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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 19 TO FACILITY OPERATING LICENSE NO. NPF-51

ARIZONA PUBLIC SERVICE COMPANY, ET AL.

PALO VERDE NUCLEAR GENERATING STATION, UNIT NO. 2

DOCKET NO. STN 50-529

1.0 INTRODUCTION

By letter dated December 3, 1987 (Ref. 1), the Arizona Public Service Company (APS) on behalf of itself, the Salt River Project Agricultural Improvement and Power District, Southern California Edison Company, El Paso Electric Company, Public Service Company of New Mexico, Los Angeles Department of Water and Power, and Southern California Public Power Authority (licensees), requested several changes to the Technical Specifications (Appendix A to Facility Operating License No. NPF-51) for the Palo Verde Nuclear Generating Station, Unit 2 (PVNGS2), relating to Cycle 2 operation for PVNGS2. In support of both the Technical Specification changes and Cycle 2 operation, the licensees submitted a Reload Analysis Report by letter dated December 2, 1987 (Ref. 2). By letter dated February 4, 1988 (Ref. 3), the licensees also provided clarifying information on the Reload Analysis Report, in response to staff questions to verify that all events affected by changes in the Moderator Temperature Coefificient were analyzed. The staff's evaluation of the reload analysis is presented in Sections 2.0 through 5.0 below. The evaluation of the specific changes to the Technical Specifications is presented in Section 6.0 below.

2.0 EVALUATION OF FUEL DESIGN

2.1 Mechanical Design

The Cycle 2 core consists of 241 fuel assemblies. One hundred and eight fresh (unirradiated) Batch D assemblies will replace 69 Batch A assemblies and 39 Batch B assemblies. The remaining 69 Batch B assemblies and all Batch C assemblies in the core during Cycle 1 will be retained.

The 108 Batch D assemblies will consist of 32 type D0 assemblies with 4.02 weight percent (w/o) and 3.57 w/o U-235 enriched fuel rods, 20 type D1 assemblies with 4.02 w/o and 3.57 w/o U-235 enriched rods and 16 burnable poison shims per assembly, 8 type D2 assemblies with 4.02

w/o and 3.57 w/o U-235 enriched rods and 16 burnable poison shims per assembly, 16 type D3 assemblies with 3.57 w/o and 3.09 w/o U-235 enriched rods and 16 burnable poison shims per assembly, four type D4 assemblies with 3.57 w/o and 3.09 w/o U-235 enriched rods and 12 burnable poison shims per assembly, and 28 type D5 assemblies with 3.57 w/o and 3.09 w/o U-235 enriched rods and 12 burnable poison shims per assembly. The mechanical design of the Batch D assemblies is identical to that of the Batch C assemblies used in Cycle 1 except for design features which were incorporated to improve fuel handling, the fabricability and quality of the fuel and the burnup capability of the poison rods. The staff, therefore, finds these modifications to be acceptable.

Attachment 5 to Reference 5 is a report entitled, "Evaluation of Interpellet Gap Formation and Clad Collapse in Modern PWR Fuel Rods," on work performed by Combustion Engineering (CE) for the Electric Power Research Institute (EPRI). The report presents the results from a review of interpellet gap formation, ovality, creepdown and clad collapse data in modern PWR fuel rods. The report concludes that modern CE fuel rods have a time to clad collapse in excess of any practical core residence time. The staff concurs with the conclusions of the CE report as it applies to PVNGS2 Cycle 2. This concurrence is supported by previous acceptance for PVNGS1 Cycle 2 and Waterford 3 Cycle 2 and by similar results of analyses by another fuel vendor. Therefore, the staff concludes that no further analysis of clad collapse need be performed for PVNGS2 Cycle 2.

2.2 Thermal Design

The thermal performance of Cycle 2 fuel was performed by analyzing a composite fuel pin that envelopes the peak pins of the various fuel assemblies (fuel Batches B, C, and D) in the Cycle 2 core using the NRC approved fuel performance code FATES3A. The NRC imposed grain size restriction (Ref. 13) was included and a power history was used that envelopes the power and burnup levels representative of the peak pin at each burnup interval from beginning-of-cycle (BOC) to end-of-cycle (EOC). The maximum peak pin burnup analyzed for Cycle 2 bounds the expected EOC maximum fuel rod burnup. Based on this analysis, the internal pressure in the most limiting hot rod will not reach the nominal reactor coolant system (RCS) pressure of 2250 psia. Since this satisfies the fuel rod internal gas pressure requirement of Standard Review Plan (SRP) 4.2, Section II.A.1(f), the staff finds it acceptable and concludes that the fuel rod internal pressure limits have been adequately considered for Cycle 2 operation.

3.0 EVALUATION OF NUCLEAR DESIGN

3.1 Fuel Management

The PVNGS2 Cycle 2 core consists of 241 fuel assemblies, each having a 16 by 16 fuel rod array. A general description of the core loading is given above in Section 2.1. The highest U-235 enrichment occurs in the Batch D fuel assemblies which contain fuel rods with 4.02 weight percent U-235. The PVNGS2 fuel storage facilities have been approved for storage of fuel with a maximum U-235 enrichment of 4.05 weight percent (Ref. 4).

The Cycle 2 core will use a low-leakage fuel management scheme in which the previously irradiated Batch B assemblies are placed on the core periphery. Most of the fresh Batch D assemblies are placed in the interior of the core and mixed with the previously irradiated fuel to minimize power peaking. With this loading and a Cycle 1 endpoint of 16,512 MWD/MTU, the Cycle 2 reactivity lifetime for full power operation is expected to be 16,945 MWD/MTU. The analyses presented by the licensees will accommodate a Cycle 2 burnup up to 17,500 MWD/MTU and is applicable for Cycle 1 termination burnups of between 15,744 and 17,280 MWD/MTU.

3.2 Power Distributions

Hot full power (HFP) fuel assembly relative power densities are given in the reload analysis report for BOC, middle-of-cycle (MOC), and EOC unrodded configurations. Radial power distributions at BOC and EOC are also given for three rodded configurations allowed by the power dependent insertion limit (PDIL) at full power. These rodded configurations consist of part length CEAs, Bank 5, and Bank 5 plus the part length CEAs.

These expected values are based on ROCS code calculations with neutron cross sections generated by the DIT code (Ref. 6). Also, the use of ROCS and DIT with the MC fine-mesh module explicitly accounts for the higher power peaking which is characteristic of fuel rods adjacent to water holes. These methods have been approved by the NRC and, therefore, the calculated power distributions are 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% 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 3.5% delta k/k shutdown margin provides adequate protection. Sufficient boration capability and net available CEA worth exist.

assuming a minimum worth stuck CEA and using appropriate calculational uncertainties, to meet these shutdown margin requirements. These results were derived by approved methods and incorporate appropriate assumptions and are, therefore, acceptable.

3.4 Augmentation Factors

CE submitted a report (Ref. 5) which gave the results of a review of interpellet gap formation, ovality, creepdown and clad collapse data in modern PWR fuel rods (non-densifying fuel in prepressurized tubes). The report concluded that since the increased power peaking associated with the small interpellet gaps found in these rods is insignificant compared to other power distribution uncertainties used in the safety analyses, augmentation factors can be removed from the reload of any reactor loaded exclusively with this type of fuel. The staff accepted this conclusion for the Cycle 2 reload review of PVNGS1, the Cycle 8 reload review of Calvert Cliffs Unit 1, the Cycle 3 review of SONGS-2, and Cycle 2 review of Waterford 3. The staff finds that the conclusion is also valid for PVNGS2 Cycle 2 since the same manufacturing process is used in the fuel fabrication. The densification augmentation factors can, therefore, be eliminated for PVNGS2 Cycle 2.

4.0 EVALUATION OF THERMAL-HYDRAULIC DESIGN

Steady-state thermal-hydraulic analysis for Cycle 2 is performed using the approved thermal-hydraulic code TORC (Ref. 7) and the CE-1 critical heat flux (CHF) correlation (Ref. 8). The design thermal margin analysis is performed with the fast running variation of the TORC code, CETOP-D (Ref. 9). 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 modified statistical combination of uncertainties (SCU) methodology described in Reference 10. This SCU methodology was evaluated and approved in the safety evaluation issued with Amendment No. 24 to Facility Operating License No. NPF-41 for PVNGS1, dated October 21, 1987. 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 2 DNBR limit of 1.24, these overall uncertainty penalty factors provide assurance with a 95% confidence and a 95% probability (95/95 confidence/probability) that the hottest fuel rod will not experience DNB. The fuel rod bow penalty is incorporated directly in the DNBR limit. It has been calculated using the approved method described in Reference 11. The value used for this analysis, 1.75% DNBR, is valid for fuel assembly burnups up to 30,000 MWD/MTU. For those assemblies with average burnup in excess of 30,000 MWD/MTU, sufficient margin exists to offset rod bow penalties.

5.0 EVALUATION OF SAFETY ANALYSES

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 licensees to assess the need for reanalysis as a result of the new core configuration for Cycle 2. Those events for which results were not bounded by the FSAR were reanalyzed by the licensees to assure that the applicable criteria are met. The AOOs were analyzed to assure that specified acceptable fuel design limits (SAFDLs) on DNBR and fuel centerline to melt (CTM) temperature are not exceeded. This assurance may require either reactor protection system (RPS) trips or RPS trips and/or sufficient initial steady state margin.

Unless otherwise stated, the plant response to the DBEs was simulated using the same methods and computer programs which were used and approved for the reference cycle analyses. These include the CESEC III, STRIKIN II, TORC and HERMITE computer programs. For some of the reanalyzed DBEs, certain initial core parameters, such as CEA trip worth and axial shape index (ASI), were assumed to be more limiting than the calculated Cycle 2 values in order to bound future cycles. All of the events reanalyzed have results which are within NRC acceptance criteria and, therefore, are acceptable. These are discussed below.

5.1 Steam System Piping Failures Inside and Outside Containment

Steam line breaks (SLBs) inside containment may cause environmental degradation of sensor input to the core protection calculators (CPCs) and pressure measurement systems. Therefore, the only credit taken for CPC action during this event is the CPC variable overpower trip (VOPT). The required input to the VOPT includes output from the resistance temperature detectors (RTDs) and the excore neutron flux detectors. These sensors have been qualified in degraded environmental conditions for a sufficient length of time to allow their use in providing input for VOPT action for this event. The outside containment SLBs, however, are not subject to the same environmental effects on the RPS as the inside containment breaks and the full array of RPS trips, including the CPC low DNBR trip, can be credited. By crediting these RPS trips, the results of both the inside and outside containment SLB events in terms of fuel pin failure caused by the pre-trip power excursion are bounded by the reference cycle.

The hot zero power SLB post-trip return to power was also reanalyzed because of the more adverse moderator cooldown reactivity insertion curve. The effect of this more adverse reactivity insertion was accommodated for in Cycle 2 by increasing the shutdown margin required by the Technical Specifications at zero power from 6.0% delta k/k to 6.5% delta k/k. With this more restrictive requirement, the results of the SLB event initiated from zero power conditions is bounded by the reference cycle analysis. The results of the SLB event initiated from full power conditions are also bounded by the corresponding reference cycle analysis. Therefore, the staff concludes that Cycle 2 operation is acceptable with respect to accidents resulting in breaks in the steam line.

5.2 CEA Drop Event

The single full length CEA drop event was reanalyzed to determine the initial thermal margin that must be maintained by the LCOs such that the SAFDLs will not be violated. Since the CEA position related penalty factors for downward single CEA deviations of the 4-fingered CEAs have been set equal to unity (no penalty) as part of the CPC improvement program, a reactor trip is not generated for a single 4-fingered CEA drop and, therefore, the expected margin degradation for the event is accounted for by reserving sufficient margin in the LCOs. Although this applies to part length CEAs also, only the single full length CEA drop is analyzed because it requires the maximum initial margin to be maintained by the LCOs. For 12-fingered CEA drops and CEA subgroup drops, the CEA position related penalty factors for downward deviations are still used by the CPCs, as in Cycle 1, to provide a reactor trip when necessary.

The event was initiated by dropping a full length CEA over a period of one second. The turbine load was not reduced, resulting in a power mismatch between the primary and secondary systems, which leads to a cooldown of the RCS. The largest change in power peaking was obtained by evaluating drops involving different individual CEAs into the radial rodded configurations allowed by the power dependent transient insertion limits in Figures 3.1-3 and 3.1-4 of the Technical Specifications. This resulted in a radial peaking factor increase of 8.5% before the effects of short term xenon redistribution set in.

A minimum DNBR of greater than 1.24 was obtained after 900 seconds, as determined from the 8.5% radial power peaking increase following the CEA drop plus 15 minutes of xenon redistribution at the final coolant conditions, resulting in a maximum peaking factor increase of 11.4%. If the dropped CEA has not been realigned within 10 minutes after the drop, the operator will take action to reduce power in accordance with Figure 3.1-2A of the Technical Specifications. A maximum allowable initial linear heat rate (LHR) of 18.0 kw/ft could exist as an initial condition without causing the acceptable fuel centerline melt limit of 21.0 kw/ft to be exceeded during the transient. This amount of margin is assured since the LHR LCO is based on the more limiting allowable LHR for the loss of coolant accident (LOCA) of 13.5 kw/ft. The staff, therefore, concludes that Cycle 2 meets the requirements of SRP Section 15.4.3 governing control rod misoperation.

5.3 Asymmetric Steam Generator Events

Of the four events which affect a single steam generator, the loss of load to one steam generator (LL/1SG) event is the most limiting. This event is initiated by the inadvertent closure of both main steam isolation valves which results in a loss of load to the affected steam generator. The CPC high differential cold leg temperature trip is the primary means of mitigating this transient with the steam generator low level trip providing an additional means of protection. The calculated minimum transient DNBR was greater than the DNBR SAFDL limit of 1.24. A maximum allowable LHR of 17.0 kw/ft could exist as an initial condition without exceeding the fuel CTM SAFDL of 21.0 kw/ft during the transient. This amount of margin is assured by setting the LHR LCO based on the more limiting allowable LHR for LOCA of 13.5 kw/ft. The staff concludes that the calculations contain sufficient conservatism to assure that fuel damage will not result from any asymmetric steam generator event during Cycle 2 operation.

A methodology change was made from the reference cycle analysis of this event. The change involved the application of the HERMITE computer code to model both the effect of the temperature tilt on radial power distribution and the space-time impact of the CEA scram. HERMITE has been approved for licensing applications (Ref. 12) and uses the core parameters generated by the CESEC III code (core flow, RCS inlet temperature, RCS pressure, and reactor trip time) as input to simulate the core in the two dimensions. The staff finds this improved modeling technique acceptable.

5.4 Loss of Coolant Accident (LOCA)

The emergency core cooling system (ECCS) performance evaluation for both the large and the small break LOCA must show conformance with the acceptance criteria required by 10 CFR 50.46. Since there are no significant changes to the RCS design characteristics compared to the reference cycle, the blowdown hydraulic calculations, refill/reflood hydraulics calculations, and steam cooling heat transfer coefficients of the reference cycle also apply to Cycle 2. Therefore, only fuel rod clad temperature and oxidation calculations were performed for the 1.0 double ended guillotine at pump discharge (DEG/PD) break to evaluate ECCS performance due to the Cycle 2 changes in fuel conditions. This was the limiting break size for the reference cycle and, since the hydraulics are identical, is also the limiting break size for Cycle 2.

The 1.0 DEG/PD limiting large break case resulted in a peak clad temperature (PCT) of 1960°F, a peak local clad oxidation percentage of 5.7%, and a total core wide clad oxidation percentage of less than 0.8%. These results meet the 10 CFR 50.46 acceptance criteria for peak clad temperature (2200°F), peak local clad oxidation percentage (17.0%), and core wide clad oxidation percentage (1.0%). These results are applicable for up to 400 plugged tubes per steam generator because of the conservatively high pressure drop through the steam generators used in the analyses.

The increase in PCT for a small break LOCA, assuming 400 plugged tubes per steam generator, is much less than 100°F. Therefore, the estimated PCT for Cycle 2 is less than 1730°F and well within the 10 CFR 50.46 limit.

Based on these results, the staff concurs that both large and small break acceptable LOCA ECCS performance has been demonstrated for Cycle 2 at a peak linear heat generation rate of 13.5 kw/ft and a reactor power level of 3876 Mwt (102% of 3800 Mwt) for up to 400 plugged tubes per steam generator.

6.0 EVALUATION OF TECHNICAL SPECIFICATION CHANGES

In support of Cycle 2 operation, the licensees have requested a number of changes to the Technical Specifications. The specific changes and the staff's evaluation are presented below.

(1) The shutdown margin versus cold leg temperature curve given in Figure 3.1-1A of Technical Specification 3.1.1.2 has been changed to increase the required shutdown margin value from 6.0% delta k/k to 6.5% delta k/k at cold leg temperatures above 500°F.

The increased shutdown margin is required to ensure that the steam line break event at hot zero power, which is the most limiting accident with regard to shutdown margin requirements for Cycle 2, is bounded by the reference cycle (Cycle 1) analysis. Sufficient CEA trip reactivity worth is available to meet the shutdown margin requirements even with the most reactive CEA assumed stuck in the fully withdrawn position. The staff, therefore, finds this change acceptable.

(2) The moderator temperature coefficient (MTC) operating band given in Figure 3.3-1 of Technical Specification 3.1.1.3 has been made more positive at low power and more negative at high power. In addition, the x axis has been changed to core power level instead of average moderator temperature.

The MTC for Cycle 2 at 100% power has a value of 0.0 at BOC which is the same value that the Cycle 1 MTC had at 100% power and BOC. The BOC zero power value has been increased from $+0.22\times10^{-4}$ to $+0.5\times10^{-4}$ delta k/k/°F and the EOC full power value has been decreased from -3.0×10^{-4} to -3.5×10^{-4} delta k/k/°F. The licensees have reevaluated the most limiting transients and accidents which can be adversely affected by the increased MTC operating band and found them to be bounded by the referance cycle (Cycle 1) analyses. In addition, by making the MTC a dependent variable of core power only and not of inlet temperature and core power, the calculation of the limiting MTC need only be performed once. There is no effect on the safety analysis results and the same approved methodology and computer codes are used in the calculations. Therefore, the proposed change is acceptable.

(3) The operational pressure band of the pressurizer given in Technical Specification 3.2.8 has been changed from 1815 psia through 2370 psia to 2025 psia through 2300 psia.

The potential transients initiated at the extremes of the Cycle 1 pressure range were not analyzed for Cycle 2 and, therefore, normal operation at these extremes cannot be supported by the safety analyses. Therefore, the operation band of the pressurizer was made more restrictive, limiting the field of possible accidents and maintaining the safety margin required by the reference cycle safety analysis or the FSAR safety limits. Therefore, this change is acceptable.

(4) Reference to the part length CEA insertion limits have been removed from Technical Specification 3.1.3.1 and a new Specification 3.1.3.7 has been added to specify the length of time for insertion and the insertion limit of the part length CEAs specifically.

The new Specification adds a more explicit LCO to clarify the allowable duration for a part length CEA to remain within the defined ranges of axial position and reduces the potential adverse consequences of a part length CEA drop or misalignment from an allowable position. The changes are, therefore, acceptable.

(5) The response time of the DNBR-low reactor coolant pump (RCP) shaft speed trip in Technical Specification 3.3.1, Table 3.3-2, has been decreased from 0.75 seconds to 0.30 seconds.

The response time has been defined as the time from when a signal is sent down the RCP shaft speed sensor line to the CPCs to the time when the control element drive mechanism coil breakers open. Since the Cycle 2 safety analysis has taken credit for the faster response time, the change to Table 3.3-2 is necessary to ensure that PVNGS2 is operated within the safety analysis. Therefore, it is acceptable.

(6) The DNBR limit specified in Technical Specification 2.1.1.1, Table 2.2-1 and Bases Sections 2.1.1 and 2.2.1, has been changed from 1.231 to 1.24. Also, the requirement to calculate additional rod bow penalties has been removed from Notation (5) of Table 2.2-1 and the low pressurizer pressure floor has been changed from 1861 psia to 1860 psia.

The modified SCU methodology discussed in Section 4.0 of this evaluation yields a DNBR limit of 1.24. The overall uncertainty factors determined by this modified methodology, which has been approved by the staff, continue to ensure that the COLSS power operating limit calculations and the CPC DNBR and LPD calculations will be conservative to at least a 95% probability and 95% confidence level. The 1.24 DNBR limit is, therefore, acceptable.

The rod bow penalty factor of 1.75% which has been applied to the DNBR limit compensates for the effects of rod bow for fuel assemblies with burnups up to 30,000 MWD/MTU. As discussed in Section 4.0 of this evaluation, sufficient available margin exists in assemblies with burnup greater than 30,000 MWD/MTU to offset any additional rod bow penalties. The deletion of these additional penalties from Table 2.2-1 is, therefore, acceptable.

A reevaluation of the Cycle 2 safety analysis was performed to determine how the low pressurizer pressure floor for the DNBR-low trip would change as a result of the DNBR limit change. Since the results show that a pressurizer pressure of 1860 psia instead of 1861 psia will ensure acceptable consequences in the event of a reactor trip on low-DNBR, the proposed change to the low pressurizer pressure floor is acceptable.

(7) The CEA insertion limits given in Technical Specification 3.1.3.6 have been revised to account for changes in the reactivity worth of the CEAs due to changes in the Cycle 2 core.

Since the reactivity worth of the CEAs has changed, the consequences of the dropped and ejected CEA events are affected. The revised CEA insertion limits chosen, which were calculated by approved methods, ensure that there is sufficient margin to mitigate such events during Cycle 2. The CEA insertion limit revision is, therefore, acceptable.

(8) The CPC penalty factors, which have been used to compensate for resistance temperature detector (RTD) response times greater than 8 seconds (but less than or equal to 13 seconds) have been removed from the Technical Specifications by modifying Table 3.3-2 and removing Table 3.3-2a.

The Cycle 2 safety analyses assume a maximum RTD response time of 8 seconds and do not include an allowance to enter CPC penalty factors to compensate for RTD response times greater than 8 seconds. Hence, the removal of the penalty factor allowances is required in order to ensure that the Cycle 2 safety analyses assumptions are met during Cycle 2 operation. Therefore, the change is acceptable.

(9) The RCS total flow rate specified in Technical Specification 3.2.5 has been reduced from greater than or equal to 164.0×10^6 lbm/hr to greater than or equal to 155.8×10^6 lbm/hr.

This change ensures that the actual total RCS flow rate is maintained at or above the minimum value used in the Cycle 2 safety analysis and is, therefore, acceptable.

(10) The LHR limit defined in Technical Specification 3.2.1 has been decreased from 14.0 kw/ft to 13.5 kw/ft. In addition, the amendment also delineates how LHR is to be monitored and changes the format of the Action statement.

As stated in Section 3.4, augmentation factors previously used to compensate for increased power peaking due to fuel densification were not used for the Cycle 2 safety analyses. The elimination of these augmentation factors as well as the increased fuel enrichment and different core loading pattern for Cycle 2, result in a change in the allowable LHR limit. Since the Cycle 2 safety analyses show that in the event of a LOCA, the peak fuel clad temperature will not exceed 2200°F, the decreased LHR is acceptable.

The change which delineates how LHR is to be monitored is also acceptable since, by providing more detail on the monitoring of LHR, there is added assurance that the LHR will be maintained below the specified limit.

The change in the format of the Action statement facilitates assessment of the actions required if the LHR limit should be exceeded and is, therefore, acceptable.

(11) Technical Specification 3.2.4 has been replaced with a new format which (a) addresses the specific conditions for monitoring DNBR with or without COLSS and/or the CEACs, (b) delinates what Actions should be taken, (c) removes reference to the DNBR penalty factor table used in Technical Specification 4.2.4.4, and (d) replaces Figures 3.2-1 and 3.2-2 with new Figures 3.2-1, 3.2-2, and 3.2-2A. In addition, as a result of being incorporated into the new Technical Specification 3.2.4, references to operation with both CEACs inoperable and the graph of DNBR margin operating limit (Figure 3.3-1) have been removed from Technical Specification 3.3.1. Bases Sections 3/4.1.3 and 3/4.2.4 have also been modified due to Cycle 2 differences and the above mentioned changes.

These changes ensure operation of the reactor within the approved Cycle 2 safety analysis by modifying the figures, increase operator reliability by placing DNBR operating limits in one place, and eliminate superfluous information. The changes are, therefore, acceptable.

(12) The refueling actuation signal trip value of the refueling water storage tank, given in Table 3.3-4 of Technical Specification 3.3.2, has been changed from ≥ 7.4% to 7.4% (the midpoint) of the allowable operational values.

The change is more restrictive since it maintains the trip value at the midpoint of the allowable band and reduces the allowable trip values to a single value which was a part of the safety analysis. Therefore, this is acceptable.

(13) A number of administrative changes have been made to Bases Sections 2.2.1, 3/4.3.1, and 3/4.3.2.

These changes were made to ensure clarity and conciseness to include updated references and to remove Cycle 1 specific information no longer needed for Cycle 2. The changes are, therefore, acceptable.

7.0 STARTUP TESTING

The licensees have presented a brief description of the low power physics tests and the power ascension testing to be performed during Cycle 2 startup. The described tests will verify that core performance is consistent with the engineering design and safety analyses. If the acceptance criteria of any of the startup physics tests are not met, an evaluation will be performed by the licensees, presented to the Plant Review Board and resolved by the Board 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 2 and finds that it conforms to accepted practices and adequately supplements normal surveillance tests which are required by the plant Technical Specifications.

8.0 EVALUATION FINDINGS

The staff has reviewed the fuels, physics, and thermal-hydraulics information presented in the PVNGS2 Cycle 2 reload report. The staff has also reviewed the proposed 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 and the Technical Specification changes to be acceptable.

9.0 CONTACT WITH STATE OFFICIAL

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

10.0 ENVIRONMENTAL CONSIDERATIONS

This amendment involves a change in the installation or use of facility components located within the restricted area as defined in 10 CFR 20 relating to a reactor refueling. The staff has determined that this amendment involve no significant increase in the amount, and no significant change in the type, of any effluent that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued proposed findings that the amendment involves no significant hazard 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 needs to be prepared in connection with the issuance of this amendment.

11.0 CONCLUSION

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

Principal Contributor: L. Kopp

Dated: May 5, 1988

REFERENCES

1. Reload Technical Specification Amendment, submitted by letter from E. E. Van Brunt, Jr. (ANPP), dated December 2, 1987.

2. "Reload Analysis Report for Palo Verde Nuclear Generating Station Unit 2 Cycle 2," submitted by letter from E. E. Van Brunt, Jr. (ANPP), dated December 3, 1987.

3. "Palo Verde Nuclear Generating Station (PVNGS) Unit 2 Cycle 2 Reload Questions," submitted by letter from E. E. Van Brunt, Jr. (ANPP), dated February 4, 1988.

4. Amendment No. 18 to Facility Operating License No. NPF-51, dated March 9, 1988.

5. "CEPAN Method of Analyzing Creep Collapse of Oval Cladding, Volume 5: Evaluation of Interpellet Gap Formation and Clad Collapse in Modern PWR Fuel Rods," EPRI NP-3966-CCM, April 1985.

6. "The ROCS and DIT Computer Codes for Nuclear Design," CENPD-266-P-A, Combustion Engineering, April 1983.

7. "TORC Code, A Computer Code for Determining the Thermal Margin of a Reactor Core," CENPD-161-P, Combustion Engineering, July 1975.

8. "Critical Heat Flux Correlation for C-E Fuel Assemblies with Standard Spacer Grids, Part 1, Uniform Axial Power Distribution," CENPD-162-P-A, Combustion Engineering, April 1975.

9. "CETOP-D Code Structure and Modeling Methods for San Onofre Nuclear Generating Station, Units 2 and 3," CEN-160(S)-P, Revision 1-P, Combustion Engineering, September 1981.

10. "Modified Statistical Combination of Uncertainties," CEN-356(V)-P, Revision 01-P, Combustion Engineering, July 1987.

11. "Fuel and Poison Rod Bowing," CENPD-225-P-A, Combustion Engineering, June 1983.

12. "HERMITE Space-Time Kinetics," CENPD-188-A, Combustion Engineering, July 1976.

13. "Safety Evaluation of CEN-161 (FATES3)," submitted by letter from R. A. Clark (NRC), to A. E. Lundvall, Jr. (BG&E), March 31, 1983.