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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 61 TO FACILITY OPERATING LICENSE NO. NPF-41

ARIZONA PUBLIC SERVICE COMPANY, ET AL.

PALO VERDE NUCLEAR GENERATING STATION, UNIT NO. 1

DOCKET NO. STN 50-528

1.0 INTRODUCTION

By letter dated December 24, 1991, the Arizona Public Service Company (APS or the licensee) submitted a request for changes to the Technical Specifications (TS) for the Palo Verde Nuclear Generating Station, Unit 1 (Appendix A to Facility Operating License No. NPF-41. 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 proposed changes would revise the technical specifications to be consistent with the reload safety analysis for operation in fuel cycle 4.

2.0 DISCUSSION

The Unit 1 cycle 4 core will consist of 241 fuel assemblies. One batch B assembly, 52 batch C assemblies, and 44 batch D assemblies will be removed from the cycle 3 core to make way for 88 fresh batch F assemblies; 108 batch E and 36 batch D assemblies now in the core will be retained. In addition, 5 batch B assemblies originally discharged at end of cycle 1 (EOC1) and 4 batch C assemblies originally discharged at the end of cycle 2 (EOC2), will be reinserted from the spent fuel storage. The burnup distribution is based on a cycle 3 length of 517 EFPD.

Control element assembly patterns and in-core instrument locations will remain unchanged from those used in cycle 3 (the reference cycle).

The staff has reviewed the licensee submittal and has prepared the following evaluation of the proposed technical specification (TS) changes, the fuel design, nuclear design, thermal-hydraulic design and accident/transient analyses associated with the cycle 4 core.

3.0 EVALUATION

3.1 Mechanical Design

The 88 batch F assemblies to be added to the cycle 4 core are identical in design to the cycle 3 batch E assemblies except for changes to the lower end fitting and center guide tube design. The lower end fitting design was changed from a two piece assembly to a single piece casting with a recess for the center guide tube to fit within the flow plate. The length of the center guide tube was increased from 163.715 inches to 163.965 inches in order to fit the new lower end fitting. The new design provides improved strength, stiffness, and quality in the 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.

3.2 Thermal Design

The thermal performance of cycle 4 fuel was analyzed using the NRC-approved code and composite fuel pins that envelope the power and burnup levels representative of the peak pin at each burnup interval, from the beginning of cycle to end of cycle burnups. The maximum peak pin burnup analyzed bounds that expected at the end of cycle 4. Based on this analysis, the internal pressure in the most limiting fuel rod will stay below the nominal reactor coolant system (RCS) pressure throughout the cycle. Because this satisfies Standard Review Plan (SRP) Section 4.2 criteria, the thermal design of the cycle 4 core is acceptable.

4.0 EVALUATION OF NUCLEAR DESIGN

4.1 Fuel Management

The cycle 4 core uses a low-leakage fuel management scheme where previously burned assemblies are placed on the periphery and most of the fresh assemblies are located throughout the core interior in a pattern which minimizes power peaking. With this loading and a cycle 3 endpoint of 491 EFPD, the cycle 4 reactive lifetime for full power operation is expected to be 400 EFPD. A comparison of the cycle 4 nominal characteristic physics parameters with those used in the safety analyses show that the latter are conservative in all cases.

4.2 Power Distribution

Calculated "all rods 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. The rodded configurations consist of part length CEAs, Bank 5, and Bank 5 plus part length CEAs. The cycle 4 nominal axial peaking factors are estimated to range from 1.16 at BOC4 to 1.08 at EOC4. 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.

4.3 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. 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.

4.4 Revised Biases and Uncertainties

Palo Verde Unit 1 has implemented improved calculational methods and a revised and extended data base to analyze the nuclear design of the core. Changes were made to these calculational methods and computer codes in order to simplify the use, improve their computational efficiency and to enhance their calculational accuracy. The physics biases and uncertainties associated with these new calculational methods, are described in C-E's proprietary Report CE-CES-129 Rev. 1-P. APS has provided a 10 CFR 50.59 safety evaluation report which states that the methods used to generate the new biases and uncertainties are the same (with the exception of the method used to calculate the N-1 rod worth) as those described in C-E's proprietary Topical Report CENPD-266-P-A, which was generically approved by the NRC.

Some of the new features included in the new methodology account for anisotropic scattering within cells and cell interfaces, assembly discontinuity factors, nodal expansion methods, and improved xenon and soluble boron treatment.

The new biases and uncertainties associated with the new calculations were selectively applied to calculations of reactivity, reactivity coefficients, rod worth and power peaking factors. The new calculational methods did not significantly impact such things as shim heating, fluence, and burnup calculations; consequently, the biases and uncertainties associated with these calculations retained their original values.

The biases and uncertainties associated with the N-1 rod worth were determined by setting the net rod worth uncertainty equal to the total worth uncertainty rather than the bank worth uncertainty. This is more representative of the higher control rod density of the N-1 configuration.

The parameters determining the biases and uncertainties were presented in tabulated form in the CE-CES-129 Rev. 1-P report, along with best estimate values and tolerance limits. When insufficient data existed to carry out technically competent statistical calculations, limiting bounding values were presented.

The changes incorporated into the new calculational methods contributed to showing that there is generally more scram worth available than previous calculations suggested. In all cases in the above calculations, the 95/95 tolerance limit was maintained between the calculated and the measured results. Based on the above analysis of the various documents submitted by the licensee, particularly document CE-CES-129 Rev. 1-P, specifically addressing the physics biases and uncertainties, the NRC staff finds the new methodologies and computer codes, and the associated biases and uncertainties to be acceptable for application to Palo Verde Unit 1.

Since CE-CES-129 Rev. 1-P is not an NRC-approved document, any future reference to the authorization for use of these biases and uncertainties for the analysis of Palo Verde Unit 1 should reference this Safety Evaluation.

5.0 EVALUATION OF THERMAL-HYDRAULIC DESIGN

Steady-state thermal hydraulic analysis for cycle 4 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 computer code, CETOP. The CETOP 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 (MSCU) 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 an overall uncertainty for use in 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 4 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 MSCUs. The rod bow value used in the analysis is 1.75 percent DNBR, for burnups up to 30 GWD/MTU. For burnups higher than 30 GWD/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.

6.0 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 4. 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 3 analyses. These include the CESEC III, STRIKIN-II, CETOP, TORCH, and HERMITE computer programs. For some of the reanalyzed DBEs, certain initial core parameters were assumed to be more limiting than the calculated cycle 4 values in order to bound future cycles. All of the events reanalyzed have results which are within NRC acceptance criteria and, therefore, are acceptable. The analytical methodology used for PVNGS-1 cycle 4 is the same as that used for Unit 1 cycle 3 with the exception of event 7.1.4, the Inadvertent Opening of a Steam Generator Safety Valve or Atmospheric Dump Valve with a loss of Offsite Power; unit 3 cycle 3 is the reference cycle for the analysis of this event because it incorporates the current NRC position. Only methodology that has previously been reviewed and approved on the PVNGS dockets and/or the CESSAR docket were used.

Since event 7.1.4 is bounded by the reference cycle, the radiological consequences are within 10 CFR Part 100 guidelines and, therefore, meets the appropriate dose criteria and is acceptable.

7.0 EVALUATION OF ECCS ANALYSIS

An ECCS performance analysis of the limiting break size was performed for PVNGS1 cycle 4 to demonstrate compliance with 10 CFR 50.46. The methodology is the same as that for cycle 3 analysis. The analysis justifies an allowable peak power linear heat generation rate (PLHGR) of 13.5 kw/ft. Because there have been no significant changes in hardware characteristics for cycle 4, 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 the 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 4 are bounded by the results of the ECCS performance reference cycle, PVNGS-1 cycle 1, 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 0.8 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 4 fuel and core data has confirmed that the small break LOCA analysis results are bounded by the reference cycle analysis.

8.0 TECHNICAL SPECIFICATION CHANGES

8.1 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 core operating limit supervisory system (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 specified acceptable fuel design limits (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 cycle 4, and are therefore acceptable.

9.0 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 4 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 4 and finds that it conforms to accepted practices and adequately supplements normal surveillance tests which are required by the plant technical specifications.

10.0 SUMMARY

The NRC staff has reviewed the material submitted by the licensee for the reload of PVNGS-1 cycle 4 and determined that it presents an appropriate change for Technical Specification Figures 3.2-2 and 3.2-2A. Also reviewed were the fuels, physics, and thermal-hydraulics information presented in the PVNGS-1 cycle 4 reload report. Based on our review, we find the proposed reload and associated TS changes acceptable.

11.0 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.

12.0 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 considera-tion, and there has been no public comment on such finding (57 FR 2586). 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.

13.0 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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