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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATING TO STEAM LINE BREAK REANALYSES

ARIZONA PUBLIC SERVICE COMPANY

PALO VERDE NUCLEAR GENERATING STATION, UNITS 1, 2, AND 3

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

INTRODUCTION

In accordance with PVNGS Unit 1 License Condition 2.C.(21) to Facility Operating License NPF-34 and Supplement No. 8 to the staff's Safety Evaluation Report (NUREG-0857) dated May 1985, Arizona Nuclear Power Project (ANPP's) submitted the results of its Main Steam Line Break (SLB) - Chapter 15 reanalyses by letter dated September 30, 1985.

THERMAL HYDRAULICS EVALUATION

To meet the conditions stated in Supplement 8 to the Safety Evaluation Report (NUREG-0857) for the Palo Verde Final Safety Analyses Report (FSAR), Arizona Nuclear Power Project (the licensee) submitted the steam line break (SLB) reanalysis for review and approval. Six SLB cases were reanalyzed:

(1) SLB at full power and concurrent loss of offsite power.
(2) Full power SLB.
(3) Zero power SLB with concurrent loss of offsite power.
(4) Zero power SLB.
(5) Full power SLB outside containment.
(6) Zero power SLB outside containment with concurrent loss of offsite power.

The plant specific conditions (i.e., those differing from the CESSAR standard plant conditions) used in the assumptions for SLB reanalysis include:

(1) Auxiliary feedwater (AFW) switched to the intact steam generator (SG) on 325 psid between two SGs.

(2) Feedwater isolation valve closure time of 9.6 seconds (instead of 4.6 seconds used in the Combustion Engineering Standard Safety Analysis Report (CESSAR).

(3) Failure of HPSI to inject for all cases.

(4) AFW flow rate of 750 gpm (instead of 875 gpm used in the CESSAR).

(5) AFW isolation to the affected SG in 16 seconds after establishment of 325 psid between two SGs.

The approved computer code, CESEC-III, was used for the SLB reanalysis with modifications in modeling of the AFW system, boron injection, and voiding in reactor vessel upper head.

The staff, with assistance from its consultant, International Technical Services Corporation (ITS) has reviewed the licensee's reanalysis and concludes that they are acceptable. The staff has reviewed the technical evaluation report prepared by ITS (Appendix A of this safety evaluation) and concurs with their conclusions. The SLB reanalysis for Palo Verde is acceptable since the licensee used acceptable methods to demonstrate that all applicable accident analysis acceptance criteria will not be violated.

DOSE EVALUATION

The staff has reviewed the licensee's revised analysis of the design basis accident offsite dose consequences against the staff's acceptance criteria in the Standard Review Plan and finds it to be acceptable.

In the staff's CESSAR SER (NUREG-0852, Supplement 2, September 1983, page 15-16), we stated that for a number of design basis accidents, including the main steam line break, the licensee's analyses would be acceptable under specific conditions for plants, such as Palo Verde, which reference the CESSAR FSAR. The pertinent conditions are: Technical Specifications on dose equivalent iodine-131 concentrations in primary and secondary coolant of 1.0 and 0.1 microcuries per gram, respectively; and site atmospheric dispersion factors equal to or less than 2.5E-3 and 1.0E-4 sec/m**3 at the exclusion area and low population zone boundaries, respectively.

The Palo Verde Technical Specifications for dose equivalent iodine-131 in primary and secondary coolant are equal to the above guideline values (FSAR Chapter 15). In the Palo Verde SER (NUREG-0857, November 1981) the staff concluded that the respective atmospheric dispersion factors at Palo Verde are only 3.1E-4 and 5.1E-5 sec/m**3, i.e., within the acceptable CESSAR SER values noted above.

Thus, the staff's conditions are satisfied and the licensee's Technical Specifications for dose equivalent iodine-131 in primary and secondary coolant at Palo Verde are acceptable.

CONCLUSION

Based on the above, the staff concludes that ANPP's Main Steam Line Break Reanalyses are acceptable.

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APPENDIX A

TECHNICAL EVALUATION

OF REVISED CHAPTER 15 STEAM LINE BREAK METHODOLOGY AND REANALYSES

FOR PALO VERDE NUCLEAR GENERATING STATION UNITS 1, 2, and 3

1.0 INTRODUCTION

Supplement 1 to the safety evaluation report (SER) for a Final Design Approval based on the Combustion-Engineering Standard Safety Analysis Report (CESSAR), dated March 30, 1983, identified certain issues with respect to which either further information was required of Combustion-Engineering or additional staff effort was necessary to complete the review. These issues were addressed in subsequently issued Supplement 2 to the SER. One of these required confirmatory information related to the analyses of steam piping failures inside and outside of containment (SRP Section 15.3) which had been performed with CESEC-III, a thermal-hydraulic computer program developed by Combustion-Engineering for FSAR Chapter 15 transient and accident analysis. CESEC-III has been reviewed and an SER issued by the NRC.

Because the CESSAR is a standard nuclear steam supply system, the CESSAR FSAR is limited in its plant-specific applicability to the extent that it describes and analyzes the CESSAR System 80 design in a generic sense. Wherever deviations exist from the standard design or where, because of these deviations, the analytical methodologies are modified, a plant-specific review is necessary. This is the case with CESEC-III methodology submitted by Arizona Nuclear Power Project (ANPP) for its Palo Verde Nuclear Generating Station (PVNGS) Units 1, 2, and 3.

Reanalysis in the case of PVNGS was necessitated by the fact that the initial test program performed on the various engineered safety feature (ESF) systems at PVNGS Unit 1 indicated that they did not perform as assumed in the

original licensing safety analyses. The systems that did not perform in accordance with the FSAR analysis assumptions are: (1) HPSI (high-pressure safety injection system (lower flow than assumed), (2) LPSI (low-pressure safety injection) system (lower flow than assumed), and (3) AFW (auxiliary feedwater) system (lower flow and later actuation time than assumed). Because of these degraded conditions in the ESF performance, it was necessary to modify the steam line break (SLB) analysis methodology as described in Appendix 15C of the PVNGS FSAR Chapter 15 to better reflect the actual PVNGS plant-specific conditions.

The licensee took exception to the following CESSAR FSAR interface requirements with respect to the main and auxiliary feedwater systems: (1) an increase in the feedwater isolation valve closure time from 4.6 to 9.6 seconds, (2) a reduction in the auxiliary feedwater flowrate from 875 to 750 gpm per pump, (3) an increase in the auxiliary feedwater pump start times when normal alternating current is available from 10 seconds to 22 and 29 seconds for the motor and turbine driven pumps, respectively, and (4) an increase from 15 to 23 seconds in the time delay in which interrupted auxiliary feedwater flow must be fully re-established in the steam generators.

In response to those exceptions, the licensee was asked to submit reanalyses of a spectrum of Chapter 15 analyses noting changes caused by the implementation of the PVNGS plant-specific data and providing sufficient detail for NRC review. To fulfill these NRC requirement, the six cases originally analyzed for the main steam line break event (MSLB) in the CESSAR FSAR were reanalyzed and submitted. The licensee initially submitted (in addition to reanalysis of the steam generator tube rupture and main feedwater line break accidents and small and large break loss of coolant accidents) revised results of MSLB Case 1 (descriptions provided in Section 2.4 of this report) to demonstrate acceptability of the impact from the changes identified above. These were found by the NRC to be acceptable (PVNGS SSER7). The staff has also accepted the licensee's assertion that Cases 2 through 6 of the CESSAR FSAR are bounded by the results presented for Case 1 (SSER7). However, the staff required reanalysis of Cases 2 through 6 to

demonstrate that conclusion. The staff further indicated in SSER7 that the results presented for Case 1 (submitted by letter dated November 5, 1984) reviewed in such SSER met the acceptance criteria for this event and were therefore acceptable. In SSER8, the staff stated that the licensee had agreed that all six MSLB accidents required reanalysis. Since the issuance of SSER7, analysis assumptions and ESF modeling were modified in the ANPP MSLB methodology. It was therefore necessary to review Case 1 as reanalyzed with the modified methodology.

This review, therefore, evaluates acceptability of (1) all six reanalyses of the main steam line break accident and (2) the modified methodology (which is based on the CESEC-III methodology submitted by ANPP for its PVNGS Units 1, 2, and 3). This review also examines the reanalyses to ensure that the conclusion reached in SSER7 that Cases 2 through 6 are bounded by Case 1 remains valid and (3) that the acceptance criteria were met in all cases.

2.0 EVALUATION

2.1 Steam Line Break Analysis Methodology

Four sets of changes were made to the mathematical model and input and initial conditions in CESEC steam line break methodology as described in Appendix C of the FSAR for the CESSAR System 80: (1) an explicit model for the PVNGS AFW model, boron injection and mixing model, and the upper head/pressurizer voiding model (2) plant initial conditions for the worst return-to-power, (3) analysis assumption; and (4) single failure assumptions.

2.1.1 Mathematical Models

2.1.1(a) Auxiliary Feedwater System

An explicit hydraulic model for the PVNGS AFW system was developed to model variations in flow with steam generator pressure and pump speed and for the possibility of spurious, coincident actuation of flow to both steam generators. The previous CESSAR analysis model assumed a constant flowrate

of 243 lbm/sec to the intact steam generator only. ANPP stated that all major pumps, valves, and recirculation lines are explicitly modeled in this new model, including elevation and frictional line loss effects based on the PVNGS specific configuration. This new model resulted in higher AFW flowrates, causing the steam generators to have more net inventory therefore causing more aggregate cooling. Thus we find these modeling changes conservative with respect to reactivity increase in the SLB event (while tending to be a more accurate representation of the plant) and therefore acceptable.

2.1.1(b) Boron injection

The CESEC boron injection model was upgraded for two effects:

1. Nodalization of the safety injection lines between the high and low pressure safety injection pumps and the cold legs was changed to account for the possibility of a non-uniform initial boron concentration distribution.

2. Mixing of newly injected boron is assumed to take place only in the downstream portion of the thermal-hydraulic node of the reactor coolant system into which it is injected. The fraction of the cold leg node which contributes to this mixing is calculated to be 0.28.

Both of these changes tend toward a more realistic model. This modeling approach is reasonable because: (1) a non-uniform boron concentration is possible; and (2) injected fluid would be expected to mix only with the downstream fluid. The downstream mixing results in higher boron concentrations than the prior model, however, it is conservative because the downstream cold leg volume into which the injection fluid is mixed is modeled to be conservatively large causing a lower boron concentration than would be expected in reality where a boron concentration front would be built up which would move through the downstream node. Therefore we find this model to be acceptable.

2.1.1(c) Voiding in Reactor Vessel Upper Head (RVUH)

A non-equilibrium model option is not available in the RVUH region in the CESEC-III computer code. Although this is not a problem during the period of decreasing RCS pressure, during the repressurization phase, vapor in the RVUH region would be expected to be in non-equilibrium with cooler entering water. Without a non-equilibrium model, the voids would be computed to collapse too rapidly. The modification implemented by ANPP to more realistically model upper head void behavior (or condensation), simply numerically shifts the RVUH steam void volume to the pressurizer (where the volume is correspondingly increased) where non-equilibrium effects are modeled. This modification keeps the RCS pressure higher, thus causing lower safety injection and boron flow into the RCS. This will result in less negative reactivity insertion due to delayed and decreased boron injection, causing more conservative return to power. Although this model change causes higher pressures and therefore has, in that respect only, a non-conservative impact on DNB, the combined effect of the increased heat flux associated with the higher return-to-power and the increased pressure still resulted in a lower (and therefore more conservative) DNB computation. Thus we find this model change acceptable.

2.1.2 Plant Initial Conditions

For PVNGS, the most adverse initial plant condition for return-to-power was found to be the maximum core power, most positive axial shape index (ASI), minimum core flowrate, maximum pressurizer water level, maximum core inlet coolant temperature, maximum reactor coolant system pressure, and maximum water level in both steam generators. The only condition different from those used in CESSAR is the initial water level in the steam generators. Use of the greater level causes greater cooldown and is therefore conservative.

2.1.3 Analytical Assumptions

Actuation of AFW increases the potential for higher return-to-power during a steam line break by permitting more inventory to blowdown from the break and

preventing earlier dry-out of the steam generators. The earlier analysis had assumed that AFW was not actuated to the affected steam generator. As a conservative analysis assumption in the reanalysis, AFW is assumed to be actuated simultaneously to both steam generators at the time of MSIV closure. When the steam generator differential pressure interlock conditions are reached AFW to the affected steam generator is isolated. The introduction of AFW to the affected steam generator causes greater over-cooling because of the increased secondary side inventory. This AFW logic is found to be conservative.

2.1.4 Single Failure Assumptions

The CESSAR FSAR identified two most adverse single failures: (i) for Cases 1 and 3 the worst single failure was failure of HPSI pump to start following SIAS with concurrent loss of offsite power, and (ii) for Cases 2 and 4 the worst single failure was the failure of the main steam isolation valve (MSIV) to close on the intact steam generator following an MSIS. However, the total steam flow path downstream of the MSIV in the PVNGS has a flow area of 0.034 ft^2 instead of 0.256 ft^2, therefore a resulting RCS cooldown is less severe for PVNGS than for System 80. Thus for PVNGS, failure of a HPSI pump was found to produce the most adverse transient results. The PVNGS single failure assumption is acceptable.

2.2 Analytical Assumptions

Because the CESSAR is a standardized plant design, the CESSAR FSAR is limited in its plant-specific applicability to the extent that it describes and analyzes the CESSAR System 80 design in a generic sense. Deviations in the PVNGS analysis assumptions caused by differences from the standard design are described below.

Differences between CESSAR steam line break analysis assumptions and PVNGS plant-specific steam line break analysis assumptions are:

1. AFW switched to intact side on 325 psid between two steam generators.

In the CESSAR FSAR MSLB analyses, the AFW logic prevented actuation to the affected steam generator. In the PVNGS MSLB analysis AFW is assumed to be actuated simultaneously to both steam generators at the time of MSIV closure. AFW to the affected steam generator is isolated when the lockout condition is reached. Steam generator differential pressure lockout setpoint of 325 psid is assumed. Actuation is assumed to occur at the time of main steam isolation valve closure following the main steam line isolation signal.

2. Feedwater isolation valve closure time of 9.6 sec vs 4.6 sec.

3. Minimum SI pump flow rates.

4. AFW flowrate of 750 gpm vs 875 gpm.

5. AFW isolation to the affected steam generator in 16 seconds after Condition 1 (described above) is satisfied.

Plant data differences are:

6. Length of piping between RWST and RCS is shorter implying that boron gets into the RCS faster.

7. A non-isolatable leak is only 1.5% of the design steam rate if MSIV fails to close.

The single failure assumed in the analysis is a failure of one of the high pressure safety injection pumps to start following SIAS. This was identified by a parametric study as the one causing the most adverse effect for all cases except Case 5. For Case 5, there is no single failure which increases the potential for fuel failure or which increases offsite dose. However, the same single failure was assumed for all six cases analyzed.

In addition to the differences resulting from the plant-specific parameters,

ANPP used a three dimensional peaking factor of 100 (instead of 45 used in CESSAR) for the post-trip, return-to-power, DNBR calculations.

Other assumptions used in the reanalyses are: the reactivity feedback data were generated for an end of equilibrium self-generated plutonium recycle (SGP) core which yields the most negative moderator coefficient which provides the largest potential for return to power. A most reactive control rod is assumed to be held in the fully withdrawn position.

Due to the presence of an integral flow restrictor in each steam generator outlet nozzle located upstream of the MSIV in the System 80 design, the largest possible steam line break size is limited by the flow restrictor throat area of 1.283 ft^2.

No operator actions are assumed for 30 minutes.

We found all of the above described assumptions used in PVNGS MSLB analysis to be consistent with the requirements of the SRP for this event and therefore acceptable. In addition, these assumptions are either consistent with or more conservative than those used in the CESSAR.

2.3 Accident Analyses

Six cases were analyzed; four to maximize potential for a post-trip return-to-power and two to maximize potential for degradation in fuel performance (i.e., with respect to MDNBR) and offsite dose:

For worst return-to-power,

1. a steam line break at full power with concurrent loss of offsite power (LOOP), a single failure, and a stuck CEA (SLBFPLOP);
2. a steam line break at full power with a single failure and a stuck CEA (SLBFP);
3. a steam line break at zero power with concurrent LOOP, a single failure, and a stuck CEA (SLBZPLOP); and

4. a steam line break at zero power with a single failure and a stuck CEA (SLBZP).

For worst dose and worst MDNBR,

5. a steam line break outside containment at full power with a single failure and a stuck CEA (SLBFPD); and
6. a steam line break outside containment at full power with concurrent LOOP, a single failure, and a stuck CEA (SLBFPLOPD).

The first four cases were chosen to assess the impact upon potential for return-to-criticality and the limiting time for tripping the reactor coolant pumps by assuming break initiation at full or zero power and with or without concurrent loss of offsite power.

The last two cases determine the limiting steam line break with respect to maximizing radiological consequences to the environment as well as the potential for fuel degradation.

2.3.1 Case 1: Steam Line Break at Full Power and Concurrent Loss of Offsite Power (SLBFPLOP)

Although this analysis had been previously approved by the NRC, since the analysis assumptions and methodology were modified, it has been reassessed in this review.

Because the limiting event occurs during the greatest cooldown, the steam generator initial mass inventory was maximized. This maximized the amount of total energy removed from the primary system. The PVNGS initial steam generator inventories are higher than those used in the CESSAR FSAR SLB analysis (197,000 lbm vs 182,000 lbm for the affected SG and 197,000 lbm vs 148,000 lbm for the intact SG). This is a conservative assumption since it results in more blowdown of the affected SG in the PVNGS. This was evidenced by the higher integrated steam flow and in longer time to the affected steam generator dry-out. This effect coupled with more feedwater delivery to the

affected SG in the first 50 seconds because of the revised AFW model. The AFW remains a secondary effect.

Higher blowdown of the secondary side caused the RCS pressure to be lower in the PVNGS analysis than in the CESSAR analysis (the pressurizer emptied at about 90 seconds for PVNGS and 120 seconds for CESSAR). Correspondingly the PVNGS core average temperature was slightly lower, which resulted in higher moderator reactivity feedback when compared to the CESSAR results.

The PVNGS RCS pressure remained lower than the CESSAR RCS pressure during the first 300 seconds. At this point the affected steam generator dried out (230 seconds for the CESSAR analysis). Re-pressurization of the PVNGS RCS began at the cessation of heat removal by the affected SG and the initiation of the SI flow, while the CESSAR RCS pressure continued to decrease. This difference is in large part due to the implementation of the non-equilibrium RVUH model in the PVNGS analysis. The increase in pressure reduces HPSI flow and is therefore conservative.

The computed boron transport time was roughly 100 seconds which is longer than approximately 80 seconds used in the CESSAR analysis despite the fact that the PVNGS specific data related to the shorter length of piping between the refueling water storage tank and the cold leg was used. This is because in the PVNGS analysis, one HPSI pump was assumed inoperable as the worst single failure criterion, while in the CESSAR calculation the worst single failure was the failure of an MSIV to close.

The MDNBR was computed to be about 3.86 in the PVNGS analysis at almost 380 seconds, roughly 100 seconds earlier than in the CESSAR analysis which resulted in an MDNBR of less than 2.5.

The maximum return-to-power was computed to occur at 372 seconds after break initiation. The peak post-trip reactivity was computed to reach +0.05%$\Delta\rho$ (which is smaller than the +0.09%$\Delta\rho$ for CESSAR Case 1). However, this was not sufficient to decrease the post-trip DNBR below 1.19. Consequently, fuel failure was assumed not to occur.

Notwithstanding the fact that this reanalysis resulted in higher MDNBR and a smaller post-trip peak reactivity than those obtain in the CESSAR analysis, the subject reanalysis reviewed at this time is based on modeling changes which are acceptable, and the results are acceptably conservative and bounded by the results of CESSAR.

2.3.2 Case 2: Full Power Steam Line Break (SLBFP)

This case differs from Case 1 in that in this case offsite power is assumed available. A reactor trip signal is generated by the core protection calculators on a projected DNBR of 1.19 or a variable overpower trip signal or a high containment pressure trip signal for a break inside containment. Due to continued availability of offsite power at the time of the MSLB, the reactor was tripped at 7.1 seconds as compared with 0.75 seconds in Case 1, and the reactor coolant pumps (RCPs) were not tripped at reactor trip. Continued power generation prevented a RCS cooldown in the same magnitude as Case 1 during the pre-trip period, and thus secondary pressure was kept higher. In addition keeping the RCPs running kept primary coolant circulating through the broken steam generators thereby continuing heat transfer to the secondary side. Keeping the RCPs running resulted in higher steaming rate out the break which led to greater RCS depressurization after the reactor trip, resulting in the actuation of SI flow earlier than did in Case 1.

The maximum transient reactivity computed for this case was $-0.45\%\Delta\rho$ and the MDNBR was computed to be higher than 10. Therefore, Case 1 bounds Case 2.

2.3.3 Case 3: Zero Power Steam Line Break with Concurrent Loss of Offsite Power (SLBZPLOP)

The difference between this case and Case 1 is that in this case the break is assumed to occur at zero power instead at full power.

The initial conditions were adjusted to reflect the plant conditions at zero power. At zero power, the maximum initial steam generator mass inventory was assumed to be 311,000 lbm (as compared with 197,000 lbm in Case 1) to maximize steaming and therefore RCS cooldown.

The reactor was computed to trip at 0.6 second on a low DNBR trip signal. Much faster depressurization was computed to occur because there was no decay power, and the pressurizer emptied at 35 seconds and safety injection began at 36 seconds.

The maximum transient reactivity was computed to be -0.22% $\Delta\rho$ and the MDNBR was computed to remain above 10.

2.3.4 Case 4: Zero Power Steam Line Break (SLBZP)

This is basically the same as Case 3 except for the availability of offsite power and therefore, in this case the RCPs were left running at reactor trip.

As was in Case 3, the maximum initial SG mass inventory was assumed to be 311,000 lbm.

The maximum transient reactivity was computed to be -0.154% $\Delta\rho$ and the MDNBR was computed to remain above 10. Thus Case 4 is also bounded by Case 1.

2.3.5 Case 5: Full Power Steam Line Break Outside Containment (SLBFPD)

Case 5 was analyzed to maximize the potential for fuel degradation and thus worst DNB and for the worst offsite dose release. Except for the break location outside the containment, the accident sequence was similar to Case 2.

The assumed initial RCS pressure was reduced to 2139 psia to worsen DNB. The affected steam generator mass inventory was reduced to 122,000 lbm to maximize the cooldown during the first 10 seconds into the transient. A lower initial inventory leads to a lower secondary pressure for a given energy loss by vaporization (i.e., for a given amount of steaming, lower

liquid mass results in lower liquid temperature). This leads to a lower secondary temperature and a corresponding increase in energy removed from the primary system (larger primary to secondary temperature differential).

The Doppler coefficient used for this case was smaller to permit the most rapid power increase during a rise in fuel temperature. This also permitted more rapid decrease in DNBR prior to reactor trip because of the increased heat flux.

This case resulted in a computed MDNBR of 1.11 at 7.49 seconds and caused the maximum transient reactivity to reach -1.95% $\Delta\rho$ at 93 seconds. The lower reactivity is due to the smaller aggregate primary fluid cooldown resulting from the smaller secondary inventory. This computed MDNBR is below the MDNBR limit of 1.19. Although the computation predicted that 0.4% of the fuel pins will experience DNB, in computing radiological consequences it was assumed that 0.7% of the fuel rods experienced DNB. This conforms to the assumption of 0.7% used in the previously approved CESSAR analysis. In terms of return-to-power, this case is bounded by Case 1 but not in terms of DNB.

2.3.6 Case 6: Zero Power Steam Line Break Outside Containment with Concurrent Loss of Offsite Power (SLBZPLOPD)

This case is bounded by Case 3 since in Case 3 the break can be inside or outside the containment.

This case assumed the technical specification steam generator tube leak and higher initial mass inventories for steam generators (311,000 lbm). This assumption, however, has no impact on the accident sequence or the computer parameters during the period of 30 minutes before the operator actions are initiated but it will maximize the offsite dose during the first 2 hour period.

This case poses no challenge from either a return-to-power or MDNB standpoint.

3.0 CONCLUSIONS

ANPP's steam line break methodology using the CESEC-III computer code as modified and described in Appendix 15C of PVNGS FSAR Chapter 15 was reviewed and found to be as or more conservative than that used in the CESSAR FSAR analysis and found to be acceptable.

Due to the use of more plant-specific data and assumptions in the parametric analyses of steam line break event, six cases were reanalyzed and presented for comparison to the accident sequences considered in the CESSAR FSAR Chapter 15 Section 15.1.3. We find that Case 1 bounds Cases 2 through 6 from a return-to-power perspective. Case 5 is the limiting SLB for worst offsite radiological consequence. The licensee has also shown that at full power, the limiting time for tripping the RCPs is at time zero.

We have adequate assurances that these analyses are conservative and meet the acceptance criteria as follows:

PVNGS has met the requirement of GDC 27 and 28 by demonstrating that limited fuel damage (less than 0.7%) resulted during the course of the event and that no loss of core cooling capability resulted.

PVNGS has met the requirement of GDC 31 with respect to demonstrating the integrity of the primary system boundary to withstand the postulated accident.

PVNGS has met the requirement of GDC 35 by demonstrating the adequacy of the emergency cooling systems to provide core cooling and reactivity control through a boron injection.

The analyses presented effects of steam line break accidents inside and outside containment, during various modes of operation with and without offsite power. These have been evaluated using a mathematical model that was reviewed and found acceptable.

4.0 REFERENCES

1. Letter from E.E. Van Brunt, Jr. (ANPP) to G.W. Knighton (NRC), "Main Steam Line Break Analyses Results - Chapter 15 Reanalyses," September 30, 1985.

2. "Analysis Methods for Large Steam Line Breaks," Appendix 15C to PVNGS-FSAR Chapter 15, February 15, 1985.

3. Letter from E.E. Van Brunt, Jr., to G. Knighton (NRC), "Palo Verde Nuclear Generating Station (PVNGS) Units 1, 2 and 3 Revised PVNGS FSAR Pages," ANPP-31053, November 5, 1984.

4. "Safety Evaluation Report related to the Final Design of the Standard Nuclear Steam Supply Reference System CESSAR System 80," NUREG-0852 Supplement No. 2, September 1983.

5. "Safety Evaluation Report CESEC Digital Simulation of a Combustion Engineering Nuclear Steam Supply System," October 27, 1983.

6. "Accident Analysis (Chapter 15)," NUREG-0857 Supplement 7, December 1985.

7. "Accident Analysis (Chapter 15)," NUREG-0857 Supplement 8, December 1985.

8. "Steam System Piping Failure Inside and Outside Containment," reanalysis of steam line break analysis for Palo Verde FSAR Chapter 15.1.5, Amendment No. 9, February 27, 1984

9. "Accident Analysis Methods for Large Steam Line Break," CESSAR-FSAR Chapter 15 Appendix 15C, Amendment No. 9, February 27, 1984

10. Letter from E.E. Van Brunt, Jr. (ANPP) to USNRC, "ANPP Responses to NRC Main Steam Line Break Questions," May 22, 1988.

11. Letter from D.B. Karner (ANPP) to USNRC, "ANPP Responses to NRC Questions on Main Steam Line Break ANalyses," December 12, 1988.