-6	1-6
3/4 1-1	4/4 1-1
3/4 1-2	3/4 1-2
3/4 1-3	3/4 1-3
3/4 3-3	3/4 3-3
3-7*	3-7*
3/4 3-8	3/4 3-8
3/4 3-16	3/4 3-16
--	3/4 3-16a
3/4 9-1	3/4 9-1
B 3/4 1-1	B 3/4 1-1
B 3/4 1-1a	B 3/4 1-1a
B 3/4 9-1	B 3/4 9-1
B 3/4 9-2*	B 3/4 9-2*
6-20a	6-20a

*No changes were made to these pages, reissued to become overleaf pages.

SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

BASES

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

REPORTABLE EVENT

1.28 A REPORTABLE EVENT shall be any of those conditions specified in Sections 50.72 and 50.73 to 10 CFR Part 50.

SHUTDOWN MARGIN

1.29 SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming:

a. No change in part-length control element assembly position, and

b. All full-length control element assemblies (shutdown and regulating) are fully inserted except for the single assembly 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 full-length CEAs must be accounted for in the determination of the SHUTDOWN MARGIN.

SITE BOUNDARY

1.30 The SITE BOUNDARY shall be that line beyond which the land is neither owned, nor leased, nor otherwise controlled by the licensee.

SOFTWARE

1.31 The digital computer SOFTWARE for the reactor protection system shall be the program codes including their associated data, documentation, and procedures.

SOURCE CHECK

1.32 A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to a source of increased radioactivity.

STAGGERED TEST BASIS

1.33 A STAGGERED TEST BASIS shall consist of:

a. A test schedule for n systems, subsystems, trains, or other designated components obtained by dividing the specified test interval into n equal subintervals, and

b. The testing of one system, subsystem, train, or other designated component at the beginning of each subinterval.

THERMAL POWER

1.34 THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

REACTIVITY CONTROL SYSTEMS

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - REACTOR TRIP BREAKERS OPEN**

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be greater than or equal to 1.0% delta k/k.

APPLICABILITY: MODES 3, 4*, and 5* with the reactor trip breaker open**.

ACTION:

With the SHUTDOWN MARGIN less than 1.0% delta k/k, immediately initiate and continue boration at greater than or equal to 26 gpm to reactor coolant system of a solution containing greater than or equal to 4000 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 1.0% delta k/k at least once per 24 hours by consideration of at least the following factors:

1. Reactor Coolant System boron concentration,
2. CEA position,
3. Reactor Coolant System average temperature,
4. Fuel burnup based on gross thermal energy generation,
5. Xenon concentration, and
6. Samarium concentration.

4.1.1.1.2 The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within ± 1.0% delta k/k at least once per 31 Effective Full Power Days (EFPD). This comparison shall consider at least those factors stated in Specification 4.1.1.1.1, above. The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 EFPD after each fuel loading.

4.1.1.1.3 With the reactor trip breakers open** and any CEA(s) fully or partially withdrawn, the SHUTDOWN MARGIN shall be verified within one hour after detection of the withdrawn CEA(s) and at least once per 12 hours thereafter while the CEA(s) are withdrawn.

* See Special Test Exception 3.10.9.
**The CEA drive system not capable of CEA withdrawal.

SHUTDOWN MARGIN - REACTOR TRIP BREAKERS CLOSED**

LIMITING CONDITION FOR OPERATION

3.1.1.2

a. The SHUTDOWN MARGIN shall be greater than or equal to that specified in the CORE OPERATING LIMITS REPORT, and

b. For T_{cold} less than or equal to 500°F, K_{N-1} shall be less than 0.99.

c. Reactor criticality shall not be achieved with shutdown group CEA movement.

APPLICABILITY: MODES 1, 2*, 3*, 4*, and 5* with the reactor trip breakers closed.**

ACTION:

a. With the SHUTDOWN MARGIN less than that specified in the CORE OPERATING LIMITS REPORT, immediately initiate and continue boration at greater than or equal to 26 gpm to the reactor coolant system of a solution containing greater than or equal to 4000 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored, and

b. With T_{cold} less than or equal to 500°F and K_{N-1} greater than or equal to 0.99, immediately vary CEA positions and/or initiate and continue boration at greater than or equal to 26 gpm to the reactor coolant system of a solution containing greater than or equal to 4000 ppm boron or equivalent until the required K_{N-1} is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.2.1 With the reactor trip breakers closed**, the SHUTDOWN MARGIN shall be determined to be greater than or equal to that specified in the CORE OPERATING LIMITS REPORT:

a. Within 1 hour after detection of an inoperable CEA(s) and at least once per 12 hours thereafter while the CEA(s) is inoperable.

* See Special Test Exceptions 3.10.1 and 3.10.9
**The CEA drive system capable of CEA withdrawal.

b. When in MODE 1 or MODE 2 with k_{eff} greater than or equal to 1.0, at least once per 12 hours by verifying that CEA group withdrawal is within the Transient Insertion Limits of Specification 3.1.3.6. If CEA group withdrawal is not within the Transient Insertion Limits of Specification 3.1.3.6, within 1 hour verify that SHUTDOWN MARGIN is greater than or equal to that specified in the CORE OPERATING LIMITS REPORT.

c. When in MODE 2 with k_{eff} less than 1.0, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical CEA position is within the limits of Specification 3.1.3.6.

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of e. below, with the CEA groups at the Transient Insertion Limits of Specification 3.1.3.6.

e. When in MODE 3, 4, or 5, at least once per 24 hours by consideration of at least the following factors:

1. Reactor Coolant System boron concentration,
2. CEA position,
3. Reactor Coolant System average temperature,
4. Fuel burnup based on gross thermal energy generation,
5. Xenon concentration, and
6. Samarium concentration.

4.1.1.2.2 When in MODE 3, 4, or 5, with the reactor trip breakers closed**. and T_{cold} less than or equal to 500°F, K_{N-1} shall be determined to be less than 0.99 at least once per 24 hours by consideration of at least the following factors:

1. Reactor Coolant System boron concentration,
2. CEA position,
3. Reactor Coolant System average temperature,
4. Fuel burnup based on gross thermal energy generation,
5. Xenon concentration, and
6. Samarium concentration.

4.1.1.2.3 When in MODES 3, 4, or 5 with the reactor trip breakers closed**, verify that criticality cannot be achieved with shutdown group CEA withdrawal at least once per 24 hours by consideration of at least the following factors:

1. Reactor Coolant System boron concentration,
2. CEA position,
3. Reactor Coolant System average temperature,
4. Fuel burnup based on gross thermal energy generation,
5. Xenon concentration, and
6. Samarium concentration.

4.1.1.2.4 The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within ± 1.0% delta k/k at least once per 31 Effective Full Power Days (EFPD). This comparison shall consider at least those factors stated in Specification 4.1.1.2.1.e or 4.1.1.2.2. The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 EFPD after each fuel loading.

**The CEA drive system capable of CEA withdrawal.

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.3 The moderator temperature coefficient (MTC) shall be within the area of Acceptable Operation specified in the CORE OPERATING LIMITS REPORT. The maximum upper limit shall be less than or equal to $+ 0.5 \times 10^{-4}$ ΔK/K°F for a power level of 0% RATED THERMAL POWER with a linear ramp to 0ΔK/K/°F at 100% RATED THERMAL POWER.

APPLICABILITY: MODES 1 and 2*#.

ACTION:

With the moderator temperature coefficient outside the area of Acceptable Operation, be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.1.3.1 The MTC shall be determined to be within its limits by confirmatory measurements. MTC measured values shall be extrapolated and/or compensated to permit direct comparison with the above limits.

4.1.1.3.2 The MTC shall be determined at the following frequencies and THERMAL POWER conditions during each fuel cycle:

a. Prior to initial operation above 5% of RATED THERMAL POWER, after each fuel loading.

b. At any THERMAL POWER, within 7 EFPD after reaching a core average exposure of 40 EFPD burnup into the current cycle.

c. At any THERMAL POWER, within 7 EFPD after reaching a core average exposure equivalent to two-thirds of the expected current cycle end-of-cycle core average burnup.

*With K_{eff} greater than or equal to 1.0.

#See Special Test Exception 3.10.2.

TABLE 3.3-1

REACTOR PROTECTIVE INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. TRIP GENERATION					
A. Process					
1. Pressurizer Pressure - High	4	2	3	1,2	2#,3#
2. Pressurizer Pressure - Low	4	2 (b)	3	1,2	2#,3#
3. Steam Generator Level - Low	4/SG	2/SG	3/SG	1,2	2#,3#
4. Steam Generator Level - High	4/SG	2/SG	3/SG	1,2	2#,3#
5. Steam Generator Pressure - Low	4/SG	2/SG	3/SG	1,2,3*,4*	2#,3#
6. Containment Pressure - High	4	2	3	1,2	2#,3#
7. Reactor Coolant Flow - Low	4/SG	2/SG	3/SG	1,2	2#,3#
8. Local Power Density - High	4	2 (c)(d)	3	1,2	2#,3#
9. DNBR - Low	4	2 (c)(d)	3	1,2	2#,3#
B. Excore Neutron Flux					
1. Variable Overpower Trip	4	2	3	1,2	2#,3#
2. Logarithmic Power Level - High					
a. Startup and Operating	4	2(a)(d)	3	1,2	2#,3#
	4	2	3	3*,4*,5*	9
b. Shutdown	4	0	2	3,4,5	4
C. Core Protection Calculator System					
1. CEA Calculators	2	1	2 (e)	1,2	6,7
2. Core Protection Calculators	4	2 (c)(d)	3	1,2,3*,4*,5*	2#,3#,7,10

TABLE 3.3-1 (Continued)

ACTION STATEMENTS

3.	Steam Generator Pressure - Low	Steam Generator Pressure - Low Steam Generator Level 1-Low (ESF) Steam Generator Level 2-Low (ESF)
4.	Steam Generator Level - Low (Wide Range)	Steam Generator Level - Low (RPS) Steam Generator Level 1-Low (ESF) Steam Generator Level 2-Low (ESF)
5.	Core Protection Calculator	Local Power Density - High (RPS) DNBR - Low (RPS)

STARTUP and/or POWER OPERATION may continue until the performance of the next required CHANNEL FUNCTIONAL TEST. Subsequent STARTUP and/or POWER OPERATION may continue if one channel is restored to OPERABLE status and the provisions of ACTION 2 are satisfied.

ACTION 4 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, suspend all operations involving positive reactivity changes.

ACTION 5 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, STARTUP and/or POWER OPERATION may continue provided the reactor trip breaker of the inoperable channel is placed in the tripped condition within 1 hour, otherwise, be in at least HOT STANDBY within 6 hours; however, the trip breaker associated with the inoperable channel may be closed for up to 1 hour for surveillance testing per Specification 4.3.1.1.

ACTION 6 -

a. With one CEAC inoperable, operation may continue for up to 7 days provided that the requirements of Specification 4.1.3.1.1 are met. After 7 days, operation may continue provided that the conditions of Action Item 6.b are met.

b. With both CEACs inoperable, operation may continue provided that:

1. Within 1 hour the DNBR margin required by Specification 3.2.4.b (COLSS in service) or 3.2.4.d (COLSS out of service) is satisfied and the Reactor Power Cutback System is disabled, and

TABLE 3.3-1 (Continued)

ACTION STATEMENTS

2. Within 4 hours:

    a) All full-length and part-length CEA groups must be withdrawn within the limits of Specifications 3.1.3.5, 3.1.3.6b, and 3.1.3.7b except during surveillance testing pursuant to the requirements of Specification 4.1.3.1.2. Specification 3.1.3.6b allows CEA group 5 insertion to no further than 127.5 inches withdrawn.

    b) The "RSPT/CEAC Inoperable" addressable constant in the CPCs is set to be indicated that both CEAC's are inoperable.

    c) The Control Element Drive Mechanism Control System (CEDMCS) is placed in and subsequently maintained in the "Standby" mode except during CEA motion permitted by Specifications 3.1.3.5, 3.1.3.6b, and 3.1.3.7b when the CEDMCS may be operated in either the "Manual Group" or "Manual Individual" mode.

3. CEA position surveillance must meet the requirements of Specifications 4.1.3.1.1, 4.1.3.5, 4.1.3.6, and 4.1.3.7 except during surveillance testing pursuant to Specification 4.1.3.1.2.

ACTION 7 - With three or more auto restarts, excluding periodic auto restarts (Code 30 and Code 33), of one non-bypassed calculator during a 12-hour interval, demonstrate calculator OPERABILITY by performing a CHANNEL FUNCTIONAL TEST within the next 24 hours.

ACTION 8 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore an inoperable channel to OPERABLE status within 48 hours or open an affected reactor trip breaker within the next hour.

ACTION 9 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or open the reactor trip breakers within the next hour.

ACTION 10 - In MODES 3, 4, or 5, the Core Protection Calculator channels are not required to be OPERABLE when the Logarithmic Power Level - High trip is OPERABLE with the trip setpoint lowered to $\leq 10^{-4}\%$ of Rated Thermal Power.

TABLE 4.3-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
D. Supplementary Protection System				
Pressurizer Pressure - High	S	R	Q	1, 2
II. RPS LOGIC				
A. Matrix Logic	N.A.	N.A.	Q	1, 2, 3*, 4*, 5*
B. Initiation Logic	N.A.	N.A.	Q	1, 2, 3*, 4*, 5*
III. RPS ACTUATION DEVICES				
A. Reactor Trip Breakers	N.A.	N.A.	M, R (10)	1, 2, 3*, 4*, 5*
B. Manual Trip	N.A.	N.A.	Q	1, 2, 3*, 4*, 5*

TABLE 4.3-1 (Continued)

TABLE NOTATIONS

* - With reactor trip breakers in the closed position and the CEA drive system capable of CEA withdrawal, and fuel in the reactor vessel.

(1) - Each STARTUP or when required with the reactor trip breakers closed and the CEA drive system capable of rod withdrawal, if not performed in the previous 7 days.

(2) - Heat balance only (CHANNEL FUNCTIONAL TEST not included):

a. Between 15% and 80% of RATED THERMAL POWER, compare the linear power level, the CPC delta T power and the CPC nuclear power signals to the calorimetric calculation.

If any signal is within -0.5% to 10% of the calorimetric then <u>do not</u> calibrate except as required during initial power ascension after refueling.

If any signal is less than the calorimetric calculation by more than 0.5%, then adjust the affected signal(s) to agree with the calorimetric calculation.

If any signal is greater than the calorimetric calculation by more than 10% then adjust the affected signal(s) to agree with the calorimetric calculation within 8% to 10%.

b. At or above 80% of RATED THERMAL POWER; compare the linear power level, the CPC delta T power and the CPC nuclear power signals to the calorimetric calculation. If any signal differs from the calorimetric calculation by an absolute difference of more than 2%, then adjust the affected signal(s) to agree with the calorimetric calculation.

During PHYSICS TESTS, these daily calibrations may be suspended provided these calibrations are performed upon reaching each major test power plateau and prior to proceeding to the next major test power plateau.

(3) - Above 15% of RATED THERMAL POWER, verify that the linear power subchannel gains of the excore detectors are consistent with the values used to establish the shape annealing matrix elements in the Core Protection Calculators.

(4) - Neutron detectors may be excluded from CHANNEL CALIBRATION.

(5) - After each fuel loading and prior to exceeding 70% of RATED THERMAL POWER, the incore detectors shall be used to determine the shape annealing matrix elements and the Core Protection Calculators shall use these elements.

TABLE 4.3-1 (Continued)

TABLE NOTATIONS

(6) - This CHANNEL FUNCTIONAL TEST shall include the injection of simulated process signals into the channel as close to the sensors as practicable to verify OPERABILITY including alarm and/or trip functions.

(7) - Above 70% of RATED THERMAL POWER, verify that the total steady-state RCS flow rate as indicated by each CPC is less than or equal to the actual RCS total flow rate determined by either using the reactor coolant pump differential pressure instrumentation or by calorimetric calculations and if necessary, adjust the CPC addressable constant flow coefficients such that each CPC indicated flow is less than or equal to the actual flow rate. The flow measurement uncertainty may be included in the BERR1 team in the CPC and is equal to or greater than 4%.

(8) - Above 70% of RATED THERMAL POWER, verify that the total steady-state RCS flow rate as indicated by each CPC is less than or equal to the actual RCS total flow rate determined by either using the reactor coolant pump differential pressure instrumentation and the ultrasonic flow meter adjusted pump curves or calorimetric calculations.

(9) - The quarterly CHANNEL FUNCTIONAL TEST shall include verification that the correct current values of addressable constants are installed in each OPERABLE CPC.

(10) - At least once per 18 months and following maintenance or adjustment of the reactor trip breakers, the CHANNEL FUNCTIONAL TEST shall include independent verification of the undervoltage and shunt trips.

3/4.9 REFUELING OPERATIONS

3/4.9.1 BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.9.1 With the reactor vessel head closure bolts less than fully tensioned or with the head removed, the boron concentration of all filled portions of the Reactor Coolant System and the refueling canal shall be maintained uniform and within the limit specified in the Core Operating Limits Report (COLR).

APPLICABILITY: MODE 6*.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes and initiate and continue boration at greater than or equal to 26 gpm of a solution containing ≥ 4000 ppm boron or its equivalent until the boron concentration is within limits.

SURVEILLANCE REQUIREMENTS

4.9.1.1 The boron concentration shall be determined to be within the limit specified in the COLR prior to:

a. Removing or unbolting the reactor vessel head, and

b. Withdrawal of any full-length CEA in excess of 3 feet from its fully inserted position within the reactor pressure vessel.

4.9.1.2 The boron concentration of the Reactor Coolant System and the refueling canal shall be determined by chemical analysis at least once per 72 hours.

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*The reactor shall be maintained in MODE 6 whenever fuel is in the reactor vessel with the reactor vessel head closure bolts less than fully tensioned or with the head removed.

3/4.9.2 INTRUMENTATION

LIMITED CONDITION FOR OPERATION

3.9.2 As a minimum, two startup channel neutron flux monitors shall be OPERABLE and operating, each with continuous visual indication in the control room and one with audible indication in the containment and control room.

APPLICABILITY: MODE 6.

ACTION:

a. With one of the above required monitors inoperable or not operating, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

b. With both of the above required monitors inoperable or not operating, determine the boron concentration of the Reactor Coolant System at least once per 12 hours.

SURVEILLANCE REQUIREMENTS

4.9.2 Each startup channel neutron flux monitor shall be demonstrated OPERABLE by performance of:

a. A CHANNEL CHECK at least once per 12 hours,

b. A CHANNEL FUNCTIONAL TEST within 8 hours prior to the initial start of CORE ALTERATIONS, and

c. A CHANNEL FUNTIONAL TEST at least once per 7 days.

3/4.1 REACTIVITY CONTROL SYSTEMS

BASES

3/4.1.1 BORATION CONTROL

3/4.1.1.1 and 3/4.1.1.2 SHUTDOWN MARGIN AND K_{N-1}

The function of SHUTDOWN MARGIN is to ensure that the reactor remains subcritical following a design basis accident or anticipated operational occurrence. The function of K_{N-1} is to maintain sufficient subcriticality to preclude inadvertent criticality following ejection of a single control element assembly (CEA). During operation in MODES 1 and 2, with k_{eff} greater than or equal to 1.0, the transient insertion limits of Specification 3.1.3.6 ensure that sufficient SHUTDOWN MARGIN is available.

SHUTDOWN MARGIN is the amount by which the core is subcritical, or would be subcritical immediately following a reactor trip, considering a single malfunction resulting in the highest worth CEA failing to insert. K_{N-1} is a measure of the core's reactivity, considering a single malfunction resulting in the highest worth inserted CEA being ejected.

SHUTDOWN MARGIN requirements vary throughout the core life as a function of fuel depletion and reactor coolant system (RCS) cold leg temperature (T_{cold}). The most restrictive condition occurs at EOL, with T_{cold} at no-load operating temperature, and is associated with a postulated steam line break accident and the resulting uncontrolled RCS cooldown. In the analysis of this accident, the specified SHUTDOWN MARGIN is required to control the reactivity transient and ensure that the fuel performance and offsite dose criteria are satisfied. As (initial) T_{cold} decreases, the potential RCS cooldown and the resulting reactivity transient are less severe and, therefore, the required SHUTDOWN MARGIN also decreases. Below T_{cold} of about 350°F, the inadvertent deboration event becomes limiting with respect to the SHUTDOWN MARGIN requirements. Below 350°F, the specified SHUTDOWN MARGIN ensures that sufficient time for operator actions exists between the initial indication of the deboration and the total loss of shutdown margin. Accordingly, with the reactor trip breakers closed and the CEA drive system capable of CEA withdrawal, the SHUTDOWN MARGIN requirements are based upon these limiting conditions.

Additional events considered in establishing requirements on SHUTDOWN MARGIN that are not limiting with respect to the Specification limits are single CEA withdrawal and startup of an inactive reactor coolant pump.

K_{N-1} requirements vary with the amount of positive reactivity that would be introduced assuming the CEA with the highest inserted worth ejects from the core. In the analysis of the CEA ejection event, the K_{N-1} requirement ensures that the radially averaged enthalpy acceptance criterion is satisfied, considering power redistribution effects. Above T_{cold} of 500°F, Doppler reactivity feedback is sufficient to preclude the need for a specific K_{N-1} requirement. With all CEAs fully inserted, K_{N-1} and SHUTDOWN MARGIN requirements are equivalent in terms of minimum acceptable core boron concentration.

SHUTDOWN MARGIN AND K_{N-1} (continued)

The requirement prohibiting criticality due to shutdown group CEA movement is associated with the assumptions used in the analysis of uncontrolled CEA withdrawal from subcritical conditions. Due to the high differential reactivity worth of the shutdown CEA groups, the analysis assumes that the initial shutdown reactivity is such that the reactor will remain subcritical in the event of unexpected or uncontrolled shutdown group withdrawal.

Other technical specifications that reference the Specifications on SHUTDOWN MARGIN or K_{N-1} are: 3/4.1.2, BORATION SYSTEMS, 3/4.1.3, MOVABLE CONTROL ASSEMBLIES, 3/4.9.1, REFUELING OPERATIONS-BORON CONCENTRATION, 3/4.10.1, SHUTDOWN MARGIN AND K_{N-1} - CEA WORTH TESTS, and 3/4.10.9, SHUTDOWN MARGIN AND K_{N-1} - CEDMS TESTING.

3/4.1.1.3 MODERATOR TEMPERATURE COEFFICIENT (MTC)

The limitations on moderator temperature coefficient (MTC) are provided to ensure that the assumptions used in the accident and transient analysis remain valid through each fuel cycle. The surveillance requirements for measurement of the MTC during each fuel cycle are adequate to confirm the MTC value since this coefficient changes slowly due principally to the reduction in RCS boron concentration associated with fuel burnup. The confirmation that the measured MTC value is within its limit provides assurances that the coefficient will be maintained within acceptable values throughout each fuel cycle.

3/4.1.1.4 MINIMUM TEMPERATURE FOR CRITICALITY

This specification ensures that the reactor will not be made critical with the Reactor Coolant System cold leg temperature less than 545°F. This limitation is required to ensure (1) the moderator temperature coefficient is within its analyzed temperature range, (2) the protective instrumentation is within its normal operating range, (3) a minimum temperature is provided for Special Test Exception 3/4.10.4, and (4) the reactor vessel is above its minimum RT_{NDT} temperature.

3/4.9 REFUELING OPERATIONS

BASES

3/4.9.1 BORON CONCENTRATION

The limitations on reactivity conditions during REFUELING ensure that: (1) the reactor will remain subcritical during CORE ALTERATIONS, and (2) a uniform boron concentration is maintained for reactivity control in the water volume having direct access to the reactor vessel. These limitations are consistent with the initial conditions assumed for the boron dilution incident in the safety analyses. The boron concentration limit specified in the COLR is based on core reactivity at the beginning of each cycle (the end of refueling) with all CEAs withdrawn and includes an uncertainty allowance. This boron concentration limit will ensure a K_{eff} of ≤ 0.95 during the refueling operation.

3/4.9.2 INSTRUMENTATION

The OPERABILITY of the startup channel neutron flux monitors ensures that redundant monitoring capability is available to detect changes in the reactivity condition of the core.

3/4.9.3 DECAY TIME

The minimum requirement for reactor subcriticality prior to movement of irradiated fuel assemblies in the reactor pressure vessel ensures that sufficient time has elapsed to allow the radioactive decay of the short lived fission products. This decay time is consistent with the assumptions used in the safety analyses.

3/4.9.4 CONTAINMENT BUILDING PENETRATIONS

The requirements on containment penetration closure and OPERABILITY ensure that a release of radioactive material within containment will be restricted from leakage to the environment. The OPERABILITY and closure restrictions are sufficient to restrict radioactive material release from a fuel element rupture based upon the lack of containment pressurization potential while in the REFUELING MODE.

3/4.9.5 COMMUNICATIONS

The requirement for communications capability ensures that refueling station personnel can be promptly informed of significant changes in the facility status or core reactivity condition during CORE ALTERATIONS.

3/4.9.6 REFUELING MACHINE

The OPERABILITY requirements for the refueling machine ensure that: (1) the machine will be used for movement of fuel assemblies, (2) the machine has sufficient load capacity to lift a fuel assembly, and (3) the core internals and pressure vessel are protected from excessive lifting force in the event they are inadvertently engaged during lifting operations.

3/4.9.7 CRANE TRAVEL - SPENT FUEL STORAGE POOL BUILDING

The restriction on movement of loads in excess of the nominal weight of a fuel assembly, CEA and associated handling tool over other fuel assemblies in the storage pool ensures that in the event this load is dropped (1) the activity release will be limited to that contained in a single fuel assembly, and (2) any possible distortion of fuel in the storage racks will not result in a critical array. This assumption is consistent with the activity release assumed in the safety analyses.

3/4.9.8 SHUTDOWN COOLING AND COOLANT CIRCULATION

The requirement that at least one shutdown cooling loop be in operation, and circulating reactor coolant at a flow rate equal to or greater than 3400 gpm (actual) ensures that (1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor pressure vessel below 135°F as required during the REFUELING MODE, (2) sufficient coolant circulation is maintained through the reactor core to minimize the effects of a boron dilution incident and prevent boron stratification, and (3) the ΔT across the core will be maintained at less than 75°F during the REFUELING MODE. The required flowrate of $\geq$ 3400 gpm (actual) ensures that at 288 hours after reactor shutdown sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor pressure vessel below 135°F as required during REFUELING MODE; this assumes a shutdown cooling heat exchanger cooling water flowrate of 14000 gpm, a cooling water inlet temperature of $\leq$ 105°F at $\geq$ 27 1/2 hours after reactor shutdown, and the decay heat curve of CESSAR-F Figure 6.2.1-1 and reactor operation for two years at 4000 MWt. The 3780 gpm in the specification includes all instrument uncertainties including the 300°F calibration temperature of the flow transmitters.

Without a shutdown cooling train in operation steam may be generated; therefore, the containment should be sealed off to prevent escape of any radioactivity, and any operations that would cause an increase in decay heat should be secured.

The requirement to have two shutdown cooling loops OPERABLE when there is less than 23 feet of water above the reactor pressure vessel flange, ensures that a single failure of the operating shutdown cooling loop will not result in a complete loss of decay heat removal capability. With the reactor vessel head removed and 23 feet of water above the reactor pressure vessel flange, a large heat sink is available for core cooling, thus in the event of a failure of the operating shutdown cooling loop, adequate time is provided to initiate emergency procedures to cool the core.

CORE OPERATING LIMITS REPORT

6.9.1.9 Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT before each reload cycle or any remaining part of a reload cycle for the following:

a. Shutdown Margin - Reactor Trip Breakers Closed for Specification 3.1.1.2
b. Moderator Temperature Coefficient BOL and EOL limits for Specification 3.1.1.3
c. Boron Dilution Alarms for Specification 3.1.2.7
d. Movable Control Assemblies - CEA Position for Specification 3.1.3.1
e. Regulating CEA Insertion Limits for Specification 3.1.3.6
f. Part Length CEA Insertion Limits for Specification 3.1.3.7
g. Linear Heat Rate for Specification 3.2.1
h. Azimuthal Power Tilt - T_q for Specification 3.2.3
i. DNBR Margin for Specification 3.2.4
j. Axial Shape Index for Specification 3.2.7
k. Boron Concentration (Mode 6) for Specification 3.9.1

6.9.1.10 The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC in:

a. "CE Method for Control Element Assembly Ejection Analysis, "CENPD-0190-A, January 1976 (Methodology for Specification 3.1.3.6, Regulating CEA Insertion Limits).

b. "The ROCS and DIT Computer Codes for Nuclear Design," CENPD-266-P-A, April 1983 [Methodology for Specifications 3.1.1.2, Shutdown Margin - Reactor Trip Breakers Closed; 3.1.1.3, Moderator Temperature Coefficient BOL and EOL limits; 3.1.3.6, Regulating CEA Insertion Limits and 3.9.1, Boron Concentration (Mode 6)].

c. "Safety Evaluation Report related to the Final Design of the Standard Nuclear Steam Supply Reference Systems CESSAR System 80, Docket No. STN 50-470, "NUREG-0852 (Novenber 1981), Supplements No. 1 (March 1983), No. 2 (September 1983), No. 3 (December 1987) (Methodology for Specifications 3.1.1.2, Reactor Trip Breakers Closed; 3.1.1.3, Moderator Temperature Coefficient BOL and EOL limits; 3.1.2.7, Boron Dilution Alarms; 3.1.3.1, Movable Control Assemblies - CEA Position; 3.1.3.6, Regulating CEA Insertion Limits; 3.1.3.7, Part Length CEA Insertion Limits and 3.2.3 Azimuthal Power Tilt - T_q).

d. "Modified Statistical Combination of Uncertainties," CEN-356(V)-P-A Revision 01-P-A, May 1988 and "System 80™ Inlet Flow Distribution," Supplement 1-P to Enclosure 1-P to LD-82-054, February 1993 (Methodology for Specification 3.2.4, DNBR Margin and 3.2.7 Axial Shape Index).

CORE OPERATING LIMITS REPORT (Continued)

e. "Calculative Methods for the CE Large Break LOCA Evaluation Model for the Analysis of CE and W Designed NSSS," CENPD-132, Supplement 3-P-A, June 1985 (Methodology for Specification 3.2.1, Linear Heat Rate).

f. "Calculative Methods for the CE Small Break LOCA Evaluation Model," CENPD-137-P, August 1974 (Methodology for Specification 3.2.1, Linear Heat Rate).

g. "Calculative Methods for the CE Small Break LOCA Evaluation Model," CENPD-137-P, Supplement 1P, January 1977 (Methodology for Specification 3.2.1, Linear Heat Rate).

h. Letter: O. D. Parr (NRC) to F. M. Stern (CE), dated June 13, 1975 (NRC Staff Review of the Combustion Engineering ECCS Evaluation Model). NRC approval for: 6.9.1.10f.

i. Letter: K. Kniel (NRC) to A. E. Scherer (CE), dated September 27, 1977 (Evaluation of Topical Reports CENPD-133, Supplement 3-P and CENPD-137, Supplement 1-P). NRC approval for 6.9.1.10.g.

The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and analysis limits) of the safety analysis are met.

The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

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. 86
License No. NPF-51

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

   A. The application for amendment by the Arizona Public Service Company (APS or the licensee) 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 dated March 31, 1995, 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 the Act, and the rules and 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; and

   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 attachment 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. 86, 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, except where otherwise stated in specific license conditions.

3. The license amendment is effective as of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Charles R. Thomas, Project Manager
Project Directorate IV-2
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: September 1, 1995

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 86 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. The corresponding overleaf pages are also provided to maintain document completeness.

REMOVE	INSERT
III*	III*
IV	IV
1-6	1-6
3/4 1-1	4/4 1-1
3/4 1-2	3/4 1-2
3/4 1-3	3/4 1-3
3/4 3-3	3/4 3-3
3/4 3-7*	3/4 3-7*
3/4 3-8	3/4 3-8
3/4 3-16	3/4 3-16
--	3/4 3-16a
3/4 9-1	3/4 9-1
B 3/4 1-1	B 3/4 1-1
B 3/4 1-1a	B 3/4 1-1a
B 3/4 9-1	B 3/4 9-1
B 3/4 9-2*	B 3/4 9-2*
6-20a	6-20a

*No changes were made to these pages; reissued to become overleaf pages.

INDEX

SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

BASES

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

REPORTABLE EVENT

1.28 A REPORTABLE EVENT shall be any of those conditions specified in Sections 50.72 and 50.73 to 10 CFR Part 50.

SHUTDOWN MARGIN

1.29 SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming:

a. No change in part-length control element assembly position, and

b. All full-length control element assemblies (shutdown and regulating) are fully inserted except for the single assembly 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 full-length CEAs must be accounted for in the determination of the SHUTDOWN MARGIN.

SITE BOUNDARY

1.30 The SITE BOUNDARY shall be that line beyond which the land is neither owned, nor leased, nor otherwise controlled by the licensee.

SOFTWARE

1.31 The digital computer SOFTWARE for the reactor protection system shall be the program codes including their associated data, documentation, and procedures.

SOURCE CHECK

1.32 A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to a source of increased radioactivity.

STAGGERED TEST BASIS

1.33 A STAGGERED TEST BASIS shall consist of:

a. A test schedule for n systems, subsystems, trains, or other designated components obtained by dividing the specified test interval into n equal subintervals, and

b. The testing of one system, subsystem, train, or other designated component at the beginning of each subinterval.

THERMAL POWER

1.34 THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - REACTOR TRIP BREAKERS OPEN**

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be greater than or equal to 1.0% delta k/k.

APPLICABILITY: MODES 3, 4* and 5* with the reactor trip breakers open.**

ACTION:

With the SHUTDOWN MARGIN less than 1.0% delta k/k, immediately initiate and continue boration at greater than or equal to 26 gpm to reactor coolant system of a solution containing greater than or equal to 4000 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 1.0% delta k/k at least once per 24 hours by consideration of at least the following factors:

1. Reactor Coolant System boron concentration,
2. CEA position,
3. Reactor Coolant System average temperature,
4. Fuel burnup based on gross thermal energy generation,
5. Xenon concentration, and
6. Samarium concentration.

4.1.1.1.2 The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within ± 1.0% delta k/k at least once per 31 Effective Full Power Days (EFPD). This comparison shall consider at least those factors stated in Specification 4.1.1.1.1, above. The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 EFPD after each fuel loading.

4.1.1.1.3 With the reactor trip breakers open** and any CEA(s) fully or partially withdrawn, the SHUTDOWN MARGIN shall be verified within one hour after detection of the withdrawn CEA(s) and at least once per 12 hours thereafter while the CEA(s) are withdrawn.

---

* See Special Test Exception 3.10.9.

**The CEA drive system not capable of CEA withdrawal.

SHUTDOWN MARGIN - REACTOR TRIP BREAKERS CLOSED**

LIMITING CONDITION FOR OPERATION

3.1.1.2

a. The SHUTDOWN MARGIN shall be greater than or equal to that specified in the CORE OPERATING LIMITS REPORT, and

b. For T_{cold} less than or equal to 500°F, K_{N-1} shall be less than 0.99.

c. Reactor criticality shall not be achieved with shutdown group CEA movement.

APPLICABILITY: MODES 1, 2*, 3*, 4*, and 5* with the reactor trip breakers closed**.

ACTION:

a. With the SHUTDOWN MARGIN less than that specified in the CORE OPERATING LIMITS REPORT, immediately initiate and continue boration at greater than or equal to 26 gpm to the reactor coolant system of a solution containing greater than or equal to 4000 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored, and

b. With T_{cold} less than or equal to 500°F and K_{N-1} greater than or equal to 0.99, immediately vary CEA positions and/or initiate and continue boration at greater than or equal to 26 gpm to the reactor coolant system of a solution containing greater than or equal to 4000 ppm boron or equivalent until the required K_{N-1} is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.2.1 With the reactor trip breakers closed**, the SHUTDOWN MARGIN shall be determined to be greater than or equal to that specified in the CORE OPERATING LIMITS REPORT:

a. Within 1 hour after detection of an inoperable CEA(s) and at least once per 12 hours thereafter while the CEA(s) is inoperable.

* See Special Test Exceptions 3.10.1 and 3.10.9.
**The CEA drive system capable of CEA withdrawal.

b. When in MODE 1 or MODE 2 with k_{eff} greater than or equal to 1.0, at least once per 12 hours by verifying that CEA group withdrawal is within the Transient Insertion Limits of Specification 3.1.3.6. If CEA group withdrawal is not within the Transient Insertion Limits of Specification 3.1.3.6, within 1 hour verify that SHUTDOWN MARGIN is greater than or equal to that specified in the CORE OPERATING LIMITS REPORT.

c. When in MODE 2 with k_{eff} less than 1.0, within 4 hours prior to achieving reactor criticality by verifying that predicted critical CEA position is within the limits of Specification 3.1.3.6.

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of e. below, with the CEA groups at the Transient Insertion Limits of Specification 3.1.3.6.

e. When in MODE 3, 4, or 5, at least once per 24 hours by consideration of at least the following factors:

1. Reactor Coolant System boron concentration,
2. CEA position,
3. Reactor Coolant System average temperature,
4. Fuel burnup based on gross thermal energy generation,
5. Xenon concentration, and
6. Samarium concentration.

4.1.1.2.2 When in MODE 3, 4, or 5, with the reactor trip breakers closed**, and T_{cold} less than or equal to 500°F, K_{N-1} shall be determined to be less than 0.99 at least once per 24 hours by consideration of at least the following factors.

1. Reactor Coolant System boron concentration,
2. CEA position,
3. Reactor Coolant System average temperature,
4. Fuel burnup based on gross thermal energy generation,
5. Xenon concentration, and
6. Samarium concentration.

4.1.1.2.3 When in MODES 3, 4, or 5 with the reactor trip breakers closed**, verify that criticality cannot be achieved with shutdown group CEA withdrawal at least once per 24 hours by consideration of at least the following factors:

1. Reactor Coolant System boron concentration,
2. CEA position,
3. Reactor Coolant System average temperature,
4. Fuel burnup based on gross thermal energy generation,
5. Xenon concentration, and
6. Samarium concentration.

4.1.1.2.4 The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within ± 1.0% delta k/k at least once per 31 Effective Full Power Days (EFPD). This comparison shall consider at least those factors stated in Specification 4.1.1.2.1.e or 4.1.1.2.2. The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 EFPD after each fuel loading.

**The CEA drive system capable of CEA withdrawal.

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.3 The moderator temperature coefficient (MTC) shall be within the area of Acceptable Operation specified in the CORE OPERATING LIMITS REPORT. The maximum upper limit shall be less than or equal to + 0.5 x 10^{-4} ΔK/K/°F for a power level of 0% RATED THERMAL POWER with a linear ramp to 0ΔK/K/°F at 100% RATED THERMAL POWER.

APPLICABILITY: MODES 1 and 2*#.

ACTION:

With the moderator temperature coefficient outside the area of Acceptable Operation, be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.1.3.1 The MTC shall be determined to be within its limits by confirmatory measurements. MTC measured values shall be extrapolated and/or compensated to permit direct comparison with the above limits.

4.1.1.3.2 The MTC shall be determined at the following frequencies and THERMAL POWER conditions during each fuel cycle:

a. Prior to initial operation above 5% of RATED THERMAL POWER, after each fuel loading.

b. At any THERMAL POWER, within 7 EFPD after reaching a core average exposure of 40 EFPD burnup into the current cycle.

c. At any THERMAL POWER, within 7 EFPD after reaching a core average exposure equivalent to two-thirds of the expected current cycle end-of-cycle core average burnup.

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*With K_{eff} greater than or equal to 1.0.

#See Special Test Exception 3.10.2.

TABLE 3.3-1

REACTOR PROTECTIVE INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
I. TRIP GENERATION					
A. Process					
1. Pressurizer Pressure - High	4	2	3	1, 2	2#, 3#
2. Pressurizer Pressure - Low	4	2 (b)	3	1, 2	2#, 3#
3. Steam Generator Level - Low	4/SG	2/SG	3/SG	1, 2	2#, 3#
4. Steam Generator Level - High	4/SG	2/SG	3/SG	1, 2	2#, 3#
5. Steam Generator Pressure - Low	4/SG	2/SG	3/SG	1, 2, 3*, 4*	2#, 3#
6. Containment Pressure - High	4	2	3	1, 2	2#, 3#
7. Reactor Coolant Flow - Low	4/SG	2/SG	3/SG	1, 2	2#, 3#
8. Local Power Density - High	4	2 (c)(d)	3	1, 2	2#, 3#
9. DNBR - Low	4	2 (c)(d)	3	1, 2	2#, 3#
B. Excore Neutron Flux					
1. Variable Overpower Trip	4	2	3	1, 2	2#, 3#
2. Logarithmic Power Level - High					
a. Startup and Operating	4	2 (a)(d)	3	1, 2	2#, 3#
	4	2	3	3*, 4*, 5*	9
b. Shutdown	4	0	2	3, 4, 5	4
C. Core Protection Calculator System					
1. CEA Calculators	2	1	2 (e)	1, 2	6, 7
2. Core Protection Calculators	4	2 (c)(d)	3	1,2,3*,4*,5*	2#, 3#, 7, 10

TABLE 3.3-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION

ACTION STATEMENTS

3.	Steam Generator Pressure - Low	Steam Generator Pressure - Low Steam Generator Level 1-Low (ESF) Steam Generator Level 2-Low (ESF)
4.	Steam Generator Level - Low (Wide Range)	Steam Generator Level - Low (RPS) Steam Generator Level 1-Low (ESF) Steam Generator Level 2-Low (ESF)
5.	Core Protection Calculator	Local Power Density - High (RPS) DNBR - Low (RPS)

STARTUP and/or POWER OPERATION may continue until the performance of the next required CHANNEL FUNCTIONAL TEST. Subsequent STARTUP and/or POWER OPERATION may continue if one channel is restored to OPERABLE status and the provisions of ACTION 2 are satisfied.

ACTION 4 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, suspend all operations involving positive reactivity changes.

ACTION 5 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, STARTUP and/or POWER OPERATION may continue provided the reactor trip breaker of the inoperable channel is placed in the tripped condition within 1 hour, otherwise, be in at least HOT STANDBY within 6 hours; however, the trip breaker associated with the inoperable channel may be closed for up to 1 hour for surveillance testing per Specification 4.3.1.1.

ACTION 6 - a. With one CEAC inoperable, operation may continue for up to 7 days provided that the requirements of Specification 4.1.3.1.1 are met. After 7 days, operation may continue provided that the conditions of Action Item 6.b are met.

b. With both CEACs inoperable, operation may continue provided that:

1. Within 1 hour the DNBR margin required by Specification 3.2.4.b (COLSS in service) or 3.2.4.d (COLSS out of service) is satisfied and the Reactor Power Cutback System is disabled, and

TABLE 3.3-1 (Continued)
REACTOR PROTECTIVE INSTRUMENTATION
ACTION STATEMENTS

2. Within 4 hours:

   a) All full-length and part-length CEA groups must be withdrawn within the limits of Specifications 3.1.3.5, 3.1.3.6b, and 3.1.3.7b, except during surveillance testing pursuant to the requirements of Specification 4.1.3.1.2. Specification 3.1.3.6b allows CEA group 5 insertion to no further than 127.5 inches withdrawn.

   b) The "RSPT/CEAC Inoperable" addressable constant in the CPCs is set to indicate that both CEACs are inoperable.

   c) The Control Element Drive Mechanism Control System (CEDMCS) is placed in and subsequently maintained in the "Standby" mode except during CEA motion permitted by Specifications 3.1.3.5, 3.1.3.6b and 3.1.3.7b when the CEDMCS may be operated in either the "Manual Group" or "Manual Individual" mode.

3. CEA position surveillance must meet the requirements of Specifications 4.1.3.1.1, 4.1.3.5, 4.1.3.6, and 4.1.3.7 except during surveillance testing pursuant to Specification 4.1.3.1.2.

ACTION 7 - With three or more auto restarts, excluding periodic auto restarts (Code 30 and Code 33), of one non-bypassed calculator during a 12-hour interval, demonstrate calculator OPERABILITY by performing a CHANNEL FUNCTIONAL TEST within the next 24 hours.

ACTION 8 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore an inoperable channel to OPERABLE status within 48 hours or open an affected reactor trip breaker within the next hour.

ACTION 9 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or open the reactor trip breakers within the next hour.

ACTION 10 - In MODES 3, 4, or 5, the Core Protection Calculator channels are not required to be OPERABLE when the Logarithmic Power Level - High trip is OPERABLE with the trip setpoint lowered to $\leq 10^{-4}$% of RATED THERMAL Power.

TABLE 4.3-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
D. Supplementary Protection System				
Pressurizer Pressure - High	S	R	Q	1, 2
II. RPS LOGIC				
A. Matrix Logic	N.A.	N.A.	Q	1, 2, 3*, 4*, 5*
B. Initiation Logic	N.A.	N.A.	Q	1, 2, 3*, 4*, 5*
III. RPS ACTUATION DEVICES				
A. Reactor Trip Breakers	N.A.	N.A.	M, R(10)	1, 2, 3*, 4*, 5*
B. Manual Trip	N.A.	N.A.	Q	1, 2, 3*, 4*, 5*

TABLE 4.3-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TABLE NOTATIONS

* - With reactor trip breakers in the closed position and the CEA drive system capable of CEA withdrawal, and fuel in the reactor vessel.

(1) - Each STARTUP or when required with the reactor trip breakers closed and the CEA drive system capable of rod withdrawal, if not performed in the previous 7 days.

(2) - Heat balance only (CHANNEL FUNCTIONAL TEST not included):

a. Between 15% and 80% of RATED THERMAL POWER, compare the linear power level, the CPC delta T power and the CPC nuclear power signals to the calorimetric calculation.

If any signal is within -0.5% to 10% of the calorimetric then do not calibrate except as required during initial power ascension after refueling.

If any signal is less than the calorimetric calculation by more than 0.5%, then adjust the affected signal(s) to agree with the calorimetric calculation.

If any signal is greater than the calorimetric calculation by more than 10% then adjust the affected signal(s) to agree with the calorimetric calculation within 8% to 10%.

b. At or above 80% of RATED THERMAL POWER; compare the linear power level, the CPC delta T power and the CPC nuclear power signals to the calorimetric calculation. If any signal differs from the calorimetric calculation by an absolute difference of more than 2%, then adjust the affected signal(s) to agree with the calorimetric calculation.

During PHYSICS TESTS, these daily calibrations may be suspended provided these calibrations are performed upon reaching each major test power plateau and prior to proceeding to the next major test power plateau.

(3) - Above 15% of RATED THERMAL POWER, verify that the linear power sub-channel gains of the excore detectors are consistent with the values used to establish the shape annealing matrix elements in the Core Protection Calculators.

(4) - Neutron detectors may be excluded from CHANNEL CALIBRATION.

(5) - After each fuel loading and prior to exceeding 70% of RATED THERMAL POWER, the incore detectors shall be used to determine the shape annealing matrix elements and the Core Protection Calculators shall use these elements.

TABLE 4.3-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TABLE NOTATIONS

(6) - This CHANNEL FUNCTIONAL TEST shall include the injection of simulated process signals into the channel as close to the sensors as practicable to verify OPERABILITY including alarm and/or trip functions.

(7) - Above 70% of RATED THERMAL POWER, verify that the total steady-state RCS flow rate as indicated by each CPC is less than or equal to the actual RCS total flow rate determined by either using the reactor coolant pump differential pressure instrumentation or by calorimetric calculations and if necessary, adjust the CPC addressable constant flow coefficients such that each CPC indicated flow is less than or equal to the actual flow rate. The flow measurement uncertainty may be included in the BERR1 term in the CPC and is equal to or greater than 4%.

(8) - Above 70% of RATED THERMAL POWER, verify that the total steady-state RCS flow rate as indicated by each CPC is less than or equal to the actual RCS total flow rate determined by either using the reactor coolant pump differential pressure instrumentation and the ultrasonic flow meter adjusted pump curves or calorimetric calculations.

(9) - The quarterly CHANNEL FUNCTIONAL TEST shall include verification that the correct (current) values of addressable constants are installed in each OPERABLE CPC.

(10) - At least once per 18 months and following maintenance or adjustment of the reactor trip breakers, the CHANNEL FUNCTIONAL TEST shall include independent verification of the undervoltage and shunt trips.

3/4.9 REFUELING OPERATIONS

3/4.9.1 BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.9.1 With the reactor vessel head closure bolts less than fully tensioned or with the head removed, the boron concentration of all filled portions of the Reactor Coolant System and the refueling canal shall be maintained uniform and within the limit specified in the Core Operating Limits Report (COLR).

APPLICABILITY: MODE 6*.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes and initiate and continue boration at greater than or equal to 26 gpm of a solution containing $\geq$ 4000 ppm boron or its equivalent until the boron concentration is within limits.

SURVEILLANCE REQUIREMENTS

4.9.1.1 The boron concentration shall be determined to be within the limit specified in the COLR prior to:

a. Removing or unbolting the reactor vessel head, and

b. Withdrawal of any full-length CEA in excess of 3 feet from its fully inserted position within the reactor pressure vessel.

4.9.1.2 The boron concentration of the Reactor Coolant System and the refueling canal shall be determined by chemical analysis at least once per 72 hours.

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*The reactor shall be maintained in MODE 6 whenever fuel is in the reactor vessel with the reactor vessel head closure bolts less than fully tensioned or with the head removed.

3/4.9.2 INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.9.2 As a minimum, two startup channel neutron flux monitors shall be OPERABLE and operating, each with continuous visual indication in the control room and one with audible indication in the containment and control room.

APPLICABILITY: MODE 6.

ACTION:

a. With one of the above required monitors inoperable or not operating, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

b. With both of the above required monitors inoperable or not operating, determine the boron concentration of the Reactor Coolant System at least once per 12 hours.

SURVEILLANCE REQUIREMENTS

4.9.2 Each startup channel neutron flux monitor shall be demonstrated OPERABLE by performance of:

a. A CHANNEL CHECK at least once per 12 hours,

b. A CHANNEL FUNCTIONAL TEST within 8 hours prior to the initial start of CORE ALTERATIONS, and

c. A CHANNEL FUNCTIONAL TEST at least once per 7 days.

3/4.1 REACTIVITY CONTROL SYSTEMS

BASES

3/4.1.1 BORATION CONTROL

3/4.1.1.1 and 3/4.1.1.2 SHUTDOWN MARGIN and K_{N-1}

The function of SHUTDOWN MARGIN is to ensure that the reactor remains subcritical following a design basis accident or anticipated operational occurrence. The function of K_{N-1} is to maintain sufficient subcriticality to preclude inadvertent criticality following ejection of a single control element assembly (CEA). During operation in MODES 1 and 2, with k_{eff} greater than or equal to 1.0, the transient insertion limits of Specification 3.1.3.6 ensure that sufficient SHUTDOWN MARGIN is available.

SHUTDOWN MARGIN is the amount by which the core is subcritical, or would be subcritical immediately following a reactor trip, considering a single malfunction resulting in the highest worth CEA failing to insert. K_{N-1} is a measure of the core's reactivity, considering a single malfunction resulting in the highest worth inserted CEA being ejected.

SHUTDOWN MARGIN requirements vary throughout the core life as a function of fuel depletion and reactor coolant system (RCS) cold leg temperature (T_{cold}). The most restrictive condition occurs at EOL, with T_{cold} at no-load operating temperature, and is associated with a postulated steam line break accident and the resulting uncontrolled RCS cooldown. In the analysis of this accident, the specified SHUTDOWN MARGIN is required to control the reactivity transient and ensure that the fuel performance and offsite dose criteria are satisfied. As (initial) T_{cold} decreases, the potential RCS cooldown and the resulting reactivity transient are less severe and, therefore, the required SHUTDOWN MARGIN also decreases. Below T_{cold} of about 350°F, the inadvertent deboration event becomes limiting with respect to the SHUTDOWN MARGIN requirements. Below 350°F, the specified SHUTDOWN MARGIN ensures that sufficient time for operator actions exists between the initial indication of the deboration and the total loss of shutdown margin. Accordingly, with the reactor trip breakers closed and the CEA drive system capable of CEA withdrawal, the SHUTDOWN MARGIN requirements are based upon these limiting conditions.

Additional events considered in establishing requirements on SHUTDOWN MARGIN that are not limiting with respect to the Specification limits are single CEA withdrawal and startup of an inactive reactor coolant pump.

K_{N-1} requirements vary with the amount of positive reactivity that would be introduced assuming the CEA with the highest inserted worth ejects from the core. In the analysis of the CEA ejection event, the K_{N-1} requirement ensures that the radially averaged enthalpy acceptance criterion is satisfied, considering power redistribution effects. Above T_{cold} of 500°F, Doppler reactivity feedback is sufficient to preclude the need for a specific K_{N-1} requirement. With all CEAs fully inserted, K_{N-1} and SHUTDOWN MARGIN requirements are equivalent in terms of minimum acceptable core boron concentration.

SHUTDOWN MARGIN and K_{N-1} (continued)

The requirement prohibiting criticality due to shutdown group CEA movement is associated with the assumptions used in the analysis of uncontrolled CEA withdrawal from subcritical conditions. Due to the high differential reactivity worth of the shutdown CEA groups, the analysis assumes that the initial shutdown reactivity is such that the reactor will remain subcritical in the event of unexpected or uncontrolled shutdown group withdrawal.

Other technical specifications that reference the Specifications on SHUTDOWN MARGIN or K_{N-1} are: 3/4.1.2, BORATION SYSTEMS, 3/4.1.3, MOVABLE CONTROL ASSEMBLIES, 3/4.9.1, REFUELING OPERATIONS-BORON CONCENTRATION, 3/4.10.1, SHUTDOWN MARGIN AND K_{N-1} - CEA WORTH TESTS, and 3/4.10.9, SHUTDOWN MARGIN AND K_{N-1} - CEDMS TESTING.

3/4.1.1.3 MODERATOR TEMPERATURE COEFFICIENT (MTC)

The limitations on moderator temperature coefficient (MTC) are provided to ensure that the assumptions used in the accident and the transient analysis remain valid through each fuel cycle. The surveillance requirements for measurement of the MTC during each fuel cycle are adequate to confirm the MTC value since this coefficient changes slowly due principally to the reduction in RCS boron concentration associated with fuel burnup. The confirmation that the measured MTC value is within its limit provides assurances that the coefficient will be maintained within acceptable values throughout each fuel cycle.

3/4.1.1.4 MINIMUM TEMPERATURE FOR CRITICALITY

This specification ensures that the reactor will not be made critical with the Reactor Coolant System cold leg temperature less than 545°F. This limitation is required to ensure (1) the moderator temperature coefficient is within its analyzed temperature range, (2) the protective instrumentation is within its normal operating range, (3) a minimum temperature is provided for Special Test Exception 3/4.10.4, and (4) the reactor vessel is above its minimum RT_{NTD} temperature.

BASES

3/4.9.1 BORON CONCENTRATION

The limitations on reactivity conditions during REFUELING ensure that: (1) the reactor will remain subcritical during CORE ALTERATIONS, and (2) a uniform boron concentration is maintained for reactivity control in the water volume having direct access to the reactor vessel. These limitations are consistent with the initial conditions assumed for the boron dilution incident in the safety analyses. The boron concentration limit specified in the COLR is based on core reactivity at the beginning of each cycle (the end of refueling) with all CEAs withdrawn and includes an uncertainty allowance. This boron concentration limit will ensure a K_{eff} of ≤ 0.95 during the refueling operation.

3/4.9.2 INSTRUMENTATION

The OPERABILITY of the startup channel neutron flux monitors ensures that redundant monitoring capability is available to detect changes in the reactivity condition of the core.

3/4.9.3 DECAY TIME

The minimum requirement for reactor subcriticality prior to movement of irradiated fuel assemblies in the reactor pressure vessel ensures that sufficient time has elapsed to allow the radioactive decay of the short lived fission products. This decay time is consistent with the assumptions used in the safety analyses.

3/4.9.4 CONTAINMENT BUILDING PENETRATIONS

The requirements on containment penetration closure and OPERABILITY ensure that a release of radioactive material within containment will be restricted from leakage to the environment. The OPERABILITY and closure restrictions are sufficient to restrict radioactive material release from a fuel element rupture based upon the lack of containment pressurization potential while in the REFUELING MODE.

3/4.9.5 COMMUNICATIONS

The requirement for communications capability ensures that refueling station personnel can be promptly informed of significant changes in the facility status or core reactivity condition during CORE ALTERATIONS.

3/4.9.6 REFUELING MACHINE

The OPERABILITY requirements for the refueling machine ensure that: (1) the machine will be used for movement of fuel assemblies, (2) the machine has sufficient load capacity to lift a fuel assembly, and (3) the core internals and pressure vessel are protected from excessive lifting force in the event they are inadvertently engaged during lifting operations.

3/4.9.7 CRANE TRAVEL - SPENT FUEL STORAGE POOL BUILDING

The restriction on movement of loads in excess of the nominal weight of a fuel assembly, CEA and associated handling tool over other fuel assemblies in the storage pool ensures that in the event this load is dropped (1) the activity release will be limited to that contained in a single fuel assembly, and (2) any possible distortion of fuel in the storage racks will not result in a critical array. This assumption is consistent with the activity release assumed in the safety analyses.

3/4.9.8 SHUTDOWN COOLING AND COOLANT CIRCULATION

The requirement that at least one shutdown cooling loop be in operation, and circulating reactor coolant at a flow rate equal to or greater than 3400 gpm (actual) ensures that (1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor pressure vessel below 135°F as required during the REFUELING MODE, (2) sufficient coolant circulation is maintained through the reactor core to minimize the effects of a boron dilution incident and prevent boron stratification, and (3) the ΔT across the core will be maintained at less than 75°F during the REFUELING MODE. The required flowrate of $\geq$ 3400 gpm (actual) ensures that at 288 hours after reactor shutdown sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor pressure vessel below 135°F as required during REFUELING MODE; this assumes a shutdown cooling heat exchanger cooling water flowrate of 14000 gpm, a cooling water inlet temperature of $\leq$ 105°F at $\geq$ 27 1/2 hours after reactor shutdown, and the decay heat curve of CESSAR-F Figure 6.2.1-1 and reactor operation for two years at 4000 MWt. The 3780 gpm in the specification includes all instrument uncertainties including the 300°F calibration temperature of the flow transmitters.

Without a shutdown cooling train in operation steam may be generated; therefore, the containment should be sealed off to prevent escape of any radioactivity, and any operations that would cause an increase in decay heat should be secured.

The requirement to have two shutdown cooling loops OPERABLE when there is less than 23 feet of water above the reactor pressure vessel flange, ensures that a single failure of the operating shutdown cooling loop will not result in a complete loss of decay heat removal capability. With the reactor vessel head removed and 23 feet of water above the reactor pressure vessel flange, a large heat sink is available for core cooling, thus in the event of a failure of the operating shutdown cooling loop, adequate time is provided to initiate emergency procedures to cool the core.

CORE OPERATING LIMITS REPORT

6.9.1.9 Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT before each reload cycle or any remaining part of a reload cycle for the following:

a. Shutdown Margin - Reactor Trip Breakers Closed for Specification 3.1.1.2
b. Moderator Temperature Coefficient BOL and EOL limits for Specification 3.1.1.3
c. Boron Dilution Alarms for Specification 3.1.2.7
d. Movable Control Assemblies - CEA Position for Specification 3.1.3.1
e. Regulating CEA Insertion Limits for Specification 3.1.3.6
f. Part Length CEA Insertion Limits for Specification 3.1.3.7
g. Linear Heat Rate for Specification 3.2.1
h. Azimuthal Power Tilt - T_q for Specification 3.2.3
i. DNBR Margin for Specification 3.2.4
j. Axial Shape Index for Specification 3.2.7
k. Boron Concentration (Mode 6) for Specification 3.9.1

6.9.1.10 The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC in:

a. "CE Method for Control Element Assembly Ejection Analysis," CENPD-0190-A, January 1976 (Methodology for Specification 3.1.3.6, Regulating CEA Insertion Limits).

b. "The ROCS and DIT Computer Codes for Nuclear Design," CENPD-266-P-A, April 1983 [Methodology for Specifications 3.1.1.2, Shutdown Margin - Reactor Trip Breakers Closed; 3.1.1.3, Moderator Temperature Coefficient BOL and EOL limits; 3.1.3.6, Regulating CEA Insertion Limits and 3.9.1, Boron Concentration (Mode 6)].

c. "Safety Evaluation Report related to the Final Design of the Standard Nuclear Steam Supply Reference Systems CESSAR System 80, Docket No. STN 50-470, "NUREG-0852 (November 1981), Supplements No. 1 (March 1983), No. 2 (September 1983), No. 3 (December 1987) (Methodology for Specifications 3.1.1.2, Shutdown Margin - Reactor Trip Breakers Closed; 3.1.1.3, Moderator Temperature Coefficient BOL and EOL limits; 3.1.2.7, Boron Dilution Alarms; 3.1.3.1, Movable Control Assemblies - CEA Position; 3.1.3.6, Regulating CEA Insertion Limits; 3.1.3.7, Part Length CEA Insertion Limits and 3.2.3 Azimuthal Power Tilt - T_q).

d. "Modified Statistical Combination of Uncertainties," CEN-356(V)-P-A Revision 01-P-A, May 1988 and "System 80™ Inlet Flow Distribution," Supplement 1-P to Enclosure 1-P to LD-82-054, February 1993 (Methodology for Specification 3.2.4, DNBR Margin and 3.2.7 Axial Shape Index).

CORE OPERATING LIMITS REPORT (Continued)

e. "Calculative Methods for the CE Large Break LOCA Evaluation Model for the Analysis of CE and W Designed NSSS," CENPD-132, Supplement 3-P-A, June 1985 (Methodology for Specification 3.2.1, Linear Heat Rate).

f. "Calculative Methods for the CE Small Break LOCA Evaluation Model," CENPD-137-P, August 1974 (Methodology for Specification 3.2.1, Linear Heat Rate).

g. "Calculative Methods for the CE Small Break LOCA Evaluation Model," CENPD-137-P, Supplement 1P, January 1977 (Methodology for Specification 3.2.1, Linear Heat Rate).

h. Letter: O. D. Parr (NRC) to F. M. Stern (CE), dated June 13, 1975 (NRC Staff Review of the Combustion Engineering ECCS Evaluation Model). NRC approval for: 6.9.1.10f.

i. Letter: K. Kniel (NRC) to A. E. Scherer (CE), dated September 27, 1977 (Evaluation of Topical Reports CENPD-133, Supplement 3-P and CENPD-137, Supplement 1-P). NRC approval for 6.9.1.10.g.

The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and analysis limits) of the safety analysis are met.

The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

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. 69
License No. NPF-74

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

   A. The application for amendment by the Arizona Public Service Company (APS or the licensee) 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 dated March 31, 1995, 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 the Act, and the rules and 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; and

   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 attachment 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. 69, 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, except where otherwise stated in specific license conditions.

3. The license amendment is effective as of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Charles R. Thomas

Charles R. Thomas, Project Manager
Project Directorate IV-2
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: September 1, 1995

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 69 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. The corresponding overleaf pages are also provided to maintain document completeness.

REMOVE	INSERT
III*	III*
IV	IV
1-6	1-6
3/4 1-1	4/4 1-1
3/4 1-2	3/4 1-2
3/4 1-3	3/4 1-3
3/4 3-3	3/4 3-3
3/4 3-7*	3/4 3-7*
3/4 3-8	3/4 3-8
3/4 3-16	3/4 3-16
--	3/4 3-16a
3/4 9-1	3/4 9-1
B 3/4 1-1	B 3/4 1-1
B 3/4 1-1a	B 3/4 1-1a
B 3/4 9-1	B 3/4 9-1
B 3/4 9-2*	B 3/4 9-2*
6-20a	6-20a

*No changes were made to these pages; reissued to become overleaf pages.

INDEX

SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

BASES

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

REPORTABLE EVENT

1.28 A REPORTABLE EVENT shall be any of those conditions specified in Sections 50.72 and 50.73 to 10 CFR Part 50.

SHUTDOWN MARGIN

1.29 SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming:

a. No change in part-length control element assembly position, and

b. All full-length control element assemblies (shutdown and regulating) are fully inserted except for the single assembly 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 full-length CEAs must be accounted for in the determination of the SHUTDOWN MARGIN.

SITE BOUNDARY

1.30 The SITE BOUNDARY shall be that line beyond which the land is neither owned, nor leased, nor otherwise controlled by the licensee.

SOFTWARE

1.31 The digital computer SOFTWARE for the reactor protection system shall be the program codes including their associated data, documentation, and procedures.

SOURCE CHECK

1.32 A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to a source of increased radioactivity.

STAGGERED TEST BASIS

1.33 A STAGGERED TEST BASIS shall consist of:

a. A test schedule for n systems, subsystems, trains, or other designated components obtained by dividing the specified test interval into n equal subintervals, and

b. The testing of one system, subsystem, train, or other designated component at the beginning of each subinterval.

THERMAL POWER

1.34 THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - REACTOR TRIP BREAKERS OPEN**

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be greater than or equal to 1.0% delta k/k.

APPLICABILITY: MODES 3, 4* and 5* with the reactor trip breakers open**.

ACTION:

With the SHUTDOWN MARGIN less than 1.0% delta k/k, immediately initiate and continue boration at greater than or equal to 26 gpm to reactor coolant system of a solution containing greater than or equal to 4000 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 1.0% delta k/k at least once per 24 hours by consideration of at least the following factors:

1. Reactor Coolant System boron concentration,
2. CEA position,
3. Reactor Coolant System average temperature,
4. Fuel burnup based on gross thermal energy generation,
5. Xenon concentration, and
6. Samarium concentration.

4.1.1.1.2 The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within ± 1.0% delta k/k at least once per 31 Effective Full Power Days (EFPD). This comparison shall consider at least those factors stated in Specification 4.1.1.1.1, above. The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 EFPD after each fuel loading.

4.1.1.1.3 With the reactor trip breakers open** and any CEA(s) fully or partially withdrawn, the SHUTDOWN MARGIN shall be verified within one hour after detection of the withdrawn CEA(s) and at least once per 12 hours thereafter while the CEA(s) are withdrawn.

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* See Special Test Exception 3.10.9.
**The CEA drive system not capable of CEA withdrawal.

SHUTDOWN MARGIN - REACTOR TRIP BREAKERS CLOSED**

LIMITING CONDITION FOR OPERATION

3.1.1.2

a. The SHUTDOWN MARGIN shall be greater than or equal to that specified in the CORE OPERATING LIMITS REPORT, and

b. For T_{cold} less than or equal to 500°F, K_{N-1} shall be less than 0.99.

c. Reactor criticality shall not be achieved with shutdown group CEA movement.

APPLICABILITY: MODES 1, 2*, 3*, 4*, and 5* with the reactor trip breakers closed**.

ACTION:

a. With the SHUTDOWN MARGIN less than that specified in the CORE OPERATING LIMITS REPORT, immediately initiate and continue boration at greater than or equal to 26 gpm to the reactor coolant system of a solution containing greater than or equal to 4000 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored, and

b. With T_{cold} less than or equal to 500°F and K_{N-1} greater than or equal to 0.99, immediately vary CEA positions and/or initiate and continue boration at greater than or equal to 26 gpm to the reactor coolant system of a solution containing greater than or equal to 4000 ppm boron or equivalent until the required K_{N-1} is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.2.1 With the reactor trip breakers closed**, the SHUTDOWN MARGIN shall be determined to be greater than or equal to that specified in the CORE OPERATING LIMITS REPORT:

a. Within 1 hour after detection of an inoperable CEA(s) and at least once per 12 hours thereafter while the CEA(s) is inoperable.

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* See Special Test Exceptions 3.10.1 and 3.10.9.
**The CEA drive system capable of CEA withdrawal.

SURVEILLANCE REQUIREMENTS (Continued)

b. When in MODE 1 or MODE 2 with k_{eff} greater than or equal to 1.0, at least once per 12 hours by verifying that CEA group withdrawal is within the Transient Insertion Limits of Specification 3.1.3.6. If CEA group withdrawal is not within the Transient Insertion Limits of Specification 3.1.3.6, within 1 hour verify that SHUTDOWN MARGIN is greater than or equal to that specified in the CORE OPERATING LIMITS REPORT.

c. When in MODE 2 with k_{eff} less than 1.0, within 4 hours prior to achieving reactor criticality by verifying that predicted critical CEA position is within the limits of Specification 3.1.3.6.

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of e. below, with the CEA groups at the Transient Insertion Limits of Specification 3.1.3.6.

e. When in MODE 3, 4, or 5, at least once per 24 hours by consideration of at least the following factors:

1. Reactor Coolant System boron concentration,
2. CEA position,
3. Reactor Coolant System average temperature,
4. Fuel burnup based on gross thermal energy generation,
5. Xenon concentration, and
6. Samarium concentration.

4.1.1.2.2 When in MODE 3, 4, or 5, with the reactor trip breakers closed**, and T_{cold} less than or equal to 500°F, K_{N-1} shall be determined to be less than 0.99 at least once per 24 hours by consideration of at least the following factors.

1. Reactor Coolant System boron concentration,
2. CEA position,
3. Reactor Coolant System average temperature,
4. Fuel burnup based on gross thermal energy generation,
5. Xenon concentration, and
6. Samarium concentration.

4.1.1.2.3 When in MODES 3, 4, or 5 with the reactor trip breakers closed**, verify that criticality cannot be achieved with shutdown group CEA withdrawal at least once per 24 hours by consideration of at least the following factors:

1. Reactor Coolant System boron concentration,
2. CEA position,
3. Reactor Coolant System average temperature,
4. Fuel burnup based on gross thermal energy generation,
5. Xenon concentration, and
6. Samarium concentration.

4.1.1.2.4 The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within ± 1.0% delta k/k at least once per 31 Effective Full Power Days (EFPD). This comparison shall consider at least those factors stated in Specification 4.1.1.2.1.e or 4.1.1.2.2. The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 EFPD after each fuel loading.

**The CEA drive system capable of CEA withdrawal.

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.3 The moderator temperature coefficient (MTC) shall be within the area of Acceptable Operation specified in the CORE OPERATING LIMITS REPORT. The maximum uppper limit shall be less than or equal to + 0.5 x 10^{-4} ΔK/K/°F for a power level of 0% RATED THERMAL POWER with a linear ramp to 0ΔK/K/°F at 100% RATED THERMAL POWER.

APPLICABILITY: MODES 1 and 2*#.

ACTION:

With the moderator temperature coefficient outside the area of Acceptable Operation, be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.1.3.1 The MTC shall be determined to be within its limits by confirmatory measurements. MTC measured values shall be extrapolated and/or compensated to permit direct comparison with the above limits.

4.1.1.3.2 The MTC shall be determined at the following frequencies and THERMAL POWER conditions during each fuel cycle:

a. Prior to initial operation above 5% of RATED THERMAL POWER, after each fuel loading.

b. At any THERMAL POWER, within 7 EFPD after reaching a core average exposure of 40 EFPD burnup into the current cycle.

c. At any THERMAL POWER, within 7 EFPD after reaching a core average exposure equivalent to two-thirds of the expected current cycle end-of-cycle core average burnup.

*With K_{eff} greater than or equal to 1.0.

#See Special Test Exception 3.10.2.

TABLE 3.3-1

REACTOR PROTECTIVE INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
I. TRIP GENERATION					
A. Process					
1. Pressurizer Pressure - High	4	2	3	1, 2	2#, 3#
2. Pressurizer Pressure - Low	4	2 (b)	3	1, 2	2#, 3#
3. Steam Generator Level - Low	4/SG	2/SG	3/SG	1, 2	2#, 3#
4. Steam Generator Level - High	4/SG	2/SG	3/SG	1, 2	2#, 3#
5. Steam Generator Pressure - Low	4/SG	2/SG	3/SG	1, 2, 3*, 4*	2#, 3#
6. Containment Pressure - High	4	2	3	1, 2	2#, 3#
7. Reactor Coolant Flow - Low	4/SG	2/SG	3/SG	1, 2	2#, 3#
8. Local Power Density - High	4	2 (c)(d)	3	1, 2	2#, 3#
9. DNBR - Low	4	2 (c)(d)	3	1, 2	2#, 3#
B. Excore Neutron Flux					
1. Variable Overpower Trip	4	2	3	1, 2	2#, 3#
2. Logarithmic Power Level - High					
a. Startup and Operating	4	2 (a)(d)	3	1, 2	2#, 3#
	4	2	3	3*, 4*, 5*	9
b. Shutdown	4	0	2	3, 4, 5	4
C. Core Protection Calculator System					
1. CEA Calculators	2	1	2 (e)	1, 2	6, 7
2. Core Protection Calculators	4	2 (c)(d)	3	1,2,3*,4*,5*	2#, 3#, 7, 10

TABLE 3.3-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION

ACTION STATEMENTS

3.	Steam Generator Pressure - Low	Steam Generator Pressure - Low Steam Generator Level 1-Low (ESF) Steam Generator Level 2-Low (ESF)
4.	Steam Ge· ·ator Level - Low (Wide Range)	Steam Generator Level - Low (RPS) Steam Generator Level 1-Low (ESF) Steam Generator Level 2-Low (ESF)
5.	Core Protection Calculator	Local Power Density - High (RPS) DNBR - Low (RPS)

STARTUP and/or POWER OPERATION may continue until the performance of the next required CHANNEL FUNCTIONAL TEST. Subsequent STARTUP and/or POWER OPERATION may continue if one cnannel is restored to OPERABLE status and the provisions of ACTION 2 are satisfied.

ACTION 4 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, suspend all operations involving positive reactivity changes.

ACTION 5 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, STARTUP and/or POWER OPERATION may continue provided the reactor trip breaker of the inoperable channel is placed in the tripped condition within 1 hour, otherwise, be in at least HOT STANDBY within 6 hours; however, the trip breaker associated with the inoperable channel may be closed for up to 1 hour for surveillance testing per Specification 4.3.1.1.

ACTION 6 -

a. With one CEAC inoperable, operation may continue for up to 7 days provided that the requirements of Specification 4.1.3.1.1 are met. After 7 days, operation may continue provided that the conditions of Action Item 6.b are met.

b. With both CEACs inoperable operation may continue provided that:

TABLE 3.3-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION

ACTION STATEMENTS

1. Within 1 hour the DNBR margin required by Specification 3.2.4b (COLSS in service) or 3.2.4d (COLSS out of service) is satisfied and the Reactor Power Cutback System is disabled, and

2. Within 4 hours:

   a) All full-length and part-length CEA groups must be withdrawn within the limits of Specifications 3.1.3.5, 3.1.3.6b, and 3.1.3.7b, except during surveillance testing pursuant to the requirements of Specification 4.1.3.1.2. Specification 3.1.3.6b allows CEA Group 5 insertion to no further than 127.5 inches withdrawn.

   b) The "RSPT/CEAC Inoperable" addressable constant in the CPCs is set to indicate that both CEAC's are inoperable.

   c) The Control Element Drive Mechanism Control System (CEDMCS) is placed in and subsequently maintained in the "Standby" mode except during CEA motion permitted by Specifications 3.1.3.5, 3.1.3.6b and 3.1.3.7b, when the CEDMCS may be operated in either the "Manual Group" or "Manual Individual" mode.

3. CEA position surveillance must meet the requirements of Specifications 4.1.3.1.1, 4.1.3.5, 4.1.3.6 and 4.1.3.7 except during surveillance testing pursuant to Specification 4.1.3.1.2.

ACTION 7 - With three or more auto restarts, excluding periodic auto restarts (Code 30 and Code 33), of one non-bypassed calculator during a 12-hour interval, demonstrate calculator OPERABILITY by performing a CHANNEL FUNCTIONAL TEST within the next 24 hours.

ACTION 8 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore an inoperable channel to OPERABLE status within 48 hours or open an affected reactor trip breaker within the next hour.

ACTION 9 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE, restore the inoperable channel to OPERABLE status within 48 hours or open the reactor trip breakers within the next hour.

ACTION 10 - In MODES 3, 4, or 5, the Core Protection Calculator channels are not required to be OPERABLE when the Logarithmic Power Level - High trip is OPERABLE with the trip setpoint lowered to $\leq 10^{-4}\%$ of Rated Thermal Power.

TABLE 4.3-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
D. Supplementary Protection System				
Pressurizer Pressure - High	S	R	Q	1, 2
II. RPS LOGIC				
A. Matrix Logic	N.A.	N.A.	Q	1, 2, 3*, 4*, 5*
B. Initiation Logic	N.A.	N.A.	Q	1, 2, 3*, 4*, 5*
III. RPS ACTUATION DEVICES				
A. Reactor Trip Breakers	N.A.	N.A.	M, R(10)	1, 2, 3*, 4*, 5*
B. Manual Trip	N.A.	N.A.	Q	1, 2, 3*, 4*, 5*

TABLE 4.3-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TABLE NOTATIONS

* - With reactor trip breakers in the closed position and the CEA drive system capable of CEA withdrawal, and fuel in the reactor vessel.

(1) - Each STARTUP or when required with the reactor trip breakers closed and the CEA drive system capable of rod withdrawal, if not performed in the previous 7 days.

(2) - Heat balance only (CHANNEL FUNCTIONAL TEST not included):

a. Between 15% and 80% of RATED THERMAL POWER, compare the linear power level, the CPC delta T power and the CPC nuclear power signals to the calorimetric calculation.

If any signal is within -0.5% to 10% of the calorimetric then do not calibrate except as required during initial power ascension after refueling.

If any signal is less than the calorimetric calculation by more than 0.5%, then adjust the affected signal(s) to agree with the calorimetric calculation.

If any signal is greater than the calorimetric calculation by more than 10% then adjust the affected signal(s) to agree with the calorimetric calculation within 8% to 10%.

b. At or above 80% of RATED THERMAL POWER; compare the linear power level, the CPC delta T power and the CPC nuclear power signals to the calorimetric calculation. If any signal differs from the calorimetric calculation by an absolute difference of more than 2%, then adjust the affected signal(s) to agree with the calorimetric calculation.

During PHYSICS TESTS, these daily calibrations may be suspended provided these calibrations are performed upon reaching each major testpower plateau and prior to proceeding to the next major test power plateau.

(3) - Above 15% of RATED THERMAL POWER, verify that the linear power sub-channel gains of the excore detectors are consistent with the values used to establish the shape annealing matrix elements in the Core Protection Calculators.

(4) - Neutron detectors may be excluded from CHANNEL CALIBRATION.

(5) - After each fuel loading and prior to exceeding 70% of RATED THERMAL POWER, the incore detectors shall be used to determine the shape annealing matrix elements and the Core Protection Calculators shall use these elements.

TABLE 4.3-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TABLE NOTATIONS

(6) - This CHANNEL FUNCTIONAL TEST shall include the injection of simulated process signals into the channel as close to the sensors as practicable to verify OPERABILITY including alarm and/or trip functions.

(7) - Above 70% of RATED THERMAL POWER, verify that the total steady-state RCS flow rate as indicated by each CPC is less than or equal to the actual RCS total flow rate determined by either using the reactor coolant pump differential pressure instrumentation or by calorimetric calculations and if necessary, adjust the CPC addressable constant flow coefficients such that each CPC indicated flow is less than or equal to the actual flow rate. The flow measurement uncertainty may be included in the BERR1 term in the CPC and is equal to or greater than 4%.

(8) - Above 70% of RATED THERMAL POWER, verify that the total steady-state RCS flow rate as indicated by each CPC is less than or equal to the actual RCS total flow rate determined by either using the reactor coolant pump differential pressure instrumentation and the ultrasonic flow meter adjusted pump curves or calorimetric calculations.

(9) - The quarterly CHANNEL FUNCTIONAL TEST shall include verification that the correct (current) values of addressable constants are installed in each OPERABLE CPC.

(10) - At least once per 18 months and following maintenance or adjustment of the reactor trip breakers, the CHANNEL FUNCTIONAL TEST shall include independent verification of the undervoltage and shunt trips.

3/4.9 REFUELING OPERATIONS

3/4.9.1 BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.9.1 With the reactor vessel head closure bolts less than fully tensioned or with the head removed, the boron concentration of all filled portions of the Reactor Coolant System and the refueling canal shall be maintained uniform and within the limit specified in the Core Operating Limits Report (COLR).

APPLICABILITY: MODE 6*.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes and initiate and continue boration at greater than or equal to 26 gpm of a solution containing ≥ 4000 ppm boron or its equivalent until the boron concentration is within limits.

SURVEILLANCE REQUIREMENTS

4.9.1.1 The boron concentration shall be determined to be within the limit specified in the COLR prior to:

a. Removing or unbolting the reactor vessel head, and

b. Withdrawal of any full-length CEA in excess of 3 feet from its fully inserted position within the reactor pressure vessel.

4.9.1.2 The boron concentration of the Reactor Coolant System and the refueling canal shall be determined by chemical analysis at least once per 72 hours.

*The reactor shall be maintained in MODE 6 whenever fuel is in the reactor vessel with the reactor vessel head closure bolts less than fully tensioned or with the head removed.

3/4.9.2 INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.9.2 As a minimum, two startup channel neutron flux monitors shall be OPERABLE and operating, each with continuous visual indication in the control room and one with audible indication in the containment and control room.

APPLICABILITY: MODE 6.

ACTION:

a. With one of the above required monitors inoperable or not operating, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

b. With both of the above required monitors inoperable or not operating, determine the boron concentration of the Reactor Coolant System at least once per 12 hours.

SURVEILLANCE REQUIREMENTS

4.9.2 Each startup channel neutron flux monitor shall be demonstrated OPERABLE by performance of:

a. A CHANNEL CHECK at least once per 12 hours,

b. A CHANNEL FUNCTIONAL TEST within 8 hours prior to the initial start of CORE ALTERATIONS, and

c. A CHANNEL FUNCTIONAL TEST at least once per 7 days.

BASES

3/4.1.1 BORATION CONTROL

3/4.1.1.1 and 3/4.1.1.2 SHUTDOWN MARGIN AND K_{N-1}

The function of SHUTDOWN MARGIN is to ensure that the reactor remains subcritical following a design basis accident or anticipated operational occurrence. The function of K_{N-1} is to maintain sufficient subcriticality to preclude inadvertent criticality following ejection of a single control element assembly (CEA). During operation in MODES 1 and 2, with k_{eff} greater than or equal to 1.0, the transient insertion limits of Specification 3.1.3.6 ensure that sufficient SHUTDOWN MARGIN is available.

SHUTDOWN MARGIN is the amount by which the core is subcritical, or would be subcritical immediately following a reactor trip, considering a single malfunction resulting in the highest worth CEA failing to insert. K_{N-1} is a measure of the core's reactivity, considering a single malfunction resulting in the highest worth inserted CEA being ejected.

SHUTDOWN MARGIN requirements vary throughout the core life as a function of fuel depletion and reactor coolant system (RCS) cold leg temperature (T_{cold}). The most restrictive condition occurs at EOL, with T_{cold} at no-load operating temperature, and is associated with a postulated steam line break accident and the resulting uncontrolled RCS cooldown. In the analysis of this accident, the specified SHUTDOWN MARGIN is required to control the reactivity transient and ensure that the fuel performance and offsite dose criteria are satisfied. As (initial) T_{cold} decreases, the potential RCS cooldown and the resulting reactivity transient are less severe and, therefore, the required SHUTDOWN MARGIN also decreases. Below T_{cold} of about 350°F, the inadvertent deboration event becomes limiting with respect to the SHUTDOWN MARGIN requirements. Below 350°F, the specified SHUTDOWN MARGIN ensures that sufficient time for operator actions exists between the initial indication of the deboration and the total loss of shutdown margin. Accordingly, with the reactor trip breakers closed and the CEA drive system capable of CEA withdrawal, the SHUTDOWN MARGIN requirements are based upon these limiting conditions.

Additional events considered in establishing requirements on SHUTDOWN MARGIN that are not limiting with respect to the Specification limits are single CEA withdrawal and startup of an inactive reactor coolant pump.

K_{N-1} requirements vary with the amount of positive reactivity that would be introduced assuming the CEA with the highest inserted worth ejects from the core. In the analysis of the CEA ejection event, the K_{N-1} requirement ensures that the radially averaged enthalpy acceptance criterion is satisfied, considering power redistribution effects. Above T_{cold} of 500°F, Doppler reactivity feedback is sufficient to preclude the need for a specific K_{N-1} requirement. With all CEAs fully inserted, K_{N-1} and SHUTDOWN MARGIN requirements are equivalent in terms of minimum acceptable core boron concentration.

SHUTDOWN MARGIN AND K_{N-1} (continued)

The requirement prohibiting criticality due to shutdown group CEA movement is associated with the assumptions used in the analysis of uncontrolled CEA withdrawal from subcritical conditions. Due to the high differential reactivity worth of the shutdown CEA groups, the analysis assumes that the initial shutdown reactivity is such that the reactor will remain subcritical in the event of unexpected or uncontrolled shutdown group withdrawal.

Other technical specifications that reference the Specifications on SHUTDOWN MARGIN or K_{N-1} are: 3/4.1.2, BORATION SYSTEMS, 3/4.1.3, MOVABLE CONTROL ASSEMBLIES, 3/4.9.1, REFUELING OPERATIONS-BORON CONCENTRATION, 3/4.10.1, SHUTDOWN MARGIN AND K_{N-1} - CEA WORTH TESTS, and 3/4.10.9, SHUTDOWN MARGIN AND K_{N-1} - CEDMS TESTING.

3/4.1.1.3 MODERATOR TEMPERATURE COEFFICIENT (MTC)

The limitations on moderator temperature coefficient (MTC) are provided to ensure that the assumptions used in the accident and transient analysis remain valid through each fuel cycle. The surveillance requirements for measurement of the MTC during each fuel cycle are adequate to confirm the MTC value since this coefficient changes slowly due principally to the reduction in RCS boron concentration associated with fuel burnup. The confirmation that the measured MTC value is within its limit provides assurances that the coefficient will be maintained within acceptable values throughout each fuel cycle.

3/4.1.1.4 MINIMUM TEMPERATURE FOR CRITICALITY

This specification ensures that the reactor will not be made critical with the Reactor Coolant System cold leg temperature less than 545°F. This limitation is required to ensure (1) the moderator temperature coefficient is within its analyzed temperature range, (2) the protective instrumentation is within its normal operating range, (3) a minimum temperature is provided for Special Test Exception 3/4.10.4, and (4) the reactor vessel is above its minimum RT_{NDT} temperature.

3/4.9 REFUELING OPERATIONS

BASES

3/4.9.1 BORON CONCENTRATION

The limitations on reactivity conditions during REFUELING ensure that: (1) the reactor will remain subcritical during CORE ALTERATIONS, and (2) a uniform boron concentration is maintained for reactivity control in the water volume having direct access to the reactor vessel. These limitations are consistent with the initial conditions assumed for the boron dilution incident in the safety analyses. The boron concentration limit specified in the COLR is based on core reactivity at the beginning of each cycle (the end of refueling) with all CEAs withdrawn and includes an uncertainty allowance. This boron concentration limit will ensure a K_{eff} of ≤ 0.95 during the refueling operation.

3/4.9.2 INSTRUMENTATION

The OPERABILITY of the startup channel neutron flux monitors ensures that redundant monitoring capability is available to detect changes in the reactivity condition of the core.

3/4.9.3 DECAY TIME

The minimum requirement for reactor subcriticality prior to movement of irradiated fuel assemblies in the reactor pressure vessel ensures that sufficient time has elapsed to allow the radioactive decay of the short lived fission products. This decay time is consistent with the assumptions used in the safety analyses.

3/4.9.4 CONTAINMENT BUILDING PENETRATIONS

The requirements on containment penetration closure and OPERABILITY ensure that a release of radioactive material within containment will be restricted from leakage to the environment. The OPERABILITY and closure restrictions are sufficient to restrict radioactive material release from a fuel element rupture based upon the lack of containment pressurization potential while in the REFUELING MODE.

3/4.9.5 COMMUNICATIONS

The requirement for communications capability ensures that refueling station personnel can be promptly informed of significant changes in the facility status or core reactivity condition during CORE ALTERATIONS.

3/4.9.6 REFUELING MACHINE

The OPERABILITY requirements for the refueling machine ensure that: (1) the machine will be used for movement of fuel assemblies, (2) the machine has sufficient load capacity to lift a fuel assembly, and (3) the core internals and pressure vessel are protected from excessive lifting force in the event they are inadvertently engaged during lifting operations.

3/4.9.7 CRANE TRAVEL - SPENT FUEL STORAGE POOL BUILDING

The restriction on movement of loads in excess of the nominal weight of a fuel assembly, CEA and associated handling tool over other fuel assemblies in the storage pool ensures that in the event this load is dropped (1) the activity release will be limited to that contained in a single fuel assembly, and (2) any possible distortion of fuel in the storage racks will not result in a critical array. This assumption is consistent with the activity release assumed in the safety analyses.

3/4.9.8 SHUTDOWN COOLING AND COOLANT CIRCULATION

The requirement that at least one shutdown cooling loop be in operation, and circulating reactor coolant at a flow rate equal to or greater than 3400 gpm (actual) ensures that (1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor pressure vessel below 135°F as required during the REFUELING MODE, (2) sufficient coolant circulation is maintained through the reactor core to minimize the effects of a boron dilution incident and prevent boron stratification, and (3) the ΔT across the core will be maintained at less than 75°F during the REFUELING MODE. The required flowrate of $\geq$ 3400 gpm (actual) ensures that at 288 hours after reactor shutdown sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor pressure vessel below 135°F as required during REFUELING MODE; this assumes a shutdown cooling heat exchanger cooling water flowrate of 14000 gpm, a cooling water inlet temperature of $\leq$ 105°F at $\geq$ 27 1/2 hours after reactor shutdown, and the decay heat curve of CESSAR-F Figure 6.2.1-1 and reactor operation for two years at 4000 MWt. The 3780 gpm in the specification includes all instrument uncertainties including the 300°F calibration temperature of the flow transmitters.

Without a shutdown cooling train in operation steam may be generated; therefore, the containment should be sealed off to prevent escape of any radioactivity, and any operations that would cause an increase in decay heat should be secured.

The requirement to have two shutdown cooling loops OPERABLE when there is less than 23 feet of water above the reactor pressure vessel flange, ensures that a single failure of the operating shutdown cooling loop will not result in a complete loss of decay heat removal capability. With the reactor vessel head removed and 23 feet of water above the reactor pressure vessel flange, a large heat sink is available for core cooling, thus in the event of a failure of the operating shutdown cooling loop, adequate time is provided to initiate emergency procedures to cool the core.

CORE OPERATING LIMITS REPORT

6.9.1.9 Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT before each reload cycle or any remaining part of a reload cycle for the following:

a. Shutdown Margin - Reactor Trip Breakers Closed for Specification 3.1.1.2
b. Moderator Temperature Coefficient BOL and EOL limits for Specification 3.1.1.3
c. Boron Dilution Alarms for Specification 3.1.2.7
d. Movable Control Assemblies - CEA Position for Specification 3.1.3.1
e. Regulating CEA Insertion Limits for Specification 3.1.3.6
f. Part Length CEA Insertion Limits for Specification 3.1.3.7
g. Linear Heat Rate for Specification 3.2.1
h. Azimuthal Power Tilt - Tq for Specification 3.2.3
i. DNBR Margin for Specification 3.2.4
j. Axial Shape Index for Specification 3.2.7
k. Boron Concentration (Mode 6) for Specification 3.9.1

6.9.1.10 The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC in:

a. "CE Method for Control Element Assembly Ejection Analysis," CENPD-0190-A, January 1976 (Methodology for Specification 3.1.3.6, Regulating CEA Insertion Limits).

b. "The ROCS and DIT Computer Codes for Nuclear Design," CENPD-266-P-A, April 1983 [Methodology for Specifications 3.1.1.2, Shutdown Margin - Reactor Trip Breakers Closed; 3.1.1.3, Moderator Temperature Coefficient BOL and EOL limits; 3.1.3.6, Regulating CEA Insertion Limits and 3.9.1, Boron Concentration (Mode 6)].

c. "Safety Evaluation Report related to the Final Design of the Standard Nuclear Steam Supply Reference Systems CESSAR System 80, Docket No. STN 50-470, "NUREG-0852 (November 1981), Supplements No. 1 (March 1983), No. 2 (September 1983), No. 3 (December 1987) (Methodology for Specifications 3.1.1.2, Shutdown Margin - Reactor Trip Breakers Closed; 3.1.1.3, Moderator Temperature Coefficient BOL and EOL limits; 3.1.2.7, Boron Dilution Alarms; 3.1.3.1, Movable Control Assemblies - CEA Position; 3.1.3.6, Regulating CEA Insertion Limits; 3.1.3.7, Part Length CEA Insertion Limits and 3.2.3 Azimuthal Power Tilt - Tq).

d. "Modified Statistical Combination of Uncertainties," CEN-356(V)-P-A Revision 01-P-A, May 1988 and "System 80™ Inlet Flow Distribution," Supplement 1-P to Enclosure 1-P to LD-82-054, February 1993 (Methodology for Specification 3.2.4, DNBR Margin and 3.2.7 Axial Shape Index).

CORE OPERATING LIMITS REPORT (Continued)

e. "Calculative Methods for the CE Large Break LOCA Evaluation Model for the Analysis of CE and W Designed NSSS," CENPD-132, Supplement 3-P-A, June 1985 (Methodology for Specification 3.2.1, Linear Heat Rate).

f. "Calculative Methods for the CE Small Break LOCA Evaluation Model," CENPD-137-P, August 1974 (Methodology for Specification 3.2.1, Linear Heat Rate).

g. "Calculative Methods for the CE Small Break LOCA Evaluation Model," CENPD-137-P, Supplement 1P, January 1977 (Methodology for Specification 3.2.1, Linear Heat Rate).

h. Letter: O. D. Parr (NRC) to F. M. Stern (CE), dated June 13, 1975 (NRC Staff Review of the Combustion Engineering ECCS Evaluation Model). NRC approval for: 6.9.1.10f.

i. Letter: K. Kniel (NRC) to A. E. Scherer (CE), dated September 27, 1977 (Evaluation of Topical Reports CENPD-133, Supplement 3-P and CENPD-137, Supplement 1-P). NRC approval for 6.9.1.10.g.

The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and analysis limits) of the safety analysis are met.

The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 98 TO FACILITY OPERATING LICENSE NO. NPF-41,

AMENDMENT NO. 86 TO FACILITY OPERATING LICENSE NO. NPF-51,

AND AMENDMENT NO. 69 TO FACILITY OPERATING LICENSE NO. NPF-74

ARIZONA PUBLIC SERVICE COMPANY, ET AL.

PALO VERDE NUCLEAR GENERATING STATION, UNIT NOS. 1, 2, AND 3

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

1.0 INTRODUCTION

By letter dated March 31, 1995, the Arizona Public Service Company (APS or the licensee) submitted a request for changes to the Technical Specifications (TSs) 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 Arizona Public Service Company submitted this request 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.

The licensee has proposed changes to its Technical Specifications (TS) to be consistent with Combustion Engineering (CE) Revised Technical Specifications, NUREG-1432. These proposed changes would clarify the shutdown margin definition, change the shutdown margin applicability and surveillance requirements to comply with safety analysis assumptions for subcritical inadvertent control element assembly withdrawal (UFSAR Section 15.4), and expand the applicability for core protection calculator (CPC) operability. In addition, the proposed amendment would add a reference to the Core Operating Limits Report for the MODE 6 refueling boron concentration limit. The amendment would also change the power calibration requirements for the linear power level, the CPC delta T power signal, and CPC nuclear power signal to allow more conservative settings than previously requested.

2.0 PROPOSED CHANGES TO THE TECHNICAL SPECIFICATIONS

2.1 TS Definition 1.29

This change adds a sentence to clarify that the reactivity worth of any full-length control element assemblies (CEAs) that are not capable of being fully inserted must be accounted for in the determination of the shutdown margin. This change is consistent with the definition of shutdown margin given in the Combustion Engineering Revised Standard Technical Specifications (NUREG-1432).

2.2 TS Section 3/4.1.1

This change revises the mode applicabilities for Limiting Conditions for Operation (LCOs) 3.1.1.1 and 3.1.1.2, adds a new LCO 3.1.1.2c, adds new Surveillance Requirements (SRs) 4.1.1.1.3 and 4.1.1.2.3 (existing SR 4.1.1.2.3 is renumbered as SR 4.1.1.2.4), and modifies SRs 4.1.1.2.1 and 4.1.I.2.2.

2.3 TS 3.1.1.1

This revision changes the applicability to MODES 3, 4, and 5 with reactor trip breakers open so that the CEA drive system is not capable of withdrawing the CEA. Currently, this TS requires a shutdown margin ≥1.0% delta k/k in MODES 3, 4, and 5 with all full-length CEAs fully inserted.

2.4 TS 3.1.1.2

This revision changes the applicability to MODES 1, 2, 3, 4, and 5 with reactor trip breakers closed so that the CEA drive system is capable of withdrawing the CEA. The SRs for this LCO are changed to be consistent with the revised LCO applicability requirement. Currently, this TS specifies the shutdown margin requirement in MODES 1, 2, 3, 4, and 5 with any full-length CEA fully or partially withdrawn.

2.5 TS 3.1.1.2c

This addition ensures that reactor criticality will not be achieved with shutdown group CEA movement. Currently, this TS does not contain specific requirements to ensure that the reactor will not achieve criticality upon shutdown group CEA movement.

2.6 Surveillance Requirement (SR) 4.1.1.1.3

This addition addresses situations in which one or more CEAs may be fully or partially withdrawn with the reactor trip breakers open (i.e., stuck CEA failing to insert). This additional SR requires that the shutdown margin be determined within one hour after detection of the withdrawn CEA(s) and at least once every 12 hours thereafter while the CEA(s) is withdrawn.

2.7 SR 4.1.1.2.1

This revision deletes the requirement to increase the shutdown margin by an amount equal to an immovable or untrippable CEA, since the proposed clarified shutdown margin definition (TS Section 1.29) contains this requirement.

2.8 SR 4.1.1.2.3

This addition ensures that the safety analysis assumptions are complied with using TS limits rather than administrative controls.

2.9 SR 4.1.1.2.1b

This revision requires that shutdown margin is verified within the limits specified in the COLR when CEA group withdrawal is not within the limits of TS 3.1.3.6. Currently, the TS does not require that shutdown margin be verified if the limits of TS 3.1.3.6 are not met. This change is consistent with the guidance given in NUREG-1432.

2.10 TS Bases Section 3/4.1.1

This revision is consistent with the preceding changes and revises the lower reactor coolant system (RCS) temperature when the inadvertent deboration event becomes limiting from 210°F to 350°F to be consistent with the Core Operating Limits Report (COLR).

2.11 TS Table 3.3-1

This revision requires that either the core protection calculators (CPCs) are Operable in MODES 3, 4, and 5 with the protective system trip breakers in the closed position, the CEA system capable of withdrawing the CEA, and fuel in the reactor vessel, or that the logarithmic power level - High trip is Operable with the trip setpoint lowered to the CPC bypass level (i.e., $\leq 10^{-4}\%$ of rated thermal power).

2.12 TS Table 4.3-1, Table Notation 2

This revision requires adjustment of the linear power level, the CPC delta T power signal, and CPC nuclear power signal to match or be more conservative than the calorimetric power if from 15 percent to 80 percent of RATED THERMAL POWER the difference is less than -0.5 percent or greater than 10 percent and, for greater than 80 percent of RATED THERMAL POWER, the absolute difference is greater than 2 percent.

2.13 TS Sections 3/4.9.1 and 6.9.1

These revisions replace the requirement to maintain the boron concentration in MODE 6 in the RCS and refueling canal at a K_{eff} of ≤ 0.95 or a boron concentration of ≥ 2150 ppm with the requirement to maintain the boron concentration within the limit specified in the COLR.

2.14 TS 6.9.1.9

This revision requires that the TS Section 3.9.1 MODE 6 boron concentration is included in the COLR.

2.15 TS 6.9.1.10

This revision confirms that the NRC-approved ROCS computer code is used to determine the TS Section 3.9.1 MODE 6 boron concentration limit. Upon approval of this proposed TS change, a section would be added to the COLR, noting the minimum refueling boron concentration (in ppm) required to ensure K_{eff} remains ≤ 0.95.

2.16 TS Bases Section 3/4.9.1

This revision confirms that the boron concentration limit specified in the COLR ensures a K_{eff} of ≤ 0.95 during the refueling operation.

3.0 EVALUATION

3.1 Evaluation of Changes Listed in Sections 2.1 through 2.10 above

The above changes to the TSs ensure that the safety analysis assumptions for the subcritical inadvertent CEA bank withdrawal (UFSAR Section 15.4.1) will be valid. The critical safety analysis assumptions that have not been explicitly covered by TSs are that (1) a reactor trip will occur at a power level of $\leq 10^{-2}$% of RATED THERMAL POWER for inadvertent subcritical regulating CEA bank withdrawal and (2) RCS boration is sufficient to prevent criticality based on movement of the shutdown CEA banks. The additional SRs address situations in which one or more CEAs may be partially or fully withdrawn with the reactor trip breakers open, ensuring operation within safety analysis assumptions if one or more CEAs are stuck and do not insert when the reactor trip breakers open. Since inadvertent CEA withdrawal is not possible when the reactor trip breakers are open, it is not necessary to meet the greater shutdown margin that is required when the reactor trip breakers are closed. The above changes are consistent with the guidance provided in NUREG 1432 and are acceptable.

3.2 Evaluation of Changes Listed in Sections 2.11 and 2.12 above

These changes to the TSs will ensure that the allowable power calibration errors during power ascension for the linear power level, the CPC delta T power signal and CPC nuclear power signal are conservative relative to calorimetric power. Raising the tolerance range from ±2 percent to between -0.5 percent and 10 percent from 15 percent to 80 percent of RATED THERMAL POWER will allow more conservative settings than currently required and will meet the analysis assumptions not met by a -2 percent tolerance. The above changes are consistent with the guidance provided in NUREG 1432 and are acceptable.

3.3 Evaluation of Changes Listed in Sections 2.13 through 2.16 above

These changes to the TSs will ensure that the MODE 6 boron concentration is maintained within the limit for boron concentration specified in the COLR. This change enhances the human performance process by giving plant operators the specific boron concentration requirement necessary to ensure the K_{eff} value of ≤ 0.95 required in MODE 6 is met. The boron concentration limit specified in the COLR will be based on core reactivity at the beginning of cycle (the end of refueling) with all CEAs withdrawn and will include an uncertainty allowance. The additional requirement to maintain a boron concentration of at least 2150 ppm is not necessary because maintaining the boron concentration sufficient to ensure a K_{eff} of ≤ 0.95 (based on core reactivity at the beginning of cycle) will ensure the MODE 6 requirement is met to supply the required margin of safety during refueling operation. The licensee conforms to GL 88-16, "Removal of Cycle Specific Parameters Limits from Technical Specifications," in the relocation of the boron concentration limits to the COLR. The cycle specific parameters are consistent with the PVNGS UFSAR and their 50.36 requirements are met.

3.4 Conclusion

The staff has reviewed the licensee's proposed changes and finds that these changes are consistent with Combustion Engineering Revised Technical Specifications (NUREG-1432) and are, therefore, acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Arizona State official was notified of the proposed issuance of the amendments. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and change surveillance requirements. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (60 FR 29871). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). The amendments also involve changes in recordkeeping, reporting or administrative procedures or requirements. Accordingly, with respect to these items, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(10). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

The Commission 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 amendment will not be inimical to the common defense and security or to the health and safety of the public.

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Date: September 1, 1995