May 20, 1991

Docket No. 50-530

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

Dear Mr. Conway:

SUBJECT: ISSUANCE OF AMENDMENT NO. 26 TO FACILITY OPERATING LICENSE, PALO VERDE NUCLEAR GENERATING STATION, UNIT NO. 3 (TAC NO. 79785)

The Commission has issued the enclosed Amendment No. 26 to the Facility Operating License for Palo Verde Nuclear Generating Station, Unit No. 3. The amendment consists of changes to the Technical Specifications in response to your application dated February 21, 1991.

The amendment approves your proposed operating limits and related safety analysis for fuel Cycle 3 operation.

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

Sincerely,

Catherine Thompson

Catherine Thompson, Project Manager
Project Directorate V
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 26 to NPF-74
2. Safety Evaluation

cc w/enclosures:
See next page

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(10)

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

ARIZONA PUBLIC SERVICE COMPANY, ET AL.

DOCKET NO. STN 50-530

PALO VERDE NUCLEAR GENERATING STATION, UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 26
License No. NPF-74

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

   A. The application for amendment by Arizona Public Service Company on behalf of itself and the Salt River Project Agricultural Improvement and Power District, El Paso Electric Company, Southern California Edison Company, Public Service Company of New Mexico, Los Angeles Department of Water and Power, and Southern California Public Power Authority (licensees), dated February 21, 1991, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of Act, and the 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.26 , 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. This license amendment is effective as of the date of issuance and must be fully implemented before unit startup for fuel Cycle 3.

FOR THE NUCLEAR REGULATORY COMMISSION

James E. Dyer, Director
Project Directorate V
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: May 20, 1991

ATTACHMENT TO LICENSE AMENDMENT NO. 26

TO FACILITY OPERATING LICENSE NO. NPF-74

DOCKET NO. STN 50-530

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

Remove	Insert
3/4 1-2a	3/4 1-2a
3/4 1-17	3/4 1-17
3/4 1-18	3/4 1-18
3/4 1-20	3/4 1-20
3/4 1-31	3/4 1-31
3/4 1-32	3/4 1-32
3/4 2-7	3/4 2-7
3/4 2-7a	3/4 2-7a
3/4 2-11	3/4 2-11

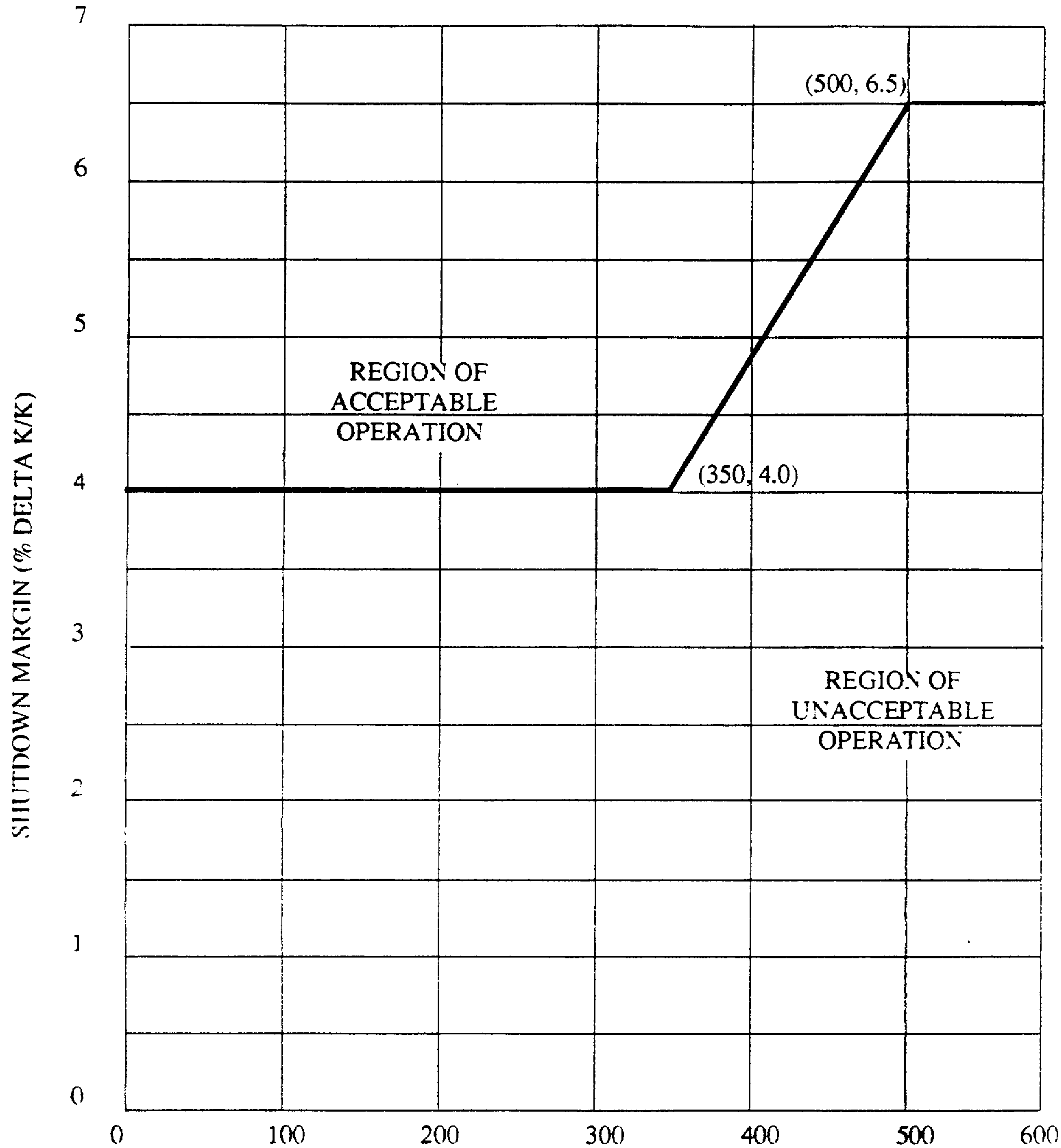


FIGURE 3.1-1A

SHUTDOWN MARGIN VERSUS COLD LEG TEMPERATURE

TABLE 3.1-2

REQUIRED MONITORING FREQUENCIES FOR BACKUP BORON DILUTION DETECTION AS A FUNCTION OF OPERATING CHARGING PUMPS AND PLANT OPERATIONAL MODES FOR $0.98 \geq K_{eff} > 0.97$

OPERATIONAL MODE	Number of Operating Charging Pumps			
	0	1	2	3
3	12 hours	2.0 hours	0.5 hour	ONA
4 not on SCS	12 hours	2.5 hours	1 hour	0.5 hours
5 not on SCS	8 hours	2.5 hours	1 hour	0.5 hours
4 & 5 on SCS	8 hours	0.5 hours	ONA	ONA

Notes: SCS = Shutdown Cooling System
ONA = Operation Not Allowed

TABLE 3.1-3

REQUIRED MONITORING FREQUENCIES FOR BACKUP BORON DILUTION DETECTION AS A FUNCTION OF OPERATING CHARGING PUMPS AND PLANT OPERATIONAL MODES FOR $0.97 \geq K_{eff} > 0.96$

OPERATIONAL MODE	Number of Operating Charging Pumps			
	0	1	2	3
3	12 hours	3.5 hours	1.5 hours	0.5 hour
4 not on SCS	12 hours	3.5 hours	1.5 hours	1 hour
5 not on SCS	8 hours	3.5 hours	1.5 hours	1 hour
4 & 5 on SCS	8 hours	1 hour	0.5 hours	ONA

Notes: SCS = Shutdown Cooling System
ONA = Operation Not Allowed

TABLE 3.1-4

REQUIRED MONITORING FREQUENCIES FOR BACKUP BORON DILUTION DETECTION AS A FUNCTION OF OPERATING CHARGING PUMPS AND PLANT OPERATIONAL MODES FOR $0.96 \geq K_{eff} > 0.95$

OPERATIONAL MODE	Number of Operating Charging Pumps			
	0	1	2	3
3	12 hours	5 hours	2 hours	1 hour
4 not on SCS	12 hours	5 hours	2 hours	1 hour
5 not on SCS	8 hours	5 hours	2 hours	1 hour
4 & 5 on SCS	8 hours	2 hours	0.5 hours	ONA

Notes: SCS = Shutdown Cooling System
ONA = Operation Not Allowed

TABLE 3.1-5

REQUIRED MONITORING FREQUENCIES FOR BACKUP BORON DILUTION DETECTION AS A FUNCTION OF OPERATING CHARGING PUMPS AND PLANT OPERATIONAL MODES FOR $K_{eff} \leq 0.95$

OPERATIONAL MODE	Number of Operating Charging Pumps 0	1	2	3
3	12 hours	6 hours	2.5 hours	1.5 hours
4 not on SCS	12 hours	6 hours	3 hours	1.5 hours
5 not on SCS	8 hours	6 hours	3 hours	1.5 hours
4 & 5 on SCS	8 hours	2 hours	1 hour	0.5 hours
6	24 hours	8 hours	4 hours	2 hours

Note: SCS = Shutdown Cooling System

FIGURE 3.1-3

CEA INSERTION LIMITS VERSUS THERMAL POWER
(COLSS IN SERVICE)

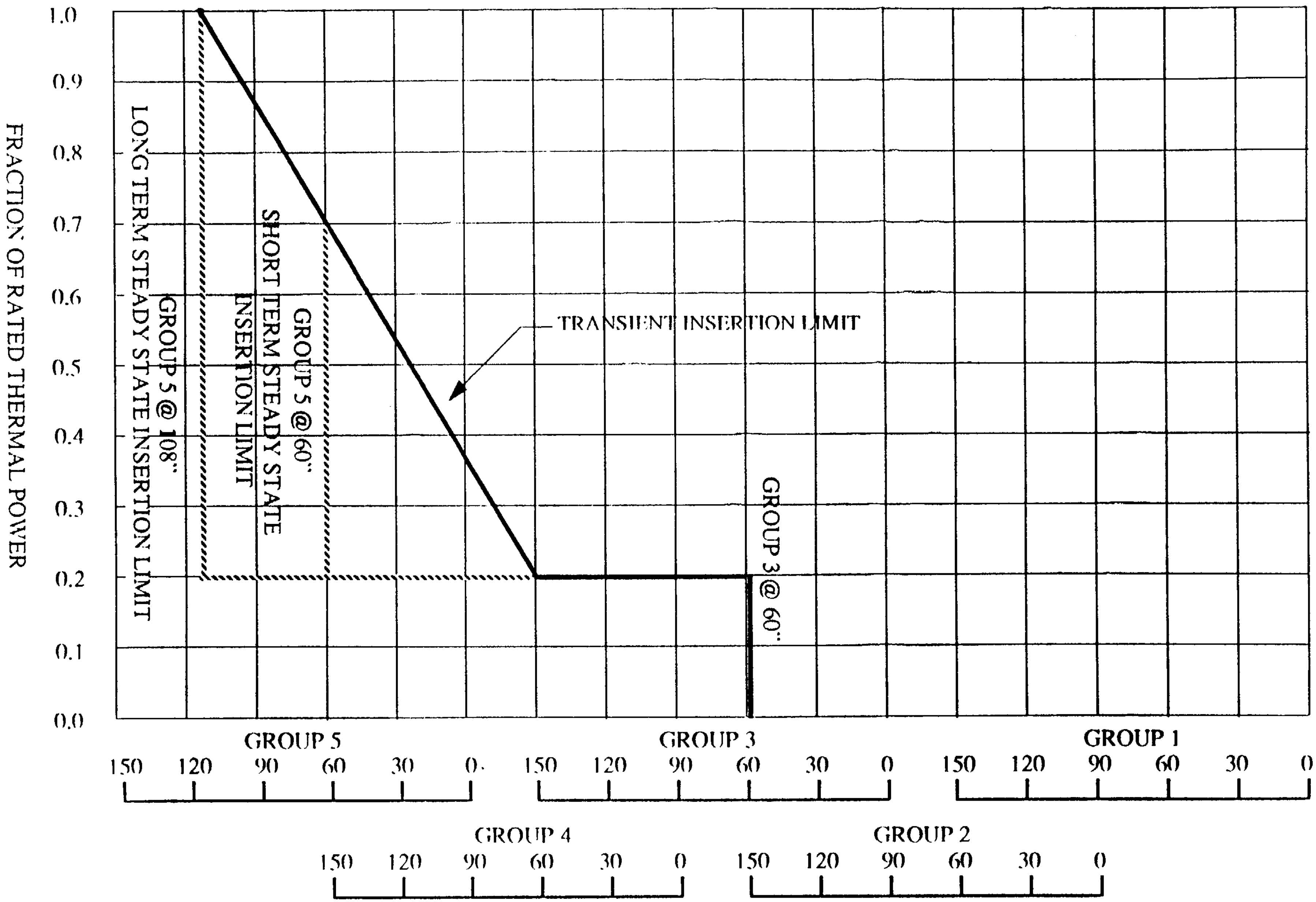
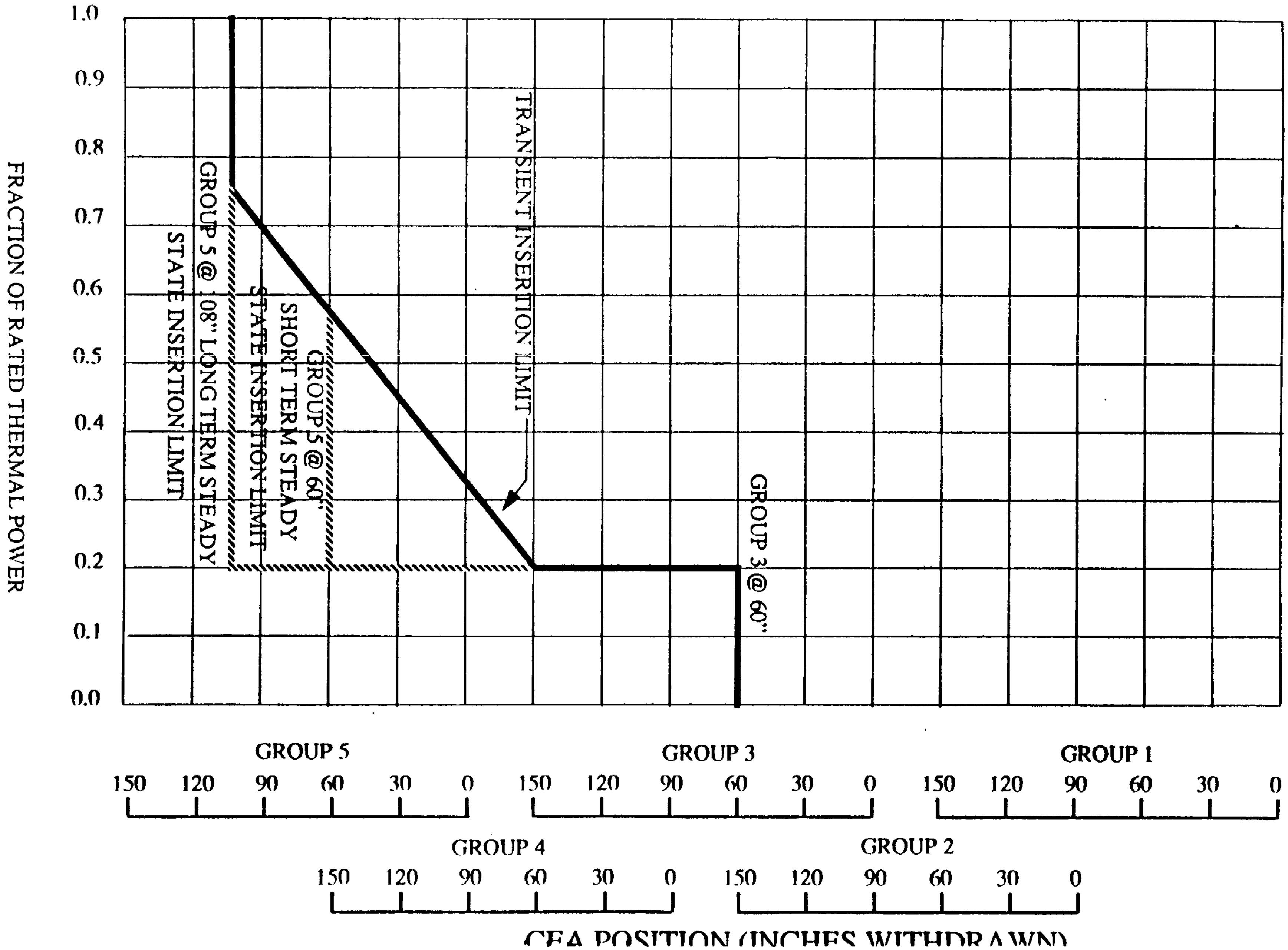
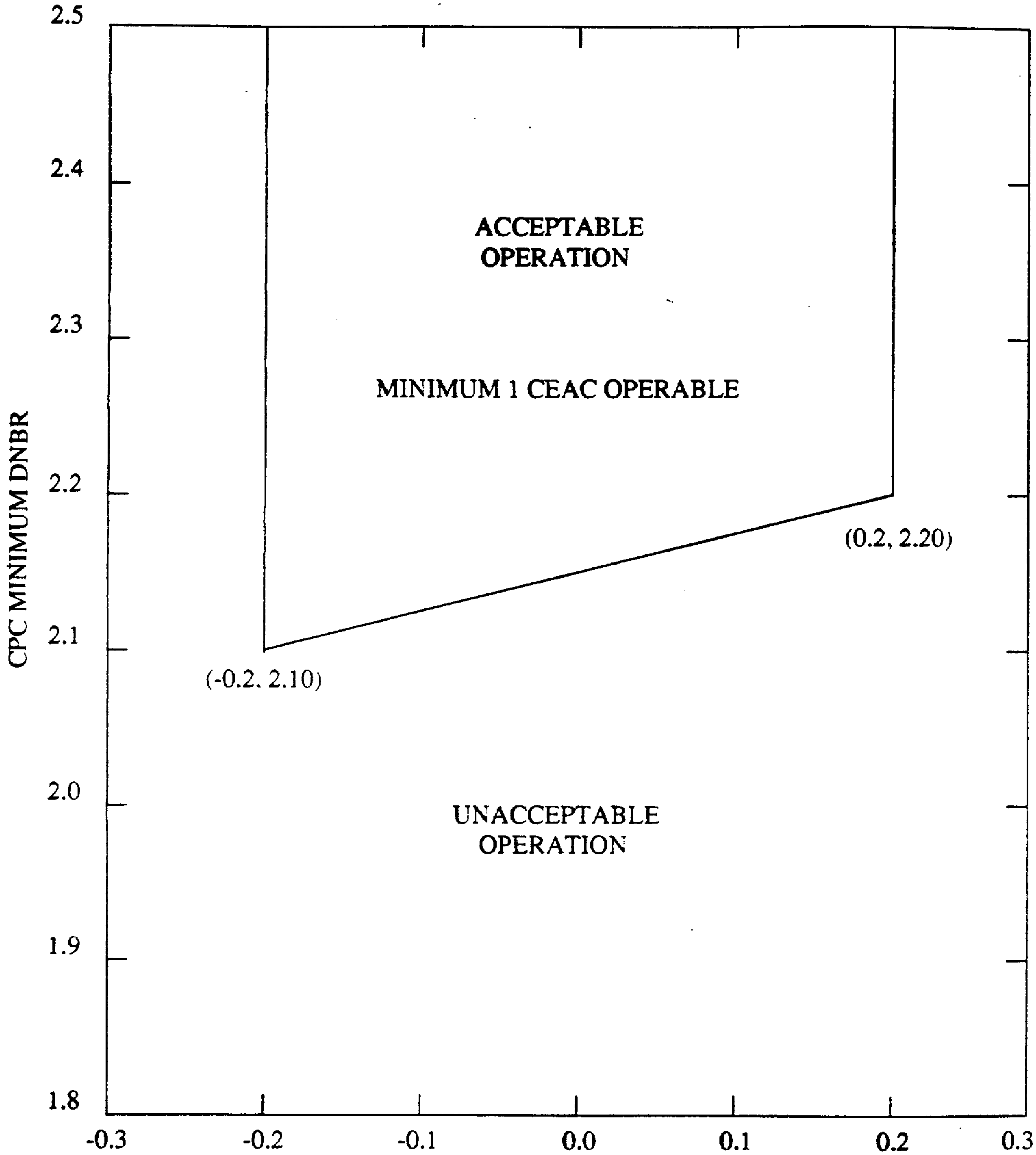


FIGURE 3.1-4

CEA INSERTION LIMITS VERSUS THERMAL POWER
(COLSS OUT OF SERVICE)



COLSS OUT OF SERVICE DNBR LIMIT LINE



CORE AVERAGE ASI

FIGURE 3.2-2

DNBR MARGIN OPERATING LIMIT BASED ON CORE PROTECTION CALCULATORS

(COLSS OUT OF SERVICE, CEACs OPERABLE)

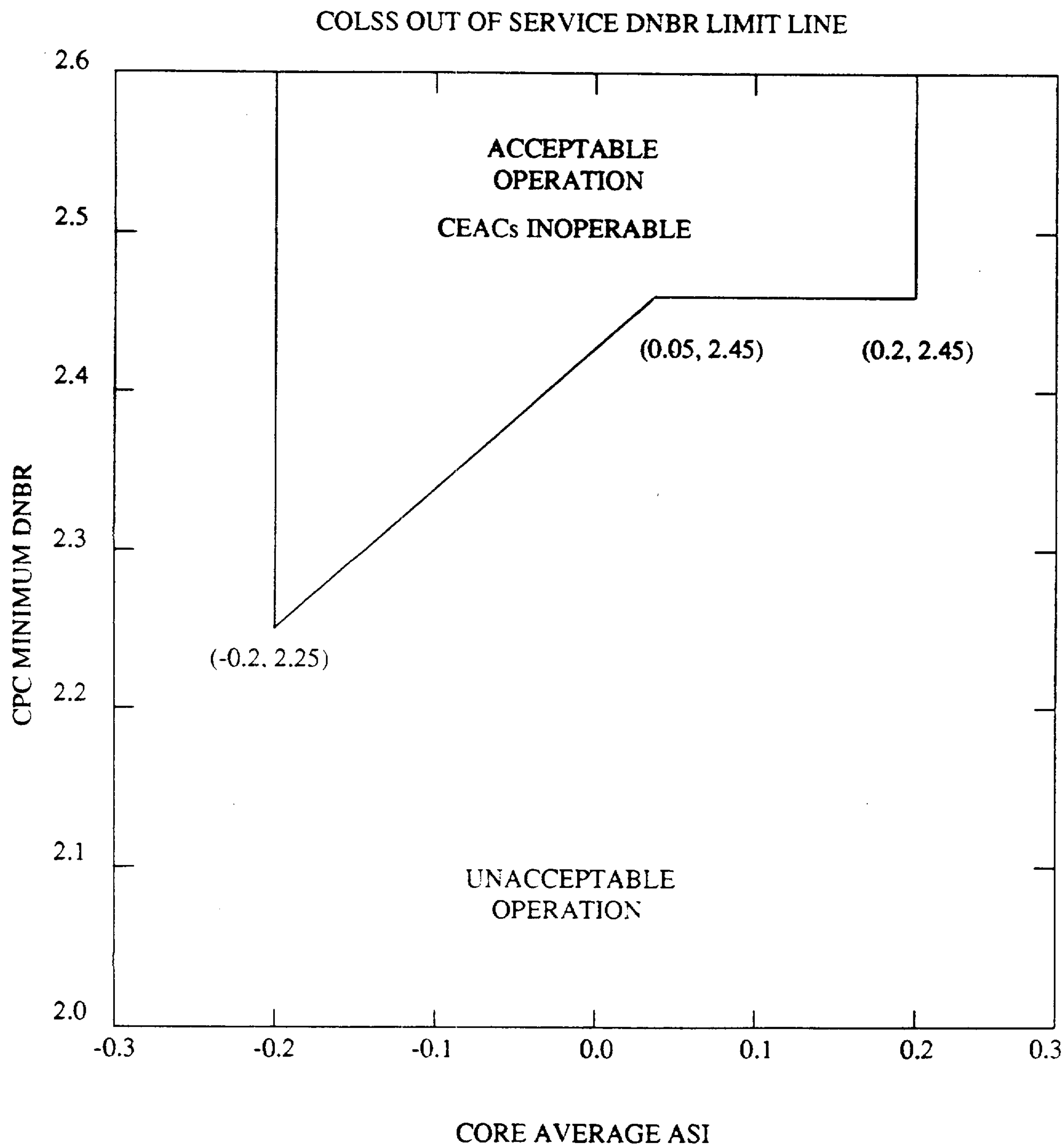


FIGURE 3.2-2A

DNBR MARGIN OPERATING LIMIT BASED ON CORE PROTECTION CALCULATORS
(COLSS OUT OF SERVICE, CEACs INOPERABLE)

3/4.2.7 AXIAL SHAPE INDEX

LIMITING CONDITION FOR OPERATION

3.2.7 The core average AXIAL SHAPE INDEX (ASI) shall be maintained within the following limits:

a. COLSS OPERABLE
$-0.27 \leq ASI \leq 0.27$

b. COLSS OUT OF SERVICE (CPC)
$-0.20 \leq ASI \leq +0.20$

APPLICABILITY: MODE 1 above 20% of RATED THERMAL POWER*.

ACTION:

With the core average AXIAL SHAPE INDEX outside its above limits, restore the core average ASI to within its limit within 2 hours or reduce THERMAL POWER to less than 20% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.7 The core average AXIAL SHAPE INDEX shall be determined to be within its limit at least once per 12 hours using the COLSS or any OPERABLE Core Protection Calculator channel.

*See Special Test Exception 3.10.2.

3/4.2.8 PRESSURIZER PRESSURE

LIMITING CONDITION FOR OPERATION

3.2.8 The pressurizer pressure shall be maintained between 2025 psia and 2300 psia.

APPLICABILITY: MODES 1 and 2*.

ACTION:

With the pressurizer pressure outside its above limits, restore the pressure to within its limit within 2 hours or be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.2.8 The pressurizer pressure shall be determined to be within its limit at least once per 12 hours.

*See Special Test Exception 3.10.5

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 26 TO FACILITY OPERATING LICENSE NO. NPF-74

ARIZONA PUBLIC SERVICE COMPANY, ET AL.

PALO VERDE NUCLEAR GENERATING STATION, UNIT NO. 3

DOCKET NO. STN 50-530

1.0 INTRODUCTION

By letter dated February 21, 1991, Arizona Public Service Company, the licensee for the Palo Verde Nuclear Generating Station Unit 3 (PVNGS3), submitted a reload safety analysis report in support of a request to reload and operate PVNGS3 for a third cycle at 100 percent rated core power of 3800 MWt. The licensee also submitted proposed changes to the Technical Specifications (TS) to support Cycle 3 operation.

The Cycle 3 core will consist of 241 fuel assemblies. Seventy-three Batch B and 48 Batch C assemblies will be removed from the Cycle 2 core and replaced by 88 unirradiated Batch E assemblies. One-hundred and four Batch D assemblies and 16 Batch C assemblies from the Cycle 2 core will be retained. In addition, 33 Batch B assemblies discharged at end of Cycle 1 will be reinserted. Burnup distribution is based on a Cycle 2 length of 436 effective full power days (EFPD). Cycle 3 control element assembly patterns and in-core instrument locations remain the same as in Cycle 2.

The staff has reviewed the licensee's submittal of February 21, 1991, and has prepared the following evaluation of the proposed TS changes, the fuel design, nuclear design, thermal-hydraulic design and accident/transient analyses associated with the Cycle 3 core.

2.0 EVALUATION OF FUEL DESIGN

2.1 Mechanical Design

The 88 Batch E assemblies to be added to the Cycle 3 core are identical in design to the Cycle 2 Batch D assemblies except for changes to the poison rod assembly, the lower end fitting, and center guide tube.

The poison rod assembly was increased in overall length from 160.918 inches to 161.168 inches to improve burnup capability and reduce end-of-life internal pressure. The two-piece lower end fitting was replaced by a one-piece casting with a recess for the center guide tube. The length of the center guide tube was increased from 163.715 inches to 163.965 inches to make it compatible with the redesigned lower end fitting.

The above design changes represent minor improvements which do not affect the fuel mechanical design basis. The staff, therefore, finds these changes acceptable. Also, based on previous staff reload evaluations, clad collapse analyses of new C-E manufactured fuel do not need to be performed because the time to clad collapse is in excess of any practical core residence time.

2.2 Thermal Design

The thermal performance of Cycle 3 fuel was analyzed using the NRC-approved FATES3A code and composite fuel pins that envelope the pins of Batches B, C, D, and E. A power history that enveloped the power and burnup levels of the peak pin at each burnup interval, from the beginning of cycle to the end of cycle, was used. The maximum peak pin burnup analyzed bounds that expected at the end of Cycle 3. Based on this analysis, the internal pressure in the most limiting fuel rod will stay below the nominal reactor coolant system (RCS) pressure of 2250 psi. Because this satisfies Standard Review Plan (SRP) Section 4.2 criteria, the thermal design of the Cycle 3 core is acceptable.

3.0 EVALUATION OF NUCLEAR DESIGN

3.1 Fuel Management

A general description of the Cycle 3 core is given in Section 1.0. The Cycle 3 core uses a low-leakage fuel management scheme where previously burned Batch B assemblies are placed on the periphery and most of the fresh Batch E assemblies are located throughout the core interior in a pattern which minimizes power peaking. The highest Batch E enrichment is 3.96 weight percent U-235; the PVNGS fuel storage facilities are approved for a maximum enrichment of 4.05 weight percent U-235. Expected Cycle 3 lifetime is 390 EFPD. A comparison of the Cycle 3 nominal characteristic physics parameters with those used in the safety analyses show that the latter are conservative in all cases.

3.2 Power Distribution

Calculated "all-rod-out" relative assembly power densities have been presented for beginning of cycle (BOC), middle of cycle, and end of cycle (EOC). Relative assembly power densities are also given at BOC and EOC for rodded configurations allowed by the power dependent insertion limit at full power. These configurations consist of part length CEAs, Bank 5, and Bank 5 plus the part length CEAs. The Cycle 3 nominal axial peaking factors are estimated to range from 1.22 to 1.08, at BOC and EOC, respectively. Physics and power distribution calculations are based on the NRC-approved ROCS and MC codes employing DIT code generated neutron cross-sections. The power distribution calculations are, therefore, acceptable.

3.3 Control Requirements

The value of the required shutdown margin varies throughout core life with the most restrictive value occurring at EOC hot zero power (HZP) conditions. This minimum shutdown margin of 6.5 percent delta k/k is required to control the reactivity transient resulting from the RCS cooldown associated with a steam line break accident at these conditions. For operating temperatures below 350°F, the reactivity transients resulting from any postulated accident are minimal and a 4.0 percent delta k/k shutdown margin (revised from a value of 3.5 for Cycle 2) provides adequate protection. Sufficient boration capability and net available CEA worth, including a minimum worth stuck CEA and appropriate calculational uncertainties, exist to meet these shutdown margin requirements. These results were derived by approved methods and incorporate appropriate assumptions and are, therefore, acceptable.

4.0 EVALUATION OF THERMAL-HYDRAULIC DESIGN

Steady-state thermal-hydraulic analysis for Cycle 3 is performed using the approved thermal-hydraulic code TORC and the CE-1 critical heat flux (CHF) correlation. The design thermal margin analysis is performed with the fast running variation of the TORC code, CETOP-D. The CETOP-D model has been verified to predict minimum departure from nucleate boiling ratio (DNBR) conservatively relative to TORC.

The uncertainties associated with the system parameters are combined statistically using the NRC-approved modified statistical combination of uncertainties methodology. Using this methodology, the engineering hot channel factors for heat flux, heat input, fuel rod pitch, and cladding diameter are combined statistically with other uncertainty factors to arrive at overall uncertainty penalty factors to be applied to the DNBR calculations performed by the core protection calculators (CPCs) and the Core Operating Limit Supervisory System (COLSS). When used with the Cycle 3 DNBR limit of 1.24, these overall uncertainty penalty factors provide assurance with a 95/95 confidence/probability that the hottest fuel rod will not experience DNB.

The 1.24 value incorporates all applicable penalties, such as for rod bow, the 0.01 DNBR for HID-1 grids, and the penalties specified in the statistical combination of uncertainties. The rod bow value used in the analysis is 1.75 percent DNBR, for burnups up to 30,000 MWD/MTU. For burnups higher than 30,000 MWD/MTU, sufficient margin exists to offset the rod bow penalty due to lower radial power peaks in these higher burnup assemblies and rods. Therefore, the rod bow penalty is adequate for all anticipated burnups.

Because the thermal-hydraulic design analyses were performed using approved codes and took into account all applicable penalties, the staff finds these analyses acceptable.

5.0 EVALUATION OF NON-LOCA SAFETY ANALYSIS

The design basis events (DBEs) considered in the safety analyses are categorized into two groups: anticipated operational occurrences (AOOs) and postulated accidents (limiting faults). All events were reviewed by the licensee to assess the need for reanalysis as a result of the new core configuration for Cycle 3. The DBEs were evaluated with respect to the following four criteria: fuel performance (DNBR and centerline melt), RCS pressure, loss of shutdown margin, and offsite dose. The limiting fault events corresponding to each criterion were reanalyzed.

Plant response to the DBEs was simulated using the same methods and computer programs which were used and approved for the Cycle 2 analyses. These include the CESEC III, STRIKIN-II, CETOP-D, TORC, and HERMITE computer programs. For some of the reanalyzed DBEs, certain initial core parameters were assumed to be more limiting than the calculated Cycle 3 values in order to bound future cycles. All of the events reanalyzed have results which are within NRC acceptance criteria and, therefore, are acceptable. Two of the reanalyzed events, however, were not bounded by the Cycle 2 analyses. These are the inadvertent opening of a steam generator safety valve or atmospheric dump valve (ADV) with loss of offsite power and the single reactor coolant pump shaft seizure/sheared shaft event with loss of offsite power and a single active failure of the ADV to close. This single failure for the latter event maximizes the radiological consequences. For the latter event, an increase in predicted fuel failure from 3.79 percent to 4.5 percent occurs. The resulting radiological consequences are within 10 CFR 100 guidelines and therefore, meets the appropriate dose criteria and are acceptable.

For the former event, the amount of predicted failed fuel increased from 8 percent to 12 percent as a result of more adverse nuclear power distributions. The major parameter of concern is the number of fuel rods which experience DNB. This parameter is used to determine if fuel cladding degradation might be anticipated and determines the source for the resulting dose calculations. An ADV may be inadvertently opened by the operator or may be open due to a failure of the control system which operates the valve. The worst single failure for this event is the loss of offsite power concurrent with a turbine trip (LOP) since this combines the greatest decrease in DNBR after initiation of a reactor trip signal with the lowest possible pretrip DNBR. The loss of flow due to the 4 pump coastdown, which results from the assumption of LOP following turbine trip, causes a greater decrease in DNBR after reactor trip than other possible single failures. In addition to the assumed single failure of loss of offsite power, the most reactive CEA is assumed to be stuck in the fully withdrawn position following reactor trip. The licensee indicated that the ADVs are air operated and are spring loaded to fail closed on loss of air. For the ADV to open and remain open, there must be 6 failures involving 2 channels of DC power. In order for an inadvertently opened ADV to remain open due to mechanical binding, the valve would need to seize up so firmly that it could not be closed neither by air pressure, spring nor manual handwheel operation.

The FSAR used a deterministic method of predicting fuel rod failure in which any fuel rod falling below a DNBR limit of 1.24 was assumed to experience cladding failure with all of the activity in the fuel-clad gap released to the primary coolant. The Cycle 3 analysis used a statistical convolution approach in which the probability of being in DNB at a given DNBR is taken into account. As in the deterministic method, a fuel rod is still assumed to fail if it experiences DNB. This approach had been found acceptable by the staff for analysis of limiting faults such as the locked rotor, sheared shaft and CEA ejection accidents at PVNGS. These are occurrences that are not expected to occur but are postulated because their consequences would include the potential for the release of significant amounts of radioactive material. Although the inadvertent opening of an ADV with loss of offsite power is not categorized as a limiting fault but rather as an infrequent incident which may be expected to occur during the lifetime of a plant, the staff concludes that the use of statistical convolution to determine fuel failures for this event is acceptable for PVNGS. This conclusion is based on the unique design of the PVNGS ADVs and the resulting low probability of failure to close which was discussed previously. It is also based on the conservative transient assumptions mentioned previously as well as the conservatisms inherent in the methodology, such as the fact that all DNBR values are calculated assuming an assembly inlet flow equal to 73% of the core average. A realistic flow distribution would result in a much smaller number of failures.

For this event, the staff practice has been to restrict the resulting two hour site boundary doses to a "small fraction" (10% or less) of 10 CFR 100 guidelines of 300 rem to the thyroid and 25 rem to the whole body. The resultant offsite dose for this event using the statistical convolution method has been calculated by the licensee to be 30 rem thyroid and less than 2 rem whole body, thereby meeting the staff's criterion.

In conclusion, the staff finds the results of the inadvertent opening of a steam generator ADV with a loss of offsite power using a statistical convolution of DNB are acceptable for PVNGS 3 Cycle 3. Since the dose consequences of this meet the acceptable limit of 10% of 10 CFR 100, this analysis is acceptable as the reference analysis for PVNGS. All future reload analyses performed for this event must use the same assumptions and methods used for the Unit 3, Cycle 3 analyses described above. Changes to these assumptions and methods should be submitted to the staff including a discussion of why the statistical convolution methodology remains acceptable for this event.

6.0 EVALUATION OF ECCS ANALYSIS

An ECCS analysis was performed for the limiting break size LOCA (a double-ended guillotine break with a 1.0 discharge coefficient) for Cycle 3 to demonstrate compliance with the requirements of 10 CFR 50.46. The methodology is the same as for the cycle 2 analysis. The analysis justifies a 13.5 kw/ft peak linear heat generation rate. Because there have been no significant changes in hardware characteristics for Cycle 3, only fuel rod clad temperature and oxidation calculations were performed. The code STRIKIN-II was used for this purpose and the fuel performance data were generated using the FATES-3A fuel evaluation code. It was demonstrated that burnup with the highest initial fuel stored energy was limiting. The ECCS analysis methods employed have been previously approved and are acceptable.

The results of the limiting break LOCA analysis for Cycle 3 are bounded by the results obtained in the Cycle 2 analysis, i.e., a peak clad temperature of 2091°F, a maximum local clad oxidation of 9.0 percent, and a core wide clad oxidation of less than 0.80 percent. These values are within the 10 CFR 50.46 limits of 2200°F, 17.0 percent, and 1.0 percent, respectively, and are, therefore, acceptable. Similarly, a review of Cycle 3 fuel and core data has confirmed that the small break LOCA analysis results are bounded by the Cycle 2 analysis.

7.0 TECHNICAL SPECIFICATION CHANGES

TS Figure 3.1-1A

The proposed change increases the required shutdown margin from 3.5 to 4.0 percent delta k/k for the RCS cold leg temperature range zero to 350°F when any full-length CEA is fully withdrawn.

The increased shutdown margin will ensure that the TS are consistent with the safety analyses performed for the Cycle 3 core and that the consequences of DBEs and anticipated operational occurrences are bounded by these analyses. The proposed change is therefore acceptable.

TS Tables 3.1-2, 3.1-3, and 3.1-5

These tables provide frequencies for monitoring RCS boron concentration in the event that one or both startup channel high neutron flux alarms are inoperable.

The proposed changes are more restrictive in that certain monitoring frequencies are increased to ensure that the TS are consistent with the safety analyses performed for the Cycle 3 core and that, in the event of an inadvertent boron dilution, sufficient time will be available to terminate the event prior to loss of shutdown margin. The proposed changes are, therefore, acceptable.

TS Figures 3.1-3 and 3.1-4

Figures 3.1-3 and 3.1-4 provide regulating group CEA insertion limits when the COLSS is in service and out of service, respectively. The proposed change to Figure 3.1-3 will prohibit insertion of regulating group 3 CEAs above 20 percent of rated thermal power. This is permitted under the existing TS. The proposed change to Figure 3.1-4 will permit slightly increased insertion of regulating group 3 CEAs between 15 percent and 20 percent of rated thermal power.

The proposed revisions are necessary to ensure consistency of the TS with the safety analyses performed for the Cycle 3 core. These analyses demonstrate that reactor operation in accordance with the revised insertion limits will ensure that the Specified Acceptable Fuel Design Limits (SAFDLs) will not be exceeded during the most limiting anticipated operational occurrence. The proposed changes are, therefore, acceptable.

TS 3.2.7a

TS 3.2.7a ensures that the actual value of the core average Axial Shape Index (ASI) remains within the range of values used in the safety analyses when the COLSS is operable. The proposed change revises the limits of core average ASI from between -.28 to +.28 to between -.27 to +.27 to make the TS consistent with the safety analyses performed for the Cycle 3 core. The proposed change is, therefore, acceptable.

TS Figures 3.2-2 and 3.2-2A

Figure 3.2-2 provides DNBR margin limits when at least one Control Element Assembly Calculator (CEAC) is operable and the COLSS is out of service. Figure 3.2-2A provides the additional DNBR margin necessary when COLSS and both CEACs are out of service. Reactor operation within these limits ensures that the SAFDLs will not be violated during an anticipated operational occurrence.

The proposed changes are necessary to ensure consistency of the TS with the safety analyses performed for the Cycle 3 core and are, therefore, acceptable.

8.0 STARTUP TESTING

The licensee has presented a brief description of the low power physics tests and the power ascension testing to be performed during Cycle 3 startup. The described tests will verify that core performance is consistent with the engineering design and safety analyses. If the acceptance criterion of any of the startup physics tests are not met, an evaluation will be performed by the licensee. Resolution will be required prior to subsequent power escalation. If an unreviewed safety question is involved, the NRC will be notified.

The staff has reviewed the proposed startup test program for Cycle 3 and finds that it conforms to accepted practices and adequately supplements normal surveillance tests which are required by the plant Technical Specifications.

9.0 EVALUATION FINDINGS

The staff has reviewed the fuels, physics, and thermal-hydraulics information presented in the PVNGS3 Cycle 3 reload report. Also reviewed were the Technical Specification revisions, the startup test procedures, and the safety reanalyses. Based on the evaluations given in the preceding sections, the staff finds the proposed reload acceptable.

10.0 STATE CONSULTATION

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

11.0 ENVIRONMENTAL CONSIDERATION

The amendment changes 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. The NRC staff has determined that the amendment involves 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 amendment involves no significant hazards consideration, and there has been no public comment on such finding. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR

51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

12.0 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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