UNITED STATES
NUCLEAR REGULATORY COMMISSION
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 24 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 June 29, 1987 (Ref. 1), as supplemented by letters dated August 20, 1987 (Ref. 4) and October 1, 1987, 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 NPF-41) for the Palo Verde Nuclear Generating Station, Unit 1 (PVNGS1), relating to Cycle 2 operation for PVNGS1. In support of both the Technical Specification changes and Cycle 2 operation, the licensees submitted (1) a Reload Analysis Report by letter dated June 29, 1987 (Ref. 2), as supplemented by letters dated July 13 and August 20, 1987 (Ref. 5 and 6) and September 4, 1987, and (2) a report concerning a modified version for a Statistical Combination of Uncertainties (SCU), dated July 1987 (Ref. 3). The staff's evaluation of the SCU report and the reload analysis is presented in Sections 2.0 through 6.0 below. The evaluation of the specific changes to the Technical Specifications is presented in Section 7.0 below.

2.0 EVALUATION OF FUEL DESIGN

2.1 Mechanical Design

The Cycle 2 core consists of 241 fuel assemblies. Eighty fresh (unirradiated) Batch D assemblies will replace 69 Batch A assemblies and 11 Batch B assemblies. The remaining 97 Batch B assemblies and all Batch C assemblies in the core during Cycle 1 will be retained.

The 80 Batch D assemblies will consist of 36 type DO assemblies with 4.05 weight percent (w/o) and 3.36 w/o U-235 enriched fuel rods, 28 type D* assemblies with 3.36 w/o and 2.78 w/o U-235 enriched rods and eight burnable poison shims per assembly, 12 type DX assemblies with 3.36 w/o and 2.78 w/o U-235 enriched rods and eight burnable poison shims per assembly, and four type D/ assemblies with 3.36 w/o and 2.78 w/o U-235 enriched rods and eight 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 the fabricability and quality of the fuel and the burnup capability of the poison rods. The staff, therefore, finds these modifications to be acceptable.

The licensees have also evaluated the criticality effects of storage of the higher enriched Batch D fuel assemblies in the PVNGS1 fuel storage facilities and have shown that the NRC acceptance criterion of k_{eff} no greater than 0.95 is met for all normal and abnormal conditions. This evaluation is discussed in more detail in Section 3.1 below. The staff, therefore, concludes that the Batch D fuel assemblies are acceptable for use during Cycle 2.

Attachment 5 to Reference 7 is a report entitled, "Evaluation of Inter-pellet Gap Formation and Clad Collapse in Modern PWR Fuel Rods," regarding 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 PVNGS1 Cycle 2. This concurrence is supported by similar results of analyses by another fuel vendor and was previously accepted for Waterford 3 Cycle 2. Therefore, the staff concludes that no further analysis of clad collapse need be performed for PVNGS1 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. 12) 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 PVNGS1 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 4.05 weight percent U-235.

The storage facility for new fuel at PVNGS1 provides dry storage for a maximum of 90 fuel assemblies (more than one-third of a core load) and includes the fuel assembly storage racks and the concrete storage vault that contains the storage racks. The fuel assemblies are normally stored in a dry environment in these racks.

The spent fuel storage facility provides underwater storage for a maximum of 1329 fuel assemblies or approximately 5-1/3 full core loads. The facility is designed to store spent fuel assemblies in three different arrays depending on the mode chosen. The storage racks are stainless steel honeycomb structures with rectangular fuel storage cells. By properly placing L-inserts and blocking devices in the fuel storage cells, the fuel may be stored in a checkerboard mode for a maximum of 665 fuel assemblies. The blocking devices are placed in appropriate cells in order to prevent the inadvertent placement of a fuel assembly in other than the prescribed spacing. By inserting neutron poison boxes in the rack cells, the fuel may be stored in a high density mode for a maximum of 1329 fuel assemblies. These two storage modes may also be utilized simultaneously within an individual storage rack resulting in a third mixed mode of storage. The spent fuel assemblies are normally stored in borated water.

By letter dated July 13, 1987 (Ref. 5), the licensees submitted a letter from C. Ferguson (CE) to Paul F. Crawley (ANPP), V-CE-34683, dated May 27, 1987 documenting the review of analyses performed to support PVNGS Units 1, 2 and 3 fuel enrichments up to 4.30 weight percent U-235. This letter verifies that the original analyses of the new fuel, spent fuel (except as noted below) and intermediate racks, as well as the fuel elevator, fuel upender and transfer machine were all performed for 4.30 weight percent U-235 fuel. Since the k_{eff} values based on the storage of 4.30 weight percent U-235 fuel meet the NRC acceptance criteria of no greater than 0.95 for fully flooded (unborated) conditions and 0.98 for optimum moderation conditions, the PVNGS fuel storage facilities are acceptable for the storage of 4.05 weight percent U-235 fuel.

One exception is the analysis of the spent fuel racks with neutron poison (boron) boxes in the cells. This high density mode was analyzed for a maximum enrichment of 4.0 weight percent U-235. Therefore, if a future decision is made to use this mode for storage of higher enriched fuel, an analysis will be required for this higher enrichment.

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

17,280 MWD/MTU, the Cycle 2 reactivity lifetime for full power operation is expected to be 13,056 MWD/MTU. The analyses presented by the licensees will accommodate a Cycle 2 burnup up to 13,098 MWD/MTU and are applicable for Cycle 1 termination burnups of between 16,512 and 19,085 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. 8). 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. 7) 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 8 reload review of Calvert Cliffs Unit 1, the Cycle 3 review of SONGS-2, and the Cycle 2 review of Waterford 3. The staff finds that the conclusion is also valid for PVNGS1 Cycle 2 since the same manufacturing process is used in the fuel fabrication. The densification augmentation factors can, therefore, be eliminated for PVNGS1 Cycle 2.

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4.0 EVALUATION OF THERMAL-HYDRAULIC DESIGN

Steady-state thermal-hydraulic analysis for Cycle 2 is performed by using the approved thermal-hydraulic code TORC (Ref. 9) and the CE-1 critical heat flux (CHF) correlation (Ref. 10). The design thermal margin analysis is performed with the fast running variation of the TORC code, CETOP-D (Ref. 11). The CETOP-D model has been verified to predict the 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 3, which is evaluated and approved below in Section 5.0 of this evaluation. 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 13. 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 MODIFIED STATISTICAL COMBINATION OF UNCERTAINTIES (SCU)

The licensees requested NRC review and approval of the topical report, "Modified Statistical Combination of Uncertainties," CEN-356(V)-P, Revision 01-P, July 1987 (Ref. 3). This report describes changes to the methodology for statistically combining uncertainties to obtain overall uncertainty factors. The overall uncertainty factors are used to determine the limiting safety system setting (LSSS) and limiting condition for operation (LCO) for the PVNGS COLSS and CPC system.

The existing SCU method treats uncertainties in two groups. The uncertainties in one group (system parameter uncertainties) include engineering factors, critical heat flux (CHF) correlation uncertainties and code modeling uncertainties which are statistically combined to generate a DNBR probability density function. The 95/95 probability/confidence level limit of this function is then used as the setpoint analysis minimum DNBR. The uncertainties in the other group (state parameter uncertainties) include measured state parameter, COLSS and CPC algorithm, radial peaking factor measurement, simulator model, computer processing and startup measurement uncertainties. These uncertainties are also statistically combined to determine the CPC and COLSS overall uncertainty factors.

Although the uncertainties within each group are combined statistically and a 95/95 probability/confidence level generated for each group, the resultant uncertainties of the two groups are effectively combined in a deterministic manner due to the separate application of the two uncertainty limits. The proposed modified SCU methodology would statistically combine uncertainty components which were previously applied deterministically. In addition, the statistical treatment of several uncertainty components would be modified so that the overall uncertainty factors can be calculated and applied as a function of burnup, axial shape index (ASI), and power in COLSS and CPC.

The staff has reviewed the uncertainties and the uncertainty treatment procedure described for the proposed modified SCU methodology and has determined that the resultant penalties applied to the COLSS power operating limit and the CPCS DNBR and local power density (LPD) calculations adequately incorporate all uncertainties at the 95/95 probability/confidence level. The analytical methods reviewed show that a DNBR limit of 1.24 with the uncertainty penalties derived in the report provides a 95/95 probability/confidence level assurance against DNB occurring during steady state operation or anticipated operational occurrences at the Palo Verde Nuclear Generating Station. The proposed methodology is, therefore, acceptable for use with the Palo Verde Nuclear Generating Station digital monitoring and protection systems.

6.0 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 APS 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 APS 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.

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6.1 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 excurson 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 is 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.

6.2 Loss of Forced Reactor Coolant Flow

The loss of coolant flow (LOF) event was reanalyzed by APS due to the change in the CPC trip. Rather than using the low DNBR trip, the LOF event for Cycle 2 was analyzed with a CPC trip based on low reactor coolant pump (RCP) shaft speed, which is initiated when the shaft speed drops to 95% of its initial speed. The analysis also used a more rapid coastdown than the reference analysis. The results show that this event initiated from the Technical Specification LCOs in conjunction with the low RCP shaft speed trip will not exceed the minimum DNBR limit of 1.24. As in the reference analysis, no credit is taken for the slight RCS pressure increase in computing this minimum DNBR. The acceptance criteria stated in SRP Section 15.3.1, therefore, are met and the staff concludes that the results of the LOF event for Cycle 2 are acceptable.

6.3 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 PDIL 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. Since there is no trip assumed, the peak will stabilize at this asymptotic value after a few minutes as the secondary side continues to demand 100% power.

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 by then, the operator will take action to reduce power in accordance with Figure 3.1-1A of the Technical Specifications. A maximum allowable initial 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.

6.4 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. 14) 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 to be acceptable.

6.5 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 1925°F, a peak local clad oxidation percentage of 4.6%, 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 for both large and small break LOCAs, acceptable 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.

7.0 EVALUATION OF TECHNICAL SPECIFICATION CHANGES

In support of Cycle 2 operation, the licensees have requested a number of changes to the Technical Specifications (Ref. 1 and 4). The specific changes and the staff's evaluation are presented below.

(1) The maximum allowed enrichment of reload fuel specified in Technical Specification 5.3.1 has been increased from 4.0 to 4.05 weight percent U-235.

The slight increase in allowed reload fuel enrichment has been properly accounted for in the Cycle 2 reload analysis and results in acceptable consequences. The effect of the higher enrichment on the storage of fuel assemblies has been evaluated in Section 3.1 above and found to be acceptable.

(2) 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.

(3) 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 and 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 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. APS has reevaluated the most limiting transients and accidents which can be adversely affected by a positive MTC and has found them to be bounded by the reference cycle (Cycle 1) analysis. 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.

(4) 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 operational band of the pressurizer was made more restrictive, thus 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.

(5) 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.

(6) 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 is 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 PVNGS1 is operated within the safety analysis. Therefore, it is acceptable.

(7) 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 changed from 1861 psia to 1860 psia.

The modified SCU methodology discussed above in Section 5.0 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.

(8) 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 the changes in 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.

(9) 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.

(10) 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.

(11) 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 revised Specification also delineates how LHR is to be monitored and changes the format of the Action statement.

As stated above in Section 3.4 of this evaluation, 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.

(12) 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 CEACs, (b) delineates 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 analyses by modifying the figures, increase operator reliability by placing DNBR operating limits in one place, and eliminate superfluous information. The changes are, therefore, acceptable.

(13) 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 $\geq$ 7.4% to 7.4% 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 change is acceptable.

(14) 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.

8.0 EVALUATION FINDINGS

The staff has reviewed the fuels, physics, and thermal-hydraulics information presented in the PVNGS1 Cycle 2 reload report. The staff has also reviewed the proposed Technical Specification revisions, the SCU modification, 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 has been advised of the proposed determination of no significant hazards consideration with regard to these changes. No comments were received.

10.0 ENVIRONMENTAL CONSIDERATION

This amendment involves changes in the installation or use of facility components located within the restricted area. The 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 proposed findings that the amendment involves no significant hazards consideration, and there has been no public comment on such findings. 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 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: October 21, 1987

REFERENCES

1. Reload Technical Specification Amendment, submitted by letter from J. G. Haynes (ANPP) dated June 29, 1987.

2. Reload Analysis Report for Palo Verde Nuclear Generating Station Unit 1 Cycle 2, submitted by letter from J. G. Haynes (ANPP) dated June 29, 1987.

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

4. Revision to Reload Technical Specification Amendment - Attachment 2, Shutdown Margin, submitted by letter from J. G. Haynes (ANPP) dated August 20, 1987.

5. Letter from J. G. Haynes (ANPP) dated July 13, 1987.

6. Response to NRC Questions Regarding the Unit 1 Cycle 2 Reload Submittal, submitted by letter from J. G. Haynes (ANPP) dated August 20, 1987.

7. "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.

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

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

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

11. "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.

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

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

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

INSTRUMENTATION

INCORE DETECTORS

LIMITING CONDITION FOR OPERATION

3.3.3.2 The incore detection system shall be OPERABLE with:

a. At least 75% of all incore detector locations, and 75% of all detectors, with at least one detector in each quadrant at each level; and

b. A minimum of six tilt estimates, with at least one at each of three levels.

An OPERABLE incore detector location shall consist of a fuel assembly containing a fixed detector string with a minimum of three OPERABLE rhodium detectors or an OPERABLE movable incore detector capable of mapping the location.

APPLICABILITY: When the incore detection system is used for monitoring:

a. AZIMUTHAL POWER TILT,

b. Radial Peaking Factors,

c. Local Power Density,

d. DNB Margin.

ACTION:

a. With the incore detection system inoperable, do not use the system for the above applicable monitoring or calibration functions.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.2 The incore detection system shall be demonstrated OPERABLE:

a. By performance of a CHANNEL CHECK within 24 hours prior to its use if the system has just been returned to OPERABLE status or if 7 days or more have elasped since last use and at least once per 7 days thereafter when required for monitoring the AZIMUTHAL POWER TILT, radial peaking factors, local power density or DNB margin:

b. At least once per 18 months by performance of a CHANNEL CALIBRATION operation which exempts the neutron detectors but includes all electronic components. The fixed incore neutron detectors shall be calibrated prior to installation in the reactor core.

INSTRUMENTATION

SEISMIC INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.3 The seismic monitoring instrumentation shown in Table 3.3-7 shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

a. With one or more seismic monitoring instruments inoperable for more than 30 days, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of the malfunction and the plans for restoring the instrument(s) to OPERABLE status.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.3.1 Each of the above seismic monitoring instruments shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations at the frequencies shown in Table 4.3-4.

4.3.3.3.2 Each of the above seismic monitoring instruments actuated during a seismic event (greater than or equal to 0.02g) shall have a CHANNEL CALIBRATION performed within 5 days. Data shall be retrieved from actuated instruments and analyzed to determine the magnitude of the vibratory ground motion. A Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 10 days describing the magnitude, frequency spectrum, and resultant effect upon facility features important to safety.