ENCLOSURE

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of PACIFIC GAS AND ELECTRIC COMPANY Diablo Canyon Power Plant Units 1 and 2	Docket No. 50-275 Facility Operating License No. DPR-80 Docket No. 50-323 Facility Operating License No. DPR-82 License Amendment Request No. 88-08

Pursuant to 10 CFR 50.90, Pacific Gas and Electric Company (PG&E) hereby applies to amend its Diablo Canyon Power Plant (DCPP) Facility Operating License Nos. DPR-80 and DPR-82 (Licenses).

The proposed changes amend the Technical Specifications (Appendix A of the Licenses) as regards Safety Limits 2.1.1, 2.2.1 and associated Bases, Technical Specifications 3/4.1.1.3, 3/4.1.3.1, 3/4.1.3.4, 3/4.1.3.5, 3/4.1.3.6, 3/4.2.1, 3/4.2.2, 3/4.2.3, 3/4.2.5, Table 3.3-4 and associated Bases, and Administrative Control 6.9.1.8.

Information on the proposed changes is provided in Attachments 1 through 4.

These changes have been reviewed and are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92 or require an environmental assessment in accordance with 10 CFR 51.22(b). Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

Subscribed in San Francisco, California, this 29th day of November 1988.

Respectfully submitted,

Pacific Gas and Electric Company

By _____
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By _____
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Subscribed and sworn to before me
this 29th day of November 1988.

_____
Therese Toliver, Notary Public in
and for the City and County of
San Francisco, State of California

OFFICIAL SEAL
THERESE TOLIVER
NOTARY PUBLIC - CALIFORNIA
CITY AND COUNTY OF SAN FRANCISCO
My Comm. Expires Dec. 25, 1990

My commission expires December 25, 1990

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Attachment 1

A. DESCRIPTION OF CHANGES AND AMENDMENT REQUEST

This license amendment request (LAR) proposes to revise the Technical Specifications in order to use Westinghouse VANTAGE 5 fuel for Diablo Canyon Units 1 and 2. The proposed Technical Specification changes are as follows:

1. The index and page numbers were revised to incorporate the page changes to the Technical Specifications.

2. Figures 2.1-1a and 2.1-1b, "Reactor Core Safety Limit," were revised. The changes to Figure 2.2-1a include a new figure applicable to Units 1 and 2, Cycle 4 and after. The figure was revised due to the Improved Thermal Design Procedure (ITDP) calculations performed to reflect the VANTAGE 5 fuel design. Figure 2.2-1b was revised to make it applicable to Unit 2, Cycle 3 and correct a typographical error.

3. Table 2.2-1, "Reactor Trip System Instrumentation Trip Setpoints," Item 12, "Reactor Coolant Flow-Low" was revised to require the trip setpoint and allowable values for Units 1 and 2, Cycle 4 and after to be greater than or equal to 90 percent of minimum measured flow per loop and 89 percent of minimum measured flow per loop, respectively. Additionally, the minimum measured flow of 89,000 gpm per loop for Unit 1 and 90,625 gpm per loop for Unit 2 were added. These changes are due to the ITDP calculations performed to reflect the VANTAGE 5 fuel design.

4. Table 2.2-1, "Reactor Trip System Instrumentation Trip Setpoints," was revised to include the values for K_1, K_2, K_3, K_4, K_5 and K_6 for Units 1 and 2, Cycle 4 and after. The requirements for setting gains for the overtemperature delta T reactor trip were revised to include the values for Units 1 and 2, Cycle 4 and after operation. These changes are due to the ITDP calculations performed to reflect the VANTAGE 5 fuel design.

5. Bases 2.1.1, "Reactor Core," was revised to address the DNB correlations and design and safety analysis DNBR limits for the VANTAGE 5 fuel. Also, the discussion for Power Range Neutron Flux High Rate was revised and a clarification added concerning measurement uncertainties for the F delta H limit. These changes reflect the DNB correlations used for the VANTAGE 5 and low-parasitic (LOPAR) fuel with the ITDP calculations.

6. Specification 3.1.3.4, "Rod Drop Time," was revised to increase the rod drop time for Units 1 and 2, Cycle 4 and after from 2.2 seconds to 2.7 seconds. This change is the result of an increase in the core hydraulic resistance due to the VANTAGE 5 fuel design.

7. Specification 3.2.1, "Axial Flux Difference," was revised to reference Figures 3.2-1a for Units 1 and 2, Cycle 4 and after operation and 3.2-1b for Unit 2, Cycle 3 operation. The implementation of VANTAGE 5 fuel requires a revision to these figures to reflect the new F(delta I) offset wings of -14% to +12% for Relaxed Axial Offset Control (RAOC). Surveillance Requirement 4.2.1.1b was deleted and Surveillance Requirement 4.2.1.1.a.2 was revised. This surveillance requirement is associated with Constant Axial Offset Control, which was superseded by RAOC for Unit 1 in License Amendments 3 and 12 and for Unit 2 in License Amendments 1 and 10. Since penalty points are no longer accumulated, it is unnecessary to monitor AFD more frequently than once per hour when the AFD monitor alarm is inoperable.

8. Specification 3.2.2, "Heat Flux Hot Channel Factor - $F_Q(z)$", was modified to be two specifications. Specification 3.2.2.1 is applicable to Units 1 and 2, Cycle 4 and after. Specification 3.2.2.1 includes an F_Q value of 2.45, which was used in the safety analysis for the VANTAGE 5 fuel design. This specification uses an F_Q surveillance instead of an F_{xy} surveillance to show compliance with the Limiting Condition for Operation. This change was made for operational flexibility and has been previously licensed at other plants. Specification 3.2.2.2 is applicable to Unit 2 Cycle 3 and is essentially the same as the present Specification 3.2.2. The associated Bases section has been revised to be consistent with the changes to the specifications.

9. Specification 3/4.2.3, "RCS Flow Rate and Nuclear Enthalpy Rise Hot Channel Factor," was revised to include an F delta H limit of 1.56 for LOPAR fuel during Units 1 and 2, Cycle 4 and after operation and an F delta H limit of 1.59 for VANTAGE 5 fuel during Units 1 and 2, Cycle 4 and after operation. Figure 3.2-3a was revised for Unit 1, Cycle 4 and after operation. Figure 3.2-3c was added to address Unit 2, Cycle 4 and after operation.

   These values for F delta H were used in the safety analysis for VANTAGE 5 and LOPAR fuel. The associated Bases section has been revised to be consistent with the specification change.

10. Table 3.2-1, "DNB Parameters," was revised to change the actual reactor coolant system Tavg from less than or equal to 582 degrees F to less than or equal to 584.3 degrees F for Units 1 and 2, Cycle 4 and after, and to change the actual pressurizer pressure from greater than or equal to 2220 psig to greater than or equal to 2212 psig for Units 1 and 2, Cycle 4 and after. These values represent the values assumed in the safety analysis for Tavg and pressurizer pressure.

11. Table 3.3-4, "Engineered Safety Features Actuation System Instrumentation Trip Setpoints," was revised to change the allowable values for Tavg low-low for Units 1 and 2, Cycle 4 and after. This change is due to a reevaluation of the temperature function for the VANTAGE 5 fuel transition.

12. Specification 3/4.10.2, "Group Height, Insertion and Power Distribution Limits," was revised to refer to the correct specification numbers as a result of the change described in Item 8 above.

13. Bases 3/4.2.5, "DNB Parameters," was revised to add the DNBR limits. This change is due to the new DNB correlations used for VANTAGE 5 and LOPAR fuel.

14. Specification 6.9.1.8, "Radial Peaking Factor Limit Report," was revised to include the W(z) function associated with the F_Q Surveillance Technical Specifications. Also, the report addressee was revised to be in accordance with 10 CFR 50.4.

B. BACKGROUND

This LAR incorporates the Technical Specification changes that are necessary for the replacement of Westinghouse LOPAR fuel assemblies with Westinghouse VANTAGE 5 fuel assemblies. This replacement will be performed over three cycles. In Cycle 4 for both units, the core will include reload regions of VANTAGE 5 assemblies. Cycles 5 and 6 in both units will contain increasing numbers of VANTAGE 5 fuel assemblies with an essentially full VANTAGE 5 core in place by Cycle 6. Detailed descriptions of the VANTAGE 5 fuel and the Westinghouse reload methodology are provided in WCAP-10444-P-A (Reference 2) and WCAP-9272-P-A (Reference 3). Both of these reports have received generic approval from the NRC. WCAP-9272-P-A was used in the reload evaluation for Diablo Canyon Unit 1 Cycle 4, as it was for the previous reload designs for both Units.

Fuel Assembly Design

The VANTAGE 5 fuel assemblies are mechanically compatible with the current LOPAR fuel assemblies, control rods, and reactor internal interfaces and they are hydraulically compatible with the LOPAR fuel assemblies presently in Units 1 and 2. A brief summary of the VANTAGE 5 design features and major advantages of the improved fuel design are given below:

<u>Optimized Fuel Rod Diameter</u> - The fuel rod diameter was slightly reduced while maintaining the cladding thickness the same. This optimizes the moderation and improves the neutronics.

<u>Integral Fuel Burnable Absorber (IFBA)</u> - The IFBA features a thin boride coating on the fuel pellet surface in the central portion of the fuel column. IFBAs provide power peaking and moderator temperature coefficient control.

<u>Intermediate Flow Mixer (IFM)</u> - Three IFM grids located in the upper grid spans provide increased DNB margin.

Reconstitutable Top Nozzle - A mechanical disconnect feature facilitates the top nozzle removal. Changes in the design of both the top and bottom nozzles and increased fuel rod length provide extended fuel burnup by providing additional plenum space and room for fuel rod growth.

Extended Burnup Capability - The VANTAGE 5 fuel design will be capable of achieving region average burnups in excess of 40,000 MWD/MTU. The basis for designing to extended burnups is contained in the approved WCAP-10125-P-A (Reference 28).

Axial Blankets - The axial blanket consists of a section of natural UO_2 pellets at each end of the fuel column to reduce neutron leakage and to improve uranium utilization. For VANTAGE 5 reload cores, low leakage loading patterns (burned blankets in peripheral locations) are shown to further improve uranium utilization and provide additional pressurized thermal shock margin.

Analytical Changes

The analysis of the LOPAR and VANTAGE 5 fuel was based on the Improved Thermal Design Procedure (ITDP) (Reference 12) which has received generic approval from the NRC. For the accidents analyzed to justify the use of VANTAGE 5 fuel at DCPP, the LOPAR fuel was analyzed with the WRB-1 DNB correlation (Reference 16). The VANTAGE 5 fuel was analyzed with the WRB-2 correlation as described in Reference 2. Both the WRB-1 and the WRB-2 DNB correlations take credit for the significant improvement in the accuracy of critical heat flux predictions over previous DNB correlations, and the WRB-2 correlation accurately predicts the DNB performance of the VANTAGE 5 fuel assembly design with the IFM grids.

The ITDP was employed to meet the DNB design basis. Uncertainties in plant operating parameters, nuclear and thermal parameters, and fuel fabrication parameters were considered statistically such that there was at least a 95 percent probability with a 95 percent confidence level that the minimum DNBR would be greater than or equal to the safety analysis limit for the limiting power rod. Plant parameter uncertainties were used to determine the plant DNBR uncertainty. This DNBR uncertainty, combined with the DNBR limit, established a DNBR value that must be met in plant safety analyses. Since the parameter uncertainties were considered in determining the design DNBR value, the plant safety analyses were performed using values of input parameters without uncertainties.

The DNBR margin between the DNBRs used in the safety analyses and the design DNBR limits was more than sufficient to compensate for the DNBR penalties associated with transition core and rod bow effects and allowed for design flexibility.

The change to VANTAGE 5 fuel required reanalysis of the small and large break loss of coolant accidents (LOCA) to demonstrate compliance with the requirements of 10 CFR 50.46 and to develop VANTAGE 5 specific peaking factor limits. The small break LOCA analysis was performed using the Westinghouse NOTRUMP model described in Reference 19. The large break

LOCA analysis was performed using the approved Westinghouse 1981 Evaluation Model + BASH described in Reference 17. The PAD Code described in Reference 5 was used to provide fuel data in the small and large break LOCA analysis. The NOTRUMP, BASH and PAD codes have been previously approved by the NRC.

A seismic evaluation was performed for a full core of VANTAGE 5 fuel and the transition cores. The results of the seismic and LOCA accident analyses were combined and evaluated to demonstrate compliance with Appendix A to Standard Review Plan (SRP) 4.2. The fuel racks in the spent fuel pool were evaluated for the storage of VANTAGE 5 fuel. Also, the offsite doses were recalculated for three accidents identified as having changes in results that could impact offsite doses.

C. JUSTIFICATION

VANTAGE 5 fuel is being implemented at DCPP due to certain advantages it provides over the LOPAR fuel currently in use. VANTAGE 5 fuel is expected to provide more margin to seismic limits and lower plant radiation doses. The VANTAGE 5 fuel has extended burnup capabilities for longer cycles and can provide lower leakage loading patterns to reduce the fluence on reactor vessel welds.

D. SAFETY EVALUATION

Attachment 2 of this submittal contains a safety evaluation/analysis report for the region by region reload transition from the present DCPP LOPAR fueled cores to all VANTAGE 5 fueled cores. The evaluation examines the differences between VANTAGE 5 and LOPAR fuel assembly designs and evaluates the effect of these differences on the cores during the transition to an all VANTAGE 5 core. The VANTAGE 5 core evaluation/analyses used a core thermal power level of 3338 MWt for Unit 1 and 3411 MWt for Unit 2 with the following conservative assumptions: a full power F delta H of 1.65 for the VANTAGE 5 fuel and 1.62 for the LOPAR fuel, an increase in the maximum F_Q to 2.45 for both fuel designs, 15 percent plant total steam generator tube plugging for both Units 1 and 2, and a positive moderator temperature coefficient (PMTC) of +5 pcm/degree F from 0 percent to 70 percent power and then decreasing linearly to 0 pcm/degree F from 70 to 100 percent power. The axial offset strategy will be the licensed RAOC with F_Q surveillance. RAOC uses a -14%, +12%, delta I axial offset band. Attachment 2 also includes an evaluation of the spent fuel pool high density racks for storage of VANTAGE 5 fuel and an evaluation of the radiological consequences of DCPP operation with VANTAGE 5 fuel.

Based on the information provided in this submittal, PG&E believes there is reasonable assurance that the Technical Specification changes associated with operating DCPP Units 1 and 2 with VANTAGE 5 fuel will not adversely affect the health and safety of the public.

E. NO SIGNIFICANT HAZARDS EVALUATION

PG&E has evaluated the hazard considerations involved with the proposed amendment, focusing on the three standards set forth in 10 CFR 50.92(c) as quoted below:

The commission may make final determination, pursuant to the procedures in paragraph 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or paragraph 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or

(3) Involve a significant reduction in a margin of safety.

The following evaluation is provided for the three categories of the significant hazards consideration standards.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The VANTAGE 5 fuel assemblies are mechanically and hydraulically compatible with the current LOPAR fuel assemblies, control rods, and reactor internals interfaces. Also, implementation of VANTAGE 5 fuel does not cause a significant change in the physics characteristics of the DCPP Units 1 and 2 cores beyond the normal range of variation seen from cycle-to-cycle. The proposed changes have been assessed from a core design and safety analysis standpoint. No increase in the probability of occurrence of any accident was identified. Extensive reanalyses, as described in this submittal, were undertaken to demonstrate compliance with the revised DCPP Technical Specifications. The methods used to perform the analyses have been previously approved by the NRC. The results, which include transition core effects, show changes in consequences of accidents previously evaluated. However, the results are all clearly within NRC acceptance criteria and demonstrate the plant's capability to operate safely.

   Seismic and LOCA structural integrity analyses were performed for homogeneous and transition cores of VANTAGE 5 fuel. The use of VANTAGE 5 fuel does not affect the probability of a seismic or LOCA event. For a homogeneous core of VANTAGE 5 fuel, the combined forces of a seismic and LOCA event do not result in grid deformation and the peak cladding temperature (PCT) is less than the 10 CFR 50.46 limit. Limited grid deformation is predicted in the peripheral fuel assemblies for the limiting combined loads from LOCA and seismic forces in transition cores. A coolable geometry analyses was performed and the PCT limit was met. For transition cores, the

analyses show changes in the consequences of accidents previously evaluated. However, the results are all clearly within NRC acceptance criteria. The spent fuel pool high density racks were evaluated for storage of VANTAGE 5 fuel. The evaluation determined that the storage of VANTAGE 5 fuel does not affect the conclusions in PG&E's LAR 85-13 (DCL-85-333, dated October 30, 1985) and all supporting documentation regarding reracking of the spent fuel pools at Diablo Canyon Units 1 and 2.

The radiological consequences of previously analyzed accidents were reviewed for transition to VANTAGE 5 fuel. All offsite doses remain less than the 10 CFR 100 values.

Therefore, the proposed amendment does not result in an increase in the probabilities or consequences of a previously evaluated accident.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed changes do not significantly affect the overall method and manner of DCPP operation and can be accommodated without compromising the performance or qualification of safety-related equipment.

   Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the change involve a significant reduction in the margin of safety?

   The evaluations and analyses described in this submittal to support the Technical Specification changes and operation of DCPP Units 1 and 2 with VANTAGE 5 fuel show some changes in the consequences of previously analyzed accidents. In some cases, an increase in event consequences occurs. However, in all cases the results of the changes are clearly within all plant design and NRC safety acceptance criteria.

   Therefore, the proposed amendment does not significantly reduce the margin of safety.

F. NO SIGNIFICANT HAZARDS CONSIDERATIONS DETERMINED

Based on the above evaluation, PG&E concludes that the activities associated with this LAR satisfy the no significant hazards consideration standards of 10 CFR 50.92(c) and, accordingly, a no significant hazards consideration finding is justified.

G. ENVIRONMENTAL EVALUATION

The generic environmental effects of the uranium fuel cycle are provided in Table S-3 of 10 CFR 51.51 "Uranium Fuel Cycle Environmental Data," and

the environmental impact of transportation of fuel and waste to and from a reference reactor is provided in Table S-4 of 10 CFR 51.52, "Environmental Effects of Transportation of Fuel and Waste." These regulations currently limit fuel burnup to 33 Gigawatt Days/Metric Ton Uranium (GWD/MTU) and fuel enrichment to 4 weight percent U-235. In converting to VANTAGE 5 fuel, PG&E anticipates extended burnups to greater than 33 GWD/MTU and the use of fuel with initial enrichments to 4.5 weight percent U-235. The use of fuel enriched to 4.5 weight percent U-235 in DCPP Units 1 and 2 was approved by the NRC in License Amendments 7 and 5, respectively, dated May 13, 1986.

The environmental effects of extended burnup and higher initial enrichments have recently been addressed by the NRC. A notice published in the Federal Register on February 29, 1988 (53 FR 6054), states that the NRC's environmental assessment of extended burnup fuel is complete and that environmental impacts summarized in Tables S-3 of 10 CFR 51 and S-4 of 10 CFR 52 bound the corresponding impacts for burnup levels up to 60 GWD/MTU and enrichments up to 5 weight percent U-235.

The environmental impacts of transportation resulting from the use of extended burnup and higher enrichment fuel were further addressed in the Federal Register on August 11, 1988 (53 FR 30355). This notice reiterated the conclusion stated in 53 FR 6054 and further concluded that there are no significant adverse radiological or nonradiological impacts associated with the use of extended burnup and/or increased enrichment, and that burnup levels to 60 GWD/MTU and enrichments to 5 weight percent U-235 will not significantly affect the quality of the human environment. Moreover, pursuant to 10 CFR 51.31, "Determinations Based on Environmental Assessment," the Commission determined that an environmental impact statement need not be prepared for this action.

The NRC Staff is in the process of revising the regulations in 10 CFR 51.52 to reflect the findings published in the above cited Federal Register Notice. In the interim, in connection with its review of proposed license amendments to permit use of fuel enriched to 5 weight percent U-235 and burnup levels to 60 GWD/MTU, and pursuant to 10 CFR 51.52(b), the NRC Staff proposes to accept the analysis of the environmental effects of the transportation of such fuel and waste provided in 53 FR 30355 until such time as the revision to the rule is issued.

Based on the generic environmental evaluations regarding extended burnup and transportation provided by the NRC in the above cited Federal Register Notices and PG&E's evaluation of other environmental aspects of the proposed fuel design changes, PG&E has concluded that the proposed changes do not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed changes meet the eligibility criterion for categorical exclusion set forth in 10 CFR 51.52(c)(9). Therefore, pursuant to 10 CFR 51.22(b), an environmental assessment of the proposed changes is not required.

Attachment 2

SAFETY EVALUATION
FOR THE
DIABLO CANYON POWER PLANT UNITS 1 AND 2
TRANSITION TO WESTINGHOUSE 17x17 VANTAGE 5 FUEL

PACIFIC GAS AND ELECTRIC COMPANY
DIABLO CANYON POWER PLANT UNITS 1 AND 2
DOCKET NOS. 50-275 AND 50-323

November 1988

2422S/0065K

TABLE OF CONTENTS

LIST OF TABLES

LIST OF FIGURES

1.0 INTRODUCTION

The Diablo Canyon Power Plant (DCPP) Units 1 and 2 are currently operating with a Westinghouse 17x17 low-parasitic (LOPAR) fueled core. For subsequent cycles, it is planned to refuel and operate the DCPP Units 1 and 2 with the Westinghouse VANTAGE 5 improved fuel design. As a result, future core loadings would range from approximately 60 to 70 percent LOPAR, and 30 to 40 percent VANTAGE 5 transition cores (Cycle 4) to eventually an all VANTAGE 5 fueled core in Cycle 6. The VANTAGE 5 fuel assembly is designed as a modification to the current 17x17 LOPAR (standard fuel) and the optimized fuel assembly (OFA) designs, Reference 1.

The VANTAGE 5 design features were conceptually packaged to be licensed as a single entity. This was accomplished via the NRC review and approval of the "VANTAGE 5 Fuel Assembly Reference Core Report," WCAP-10444-P-A, Reference 2. The initial irradiation of a fuel region containing all the VANTAGE 5 design features occurred in the Callaway Plant in November 1987. The Callaway VANTAGE 5 licensing submittal was made to the NRC on March 31, 1987 (USNRC-1470, Docket No. 50-483). NRC approval was received on October 9, 1987. The Virgil C. Summer Nuclear Station also submitted a request to the NRC on May 20, 1988, to implement VANTAGE 5 fuel. All of the VANTAGE 5 design features have been successfully licensed as individual design features and are currently operating in Westinghouse plants. DCPP Unit 1 is operating in Cycle 3 and Unit 2 will be operating in Cycle 3 with a region of LOPAR fuel containing the licensed reconstitutable top nozzle, extended burnup modified fuel assemblies, and IFBAs.

A brief summary of the VANTAGE 5 design features and major advantages of the improved fuel design are provided in Attachment 1. These features and figures illustrating the design are presented in more detail in Section 3.0 of this attachment.

The evaluation contained in this attachment is intended to serve as a reference safety evaluation/analysis report for the region-by-region reload transition from the present DCPP LOPAR fueled core to an all VANTAGE 5 fueled core. The evaluation examines the differences between the VANTAGE 5 and LOPAR fuel assembly designs and evaluates the effect of these differences on the cores during the transition to an all VANTAGE 5 core. The VANTAGE 5 core evaluation/analyses incorporates a core thermal power level of 3338 MWt for Unit 1 and 3411 MWt for Unit 2 with the following conservative assumptions made in the safety evaluations: a full power $F\Delta H$ of 1.65 for the VANTAGE 5 fuel and 1.62 for the LOPAR fuel, an increase in the maximum F_Q to 2.45, and 15 percent plant total steam generator tube plugging for both Units 1 and 2. As in previous cycles, a positive moderator temperature coefficient (PMTC) of +5 pcm/degree F from 0 percent to 70 percent power and then decreasing linearly to 0 pcm/degree F from 70 to 100 percent power. The axial offset strategy will be the previously licensed RAOC with F_Q surveillance. RAOC uses a -14%, +12% ΔI axial offset band.

This evaluation utilizes the standard reload design methods described in Reference 3 and will be used as a basic reference document in support of future DCPP Reload Safety Evaluations (RSEs) for VANTAGE 5 fuel reloads.

Sections 3.0 through 8.0 of the evaluation summarize the mechanical, nuclear, thermal and hydraulic, spent fuel pool, accident evaluations, and radiological consequences, respectively. Parameters were chosen to maximize the applicability of the safety evaluations for future cycles. The objective of subsequent cycle-specific RSEs will be to verify that applicable safety limits are satisfied based on the reference safety evaluation/analyses established in this evaluation.

In order for Westinghouse to demonstrate performance of the VANTAGE 5 design features in a commercial reactor, four VANTAGE 5 demonstration assemblies (17x17) were loaded into the Virgil C. Summer Unit 1 Cycle 2 core and began power production in December of 1984. These assemblies completed one cycle of irradiation in October of 1985 with an average burnup of 11,357 MWD/MTU. Post-irradiation examinations showed all four demonstration assemblies were of good sound mechanical integrity. No mechanical damage or wear was evident on any of the VANTAGE 5 components. Likewise, the intermediate flow mixer (IFM) grids on the VANTAGE 5 demonstration assemblies had no effect on the adjacent fuel assemblies. All four demonstration assemblies were reinserted into Virgil C. Summer Unit 1 for a second cycle of irradiation. This cycle was completed in March of 1987, at which time the demonstration assemblies achieved an average burnup of about 30,000 MWD/MTU. The observed behavior of the four demonstration assemblies at the end of two cycles of irradiation was comparable to that observed at the end of the first cycle of irradiation. The four assemblies were reinserted for a third cycle of irradiation.

In addition to Virgil C. Summer, individual VANTAGE 5 product features have been demonstrated at other nuclear plants. IFBA demonstration fuel rods have been irradiated in Turkey Point Units 3 and 4 for two reactor cycles. Unit 4 contains 112 fuel rods equally distributed in four demonstration assemblies. The IFBA coating performed well with no loss of coating integrity or adherence. The IFM grid feature has been demonstrated at McGuire Unit 1. The demonstration assembly at McGuire was irradiated for three reactor cycles and showed good mechanical integrity. DCPP Unit 1 is operating in fuel Cycle 3 and Unit 2 will be operating in Cycle 3 with LOPAR fuel containing the licensed reconstitutable top nozzle, extended burnup modified fuel assemblies, and IFBAs.

2.0 SUMMARY AND CONCLUSIONS

Consistent with the Westinghouse standard reload methodology for analyzing cycle-specific reloads, Reference 3, parameters were selected to conservatively bound the values for each subsequent reload cycle and to facilitate determination of the applicability of 10 CFR 50.59. The objective of subsequent cycle-specific reload safety evaluations will be to verify that applicable safety limits are satisfied based on the reference evaluation/analyses established in this evaluation. The mechanical, thermal and hydraulic, nuclear, and accident evaluations considered the transition core effects described in Reference 2 for a VANTAGE 5/LOPAR mixed core.

The NRC Staff reviewed the VANTAGE 5 reference core report, WCAP-10444, Reference 2, and concluded that the report is acceptable for reference for the Westinghouse VANTAGE 5 fuel design subject to specific conditions. These conditions, summarized in Section 6.0 of the SER for WCAP-10444, have been considered in the DCPP specific safety evaluations.

The results of the evaluation/analyses described herein lead to the following conclusions:

1. The Westinghouse VANTAGE 5 reload fuel assemblies for DCPP Units 1 and 2 are mechanically compatible with the current LOPAR fuel assemblies, control rods, and reactor internals interfaces. The VANTAGE 5/LOPAR fuel assemblies satisfy the current design bases as given in Reference 2 for DCPP Units 1 and 2.

2. The VANTAGE 5 fuel assembly responses under seismic and LOCA excitations were determined using an analytical model representative of the DCPP reactor core. Analysis of the 17x17 VANTAGE 5 fuel assembly component stresses and grid impact forces due to postulated faulted condition accidents verified that the VANTAGE 5 fuel assembly design is structurally acceptable.

3. Changes in the nuclear characteristics due to the transition from LOPAR to VANTAGE 5 fuel will be within the range normally seen from cycle-to-cycle due to fuel management effects.

4. The reload VANTAGE 5 fuel assemblies are hydraulically compatible with the LOPAR fuel assemblies from previous cycles of operation.

5. The core design and safety analyses results documented in this evaluation show the core's capability for operating safely for the rated DCPP Units 1 and 2 design thermal power with $F\Delta H$ of 1.65, and 1.62 for VANTAGE 5 and LOPAR fuel, respectively, F_Q of 2.45, and steam generator tube plugging levels up to 15 percent.

6. Previously reviewed and licensed safety limits continue to be met when the DCPP Units 1 and 2 are reloaded with VANTAGE 5 fuel. Plant operating limitations given in the Technical Specifications will be satisfied with the proposed changes noted in Attachment 1. A reference is established

upon which to base Westinghouse reload safety evaluations for future reloads with VANTAGE 5 fuel.

7. The storage of VANTAGE 5 fuel in the high density spent fuel storage racks does not alter the conclusions reached in PG&E's LAR 85-13 (DCL-85-333, dated October 30, 1985) and all supporting documentation regarding reracking of the spent fuel pools at Diablo Canyon Units 1 and 2.

8. The radiological consequences of previously analyzed accidents remain much less than the 10 CFR 100 guideline values of 300 rem thyroid and 25 rem whole body.

3.0, MECHANICAL EVALUATION

This section evaluates the mechanical design and the compatibility of the 17x17 VANTAGE 5 fuel assembly with the current LOPAR fuel assemblies during the transition through mixed-fuel cores to an all VANTAGE 5 core. The VANTAGE 5 fuel assembly has been designed to be compatible with the LOPAR fuel assemblies, reactor internals interfaces, and the fuel handling equipment. The VANTAGE 5 design is intended to replace and be compatible with fuel cores containing fuel of the LOPAR designs. The VANTAGE 5 design dimensions as shown on Figure 3.1 are equivalent to the LOPAR designs from an exterior assembly envelope and reactor internals interface standpoint. The design basis and design limits are essentially the same as those for the LOPAR designs. As such, compliance with the "Acceptance Criteria" of the Standard Review Plan (SRP, NUREG 0800) Section 4.2, Fuel System Design, was fully demonstrated.

3.1 VANTAGE 5 Description

The significant new mechanical features of the VANTAGE 5 design relative to the LOPAR fuel design include the following:

- Optimized Fuel Rod Diameter
- Integral Fuel Burnable Absorber (IFBA)
- Intermediate Flow Mixer (IFM) Grids
- Reconstitutable Top Nozzle
- Extended Burnup Capability
- Axial Blankets

Table 3.1 provides a comparison of the VANTAGE 5 and LOPAR fuel assembly design parameters.

3.1.1 Fuel Rod Performance

Fuel rod performance for all fuel rod designs is shown to satisfy the SRP fuel rod design bases on a region by region basis. These same bases are applicable to all fuel rod designs, including the Westinghouse LOPAR and VANTAGE 5 fuel designs, with the only difference being that the VANTAGE 5 fuel is designed to achieve a higher burnup consistent with WCAP-10125-P-A (Reference 28) and to operate with a higher F delta H limit. The design bases for Westinghouse VANTAGE 5 fuel are discussed in Reference 2, Section 2.4.

The IFBA coated fuel pellets are identical to the enriched uranium dioxide pellets except for the addition of a thin coating on the pellet cylindrical surface. Coated pellets occupy the central portion of the fuel column. The number and pattern of IFBA rods may vary depending on the specific loading cycle. The ends of the enriched coated pellets and enriched uncoated pellets are dished to allow for greater axial expansion at the pellet centerline and to provide void volume for fission gas release. Analysis of IFBA rods includes any geometry changes necessary to model the presence of burnable absorber, and conservatively models the gas release from the coating. An evaluation and test program for the IFBA design features are given in Section 2.5 in Reference 2.

Fuel performance evaluations are completed for each fuel region to demonstrate that the design bases will be satisfied for all fuel rod types in the core under the planned operating conditions. Any changes from the plant operating conditions originally considered in the mechanical design of a fuel region (for example, a power uprating or an increase in the peaking factors) are addressed for all affected fuel regions. Fuel rod design evaluations are currently performed using the NRC-approved models in References 4, 5, 6 and 27 to demonstrate that the SRP fuel rod design criteria (including the rod internal pressure design basis in Reference 7) will be satisfied.

3.1.2 Grid Assemblies

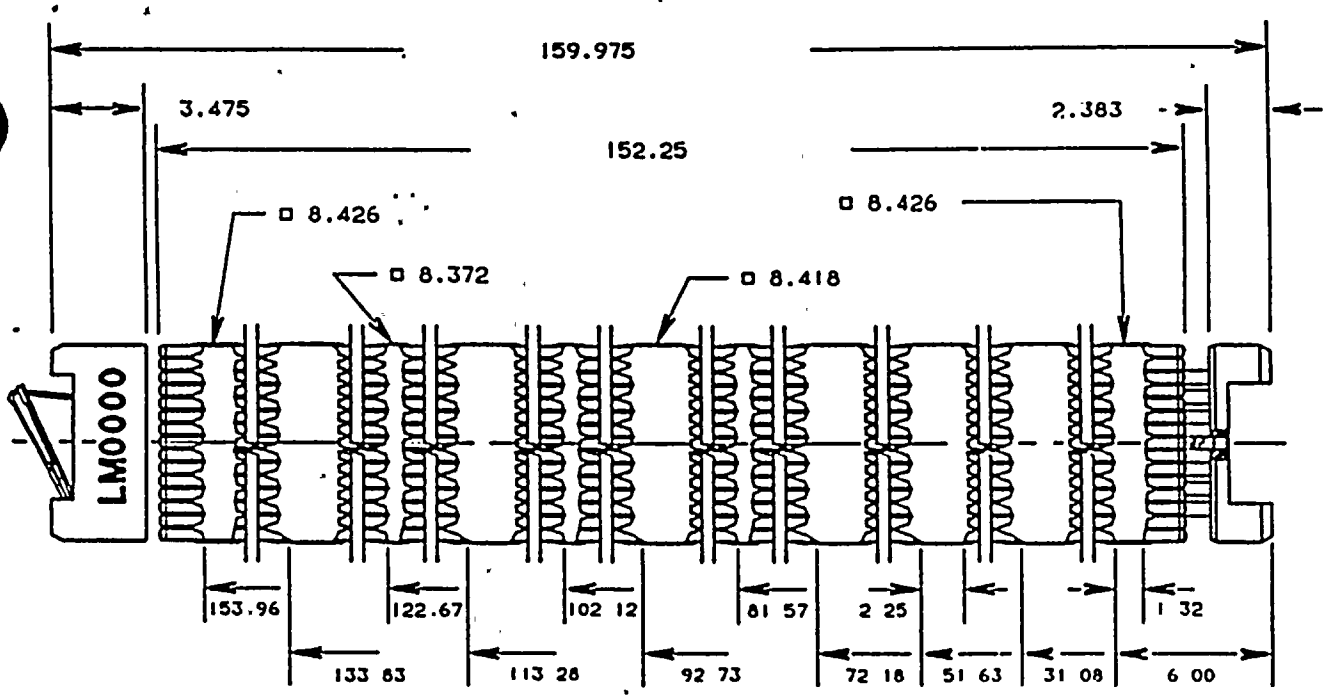
The top and bottom Inconel (non-mixing vane) grids of the VANTAGE 5 fuel assemblies are nearly identical in design to the Inconel grids of the LOPAR fuel assemblies. The only differences are: (1) the grid spring and dimple heights have been modified to accommodate the reduced diameter fuel rod, (2) the grid spring force has been reduced in the top grid, and (3) the outer grid strap configuration has been modified for the snag-resistant feature. The six intermediate (mixing vane) structural grids are made of Zircaloy material rather than the Inconel used in the LOPAR design, the straps are thicker and the grid height is greater compared to the LOPAR design.

The Intermediate Flow Mixer (IFM) grids shown in Figure 3.1 are located in the three uppermost spans between the Zircaloy mixing vane structural grids and incorporate a similar mixing vane array. Their prime function is mid-span flow mixing in the hottest fuel assembly spans. Each IFM grid cell contains four dimples which are designed to prevent midspan channel closure in the spans containing IFMs and fuel rod contact with the mixing vanes. This simplified cell arrangement allows short grid cells so that the IFM grid can accomplish its flow mixing objective with minimal pressure drop. The IFM grids are not intended to be structural members. The outer strap configuration was designed similar to current fuel designs to preclude grid hang-up and damage during fuel handling.

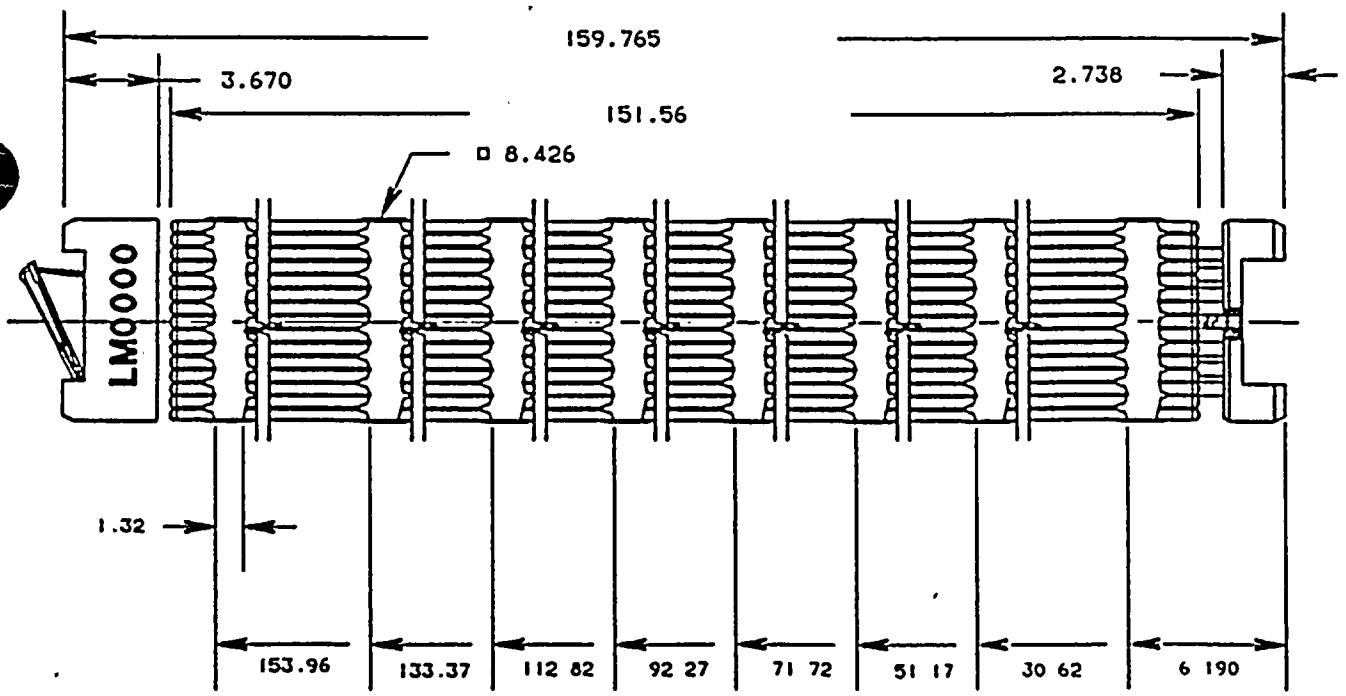
3.1.3 Reconstitutable Top Nozzle and Bottom Nozzle

The reconstitutable top nozzle for the VANTAGE 5 fuel assembly differs from the LOPAR design in two ways: a groove is provided in each thimble thru-hole in the nozzle plate to facilitate attachment and removal; and the nozzle plate thickness is reduced to provide additional axial space for fuel rod growth.

To remove the top nozzle, a tool is first inserted through a lock tube and expanded radially to engage the bottom edge of the tube. An axial force is then exerted on the tool which overrides local lock tube deformations and withdraws the lock tube from the insert. After the lock tubes have been withdrawn, the nozzle is removed by raising it off the upper slotted ends of the nozzle inserts which deflect inwardly under the axial lift load.



17X17 VANTAGE 5 FUEL ASSEMBLY



17X17 RECONSTITUTABLE LOPAR FUEL ASSEMBLY

DIABLO CANYON UNITS 1 AND 2
FIGURE 3.1 17X17 VANTAGE 5 / LOPAR FUEL ASSEMBLY COMPARISON

With the top nozzle removed, direct access is provided for fuel rod examinations or replacement. Reconstitution is completed by the remounting of the nozzle and the insertion of lock tubes. Additional details of this design feature, the design bases and evaluation of the reconstitutable top nozzle are given in Section 2.3.2 in Reference 2. The VANTAGE 5 bottom nozzle design is similar to the LOPAR design except it is shorter and has a thinner top plate to allow for fuel rod growth. The design bases and evaluation of the VANTAGE 5 bottom nozzle are given in Section 2.3.1 in Reference 2. As noted in Reference 29, the VANTAGE 5 bottom nozzle will continue to be fabricated from stainless steel, which differs from the VANTAGE 5 Inconel bottom nozzle described in Reference 2. The stainless steel bottom nozzle meets all design requirements.

3.1.4 Axial Blankets

Although noted as a new mechanical feature of the VANTAGE 5 design and licensed in Reference 2, axial blankets have been and are currently operating in Westinghouse plants. A description and design application of this feature are contained in Reference 2, Section 3.0. The DCPP axial blanket design differs from that described in Reference 2 in one aspect. Recent changes utilize a chamfered pellet physically different from the enriched pellet in the fuel stack to help prevent accidental mixing with the enriched pellet.

3.1.5 Mechanical Compatibility of Fuel Assemblies

Based on the evaluation of the VANTAGE 5/LOPAR design differences and hydraulic test results (References 1, 2), it is concluded that the two designs are mechanically compatible with each other. The VANTAGE 5 fuel rod mechanical design bases remain unchanged from that used for the LOPAR fuel assemblies.

3.1.6 Rod Bow

It is predicted that the 17x17 VANTAGE 5 rod bow magnitudes, like those of the Westinghouse OFA fuel, will be within the bounds of existing 17x17 LOPAR assembly rod bow data. The current NRC-approved methodology for comparing rod bow for two different fuel assembly designs is given in Reference 8.

Rod bow in fuel rods containing IFBAs is not expected to differ in magnitude or frequency from that currently observed in Westinghouse LOPAR fuel rods under similar operating conditions. No indications of abnormal rod bow have been observed on visual or dimensional inspections performed on the test IFBA rods. Rod growth measurements were also within predicted bounds.

3.1.7 Fuel Rod Wear

Fuel rod wear is dependent on both the support conditions and the flow environment to which the fuel rod is subjected. Due to the LOPAR and VANTAGE 5 fuel assembly designs employing different grids, there is an unequal axial pressure distribution between the assemblies. Crossflow resulting from this unequal pressure distribution was evaluated to determine the induced rod vibration and subsequent wear. Hydraulic tests, (Reference 2, Appendix A.1.4) were performed to verify that the crossflows were negligible and also to check

hydraulic compatibility of the LOPAR and VANTAGE 5 designs. The VANTAGE 5 fuel assembly was flow tested adjacent to a 17x17 OFA, since vibration test results indicated that the crossflow effects produced by this fuel assembly combination would have the most detrimental effect on fuel rod wear.

Results of wear tests with VANTAGE 5 fuel compared with both LOPAR and OFA fuel were found to be satisfactory. It was concluded that the VANTAGE 5 fuel rod wear would be less than the maximum wear depth established (Reference 9) for the 17x17 LOPAR or OFA at end-of-life.

3.2 Seismic/LOCA Effect on Fuel Assemblies

The structural adequacy of the current standard fuel assembly design under the Design, Double Design, and Hosgri earthquakes for DCPP Units 1 an 2 has been previously demonstrated and documented in the FSAR Update. The VANTAGE 5 fuel assembly to be used in DCPP core reloads is mechanically and structurally compatible with the LOPAR fuel design. Based on load bearing capability under the faulted condition transients, the VANTAGE 5 fuel assembly design is structurally superior to the LOPAR fuel.

The basic VANTAGE 5 fuel assembly structure is essentially the same as the standard fuel design consisting of 264 fuel rods, 24 guide thimble tubes, and one central instrumentation tube. The fuel rods are inserted into the skeleton assembly and retained in position by the friction of the grid springs. The fuel assembly structure is completed by means of mechanical bolting and bulge fastening of the top and bottom fuel nozzles to the assembly skeleton.

The VANTAGE 5 fuel assembly has a total of eleven grids, including three Zircaloy IFMs, six internal Zircaloy grids, and two end Inconel grids. The total mass for the VANTAGE 5 fuel assembly is about 6 percent less than that of the LOPAR design. The proportional reduction in fuel mass and assembly structural stiffness preserves the fundamental frequency of the VANTAGE 5 fuel assembly when compared with the LOPAR fuel design. Based upon the fuel assembly geometrical configurations and dimensional considerations, the VANTAGE 5 fuel is compatible with the LOPAR design.

The structural adequacy of the VANTAGE 5 fuel assembly was evaluated using the current NRC requirements for combined seismic and LOCA loads in accordance with Appendix A to the NRC SRP 4.2. The approved analysis methodology, Reference 9, including the fuel modeling, grid strength, computer code WEGAP, and analytical procedures, was used to assess the VANTAGE 5 fuel design.

The principal fuel assembly acceptance criteria for the LOCA are: (1) fuel rod fragmentation must not occur as a direct result of the blowdown loads, (2) the 10 CFR 50 temperature and oxidation limits must not be exceeded, and (3) the fuel assembly must remain amenable to control rod insertability. Two criteria for the safe shutdown earthquake (SSE) are: (1) fuel rod fragmentation must not occur as a result of the seismic loads, and (2) the fuel assembly must remain amenable to control rod insertability. An appropriate combination of loads from normal operation and accident conditions must be made.

The VANTAGE 5 design adequacy has been evaluated through a comparative analysis of load bearing capability for the two types of fuel assembly designs and the plant-specific seismic/LOCA analyses for both DCPP Units. The resultant loads acting on the structural grids and flow mixers are lower than that of the LOPAR fuel assembly design. It was concluded that the VANTAGE 5 fuel assembly has more margin in withstanding a faulted condition transient load than that of the LOPAR fuel assembly design.

The plant specific seismic and LOCA analyses were performed for the homogeneous core using the time-history method. The results of the combined seismic and LOCA analyses indicate that there is no grid deformation for both DCPP units under homogeneous VANTAGE 5 fuel core operation. In view of the plant-specific seismic and LOCA analyses for the VANTAGE 5 fuel assembly design, it was concluded that the VANTAGE 5 fuel core coolability and the requirements for control rod insertability for reactor safe shutdown are met. The details of the analysis and the results of both postulated accidents (SSE and LOCA) are presented below:

3.2.1 Seismic Analysis Results

The time-history method was used to obtain the maximum fuel assembly relative deflections and grid impact loads. The synthesized time-history was generated based on actual earthquake records. Two sets of acceleration response spectra for the synthesized time-histories envelope separately the design spectra of the Double Design and Hosgri earthquakes for the Diablo Canyon site. The acceleration spectra of the Double Design Earthquakes (DDE) have been reviewed with those for the Hosgri earthquakes at the reactor vessel support elevation. The applied spectra for Hosgri earthquakes envelope those of the DDE in the fuel assembly frequency range of about 2-5 Hz. Therefore, the fuel assembly integrity evaluation was based on the Hosgri earthquake loadings. The core plate motions were obtained from the analysis of the reactor pressure vessel/internals system model for both units. The core plate motions were applied as input forcing functions to the reactor core model consisting of discrete lumped masses, linear spring/damper and gap finite elements. The results of the seismic analysis of the VANTAGE 5 fuel cores indicated that the maximum grid impact forces were within the allowable grid strength for the DCPP Units 1 and 2 due to a Hosgri earthquake. The allowable grid strength was established as the lower 95 percent confidence level on the true mean from the distribution of experimentally determined grid deformation strength data at temperature. The maximum fuel assembly relative deflection was calculated to be 1.13 inches. Based on the conservatively assumed maximum fuel assembly deflection of 1.15 inches, the resulting stresses in fuel rod and thimble were below the established allowable values.

Based on the maximum responses obtained from the seismic analysis, together with the induced fuel assembly component stresses and the allowable limits, it was concluded that the 17x17 VANTAGE 5 fuel assembly is structurally adequate in maintaining the coolable geometry in the event of a Design, Double Design, or a Hosgri earthquake.

3.2.2 Asymmetric LOCA Analysis Results

The asymmetric LOCA analysis was performed to obtain the grid impact forces and fuel assembly deflections resulting from two separate main coolant pipe breaks. The analytical methodology, fuel assembly models, and core model are consistent with that used in the seismic analysis. The LOCA results indicated that the most limiting coolant pipe break was identified either at the reactor pressure vessel inlet nozzle or at the reactor coolant pump outlet nozzle.

The results of the asymmetric LOCA analysis of the VANTAGE 5 fuel cores indicated that the maximum grid impact forces were within the allowable grid strength for DCPP Units 1 and 2 and, therefore, there was no grid deformation. The maximum fuel assembly relative deflection was calculated to be 0.72 inches. The maximum fuel assembly deflection is well below that obtained from seismic analysis. The fuel assembly component stress resulting from LOCA transients would indicate substantial margins in maintaining fuel assembly structural integrity.

Based on the maximum responses obtained from the seismic analysis, together with the induced fuel assembly component stresses and the allowable limits, it was concluded that the 17x17 VANTAGE 5 fuel assembly is structurally capable of maintaining the coolable geometry throughout the transient, and the fuel assembly is amenable for control rod insertion in the event of a pipe rupture accident.

3.2.3 Vertical LOCA Analysis

Two major primary coolant pipe rupture transients, RPV inlet pipe and pump outlet pipe breaks, were used in the vertical blowdown analysis to establish the maximum impact force response. The maximum impact forces at both the upper and lower fuel nozzles were obtained from the RPV system model analysis. These loads were used for assessing the fuel assembly component's structural integrity. The vertical LOCA load results indicated that the most limiting coolant pipe break for the vertical impact load acting on the fuel nozzles was the double-ended reactor coolant pump outlet break.

The results of the vertical LOCA analysis of the VANTAGE 5 fuel assembly in both units indicated that the maximum grid impact forces acting on the top and bottom nozzles were within allowable limits. The maximum fuel assembly component stresses resulting from these maximum impact forces were well below the established allowable limits for faulted condition loads. Thus, it was concluded that the VANTAGE 5 fuel assembly design is acceptable in resisting the vertical LOCA loads with no grid deformation.

3.2.4 Combination of Seismic and LOCA Results

To comply with the requirements in Appendix A of SRP 4.2, the maximum grid responses obtained from seismic and LOCA accident analyses were combined using the square-root-of-sum-of-squares (SRSS) method. The maximum combined grid load for the DCPP units was below the allowable grid strength.

The fuel assembly component stresses were obtained from the maximum fuel assembly relative deflection. Since the fuel assembly displacement is limited by the maximum accumulated gap clearances between assemblies plus the grid

deformations, the fuel assembly stresses presented are basically corresponding to a limiting case. The results indicate that there are adequate stress margins for the combination of stresses for both the fuel rods and thimble tubes resulting from the vertically induced impact load.

Under the combined seismic and LOCA loading conditions, the grids in a homogeneous VANTAGE 5 core will not be deformed. The fracturing of guide tubes and fragmentation of fuel rods will not occur. The reactor can be safely shut down. In conclusion, the VANTAGE 5 fuel design is structurally acceptable for use in both DCPP Units 1 and 2.

3.3 Transition Core Considerations

During transition core operation, the VANTAGE 5 fuel assemblies will be placed either next to a LOPAR fuel assembly or next to a VANTAGE 5 fuel assembly in DCPP Units 1 and 2. There is no effect from a fuel rod design standpoint due to having fuel with more than one type of geometry simultaneously residing in the core during the transition cycles. The location of the two types of fuel assemblies in a transition core is dependent on the core loading patterns. Two typical core loading patterns correspond to a one-third and a two-thirds of VANTAGE 5 fueled cores. Fuel performance evaluations were completed for each fuel region to demonstrate that the design criteria will be satisfied for all fuel rod types in the core under the planned operating conditions. Any changes from the plant operating conditions originally considered in the mechanical fuel design of a fuel region (for example, a power uprating or an increase in the peaking factors) were addressed for all affected fuel regions.

The plant-specific seismic and LOCA analyses performed for transition cores indicated that there is an incremental increase in applied grid loads on the VANTAGE 5 fuel assemblies during transition core operation. This evaluation also indicated that limited deformation of grids may occur depending on the fuel assembly loading pattern. For grid elevations and fuel assembly locations where grid deformation is postulated to occur, the core coolability evaluation was also performed using the maximum theoretical channel blockage for the both the Inconel (LOPAR) and Zircaloy (VANTAGE 5) grids. (See Section 7.2.3 for further discussion of channel blockage.) The resultant peak cladding temperature calculation indicated that the coolable geometry requirement was met. The deformation of the grids is minor and does not prevent RCCA rod insertion.

Under a separate faulted condition loading, the grid impact loads obtained from these basic loading patterns indicate that the VANTAGE 5 and LOPAR fuel assemblies are capable of withstanding LOCA transients without grid deformation. However, there will be one to two grids with limited deformation in the peripheral fuel assemblies (LOPAR and VANTAGE 5) under the Hosgri earthquake loading during transition core operation. Under combined seismic and LOCA loadings, there will also be limited grid deformation of some peripheral fuel assemblies. An evaluation of fuel assembly component stresses and critical buckling stress for the thimble tubes indicates that the limited deformation in these fuel assemblies will not impair control rod insertion, cause fuel rod fragmentation, or guide tube fracturing. Analysis showed that the 10 CFR 50.46 cladding temperature limit is met. (See Section 7.2.3 which discusses peak cladding temperature.)

3.4 Conclusions

The VANTAGE 5 fuel assembly responses under seismic and LOCA excitations were determined using an analytical model representation of the DCPP core. Table 3.2 presents the results of the seismic/LOCA analyses for (1) all LOPAR cores, (2) transition cores, and (3) all VANTAGE 5 cores. The table shows the results for LOCA forces alone, seismic forces alone, and combined seismic and LOCA forces. During the transition cycles, no grid deformation is predicted for either the VANTAGE 5 or LOPAR fuel assemblies from LOCA forces alone. The seismic forces alone result in limited grid deformation during transition cycles. Limited grid spacer deformation is also predicted in the peripheral fuel assemblies for the limiting combined loads from LOCA and seismic forces during transition cores. No grid deformation is predicted for an all VANTAGE 5 core due to seismic, LOCA, or combined seismic/LOCA forces. In all cases, the fuel assembly grid deformation is limited and will not impair control rod insertion, cause fuel rod fragmentation, or guide tube fracturing. The peak cladding temperature in all cases is below the limitation of 10 CFR 50.46.

Table 3.1

COMPARISON OF 17x17 LOPAR
and
17x17 VANTAGE 5 FUEL ASSEMBLY DESIGN PARAMETERS

PARAMETER	17x17 LOPAR DESIGN	17x17 VANTAGE 5 DESIGN
Fuel Assy Length, in.	159.765	159.975
Fuel Rod Length, in.	151.56	152.255
Assembly Envelope, in.	8.426	8.426
Compatible with Core Internals	Yes	Yes
Fuel Rod Pitch, in.	0.496	0.496
Number of Fuel Rods/Assy	264	264
Number Guide Thimble Tubes/Assy	24	24
Number Instrumentation Tubes/Assy	1	1
Fuel Tube Material	Zircaloy 4	Zircaloy 4
Fuel Rod Cladding OD, in.	0.374	0.360
Fuel Rod Cladding Thickness, in.	0.0225	0.0225
Fuel/Cladding Gap, mil	6.5	6.2
Fuel Pellet Diameter, in.	0.3225	0.3088
Fuel Pellet Length, in.	0.387	0.370

Table 3.2

TYPICAL SEISMIC/LOCA ANALYSIS GRID DEFORMATION RESULTS
FOR DIABLO CANYON UNITS 1 AND 2
(LIMITING CASES)

Grid Spacer Deformation

Core	LOCA		Seismic		Combined Seismic/LOCA	
	V5	LOPAR	V5	LOPAR	V5	LOPAR
ALL LOPAR	--	No	---	Yes	---	Yes
1/3 V5 Transition	No	No	Yes	Yes	Yes	Yes
2/3 V5 Transition	No	No	Yes	No	Yes	Yes
All V5	No	--	No	--	No	--

4.0 NUCLEAR DESIGN

The transition from the current all LOPAR fuel core to VANTAGE 5 fuel will not cause changes to the current nuclear design bases given in the Diablo Canyon FSAR Update. The evaluation of the transition and equilibrium cycle VANTAGE 5 cores presented in Reference 2, as well as the Diablo Canyon specific transition and equilibrium core evaluations, demonstrate that the impact of implementing VANTAGE 5 does not cause a significant change to the physics characteristics of the DCPP cores beyond the normal range of variations seen from cycle-to-cycle.

The methods and core models used in the DCPP reload transition core analysis are described in References 2, 3, and 10. These licensed methods and models have been used for DCPP and other previous Westinghouse reload designs using the LOPAR and VANTAGE 5 fuel. Advanced nodal analysis methods described in Reference 11 and a multidimensional transport theory based lattice cross section methodology described in Reference 25 will also be used in future cycle-specific DCPP reload design analyses.

The following DCPP Technical Specification changes are proposed for the nuclear design area:

1. Increased $F\Delta H$ limits. These higher limits will allow loading pattern designs to accommodate longer cycles as well as a greater flexibility to design loading patterns which maximize low leakage.

2. Increased F_Q limit. The increased F_Q limit will provide greater flexibility with regard to accommodating the axially heterogeneous cores (blankets and reduced length burnable absorbers).

3. Widened RAOC bandwidth. The RAOC bandwidth is being widened to allow greater operator flexibility with regard to core operation. The margin created by the increased F_Q limit is being partly converted into operational flexibility.

4. F_Q Surveillance. This revision to surveillance requirements on the heat flux hot channel factor, $F_Q(z)$, has been proposed to increase plant operating flexibility while more directly monitoring F_Q. Rather than performing surveillance on $F_{xy}(z)$, the radial component of the total peaking factor, surveillance is performed directly on $F_Q(z)$. The steady-state $F_Q(z)$ is measured and increased by applicable uncertainties. This quantity is further increased by an analytical factor called W(z) which accounts for possible increases in the steady-state $F_Q(z)$ resulting from operation within the allowed axial flux difference limits. The resulting $F_Q(z)$ is compared to the $F_Q(z)$ limit to demonstrate operation below the heat flux hot channel factor limit.

Power distributions and peaking factors show slight changes as a result of the incorporation of axial blankets, reduced length IFBAs, and increased radial peaking factor limits, in addition to the normal variations experienced with different loading patterns. The usual methods of enrichment variation and burnable absorber usage can be employed in the transition and full VANTAGE 5 cores to ensure compliance with the peaking factor Technical Specifications.

The key safety parameters evaluated for Diablo Canyon as it transitions to an all VANTAGE 5 core show little change relative to the range of parameters experienced for the all LOPAR core. The changes in values of the key safety parameters are typical of the normal cycle-to-cycle variations experienced as loading patterns change. As is current practice, each reload core design will be evaluated to assure that design and safety limits are satisfied according to the reload methodology. The design and safety limits will be documented in each cycle-specific reload safety evaluation report which serves as a basis for any significant changes which may require future NRC review.

5.0 THERMAL AND HYDRAULIC DESIGN

The analysis of the LOPAR and VANTAGE 5 fuel was based on the Improved Thermal Design Procedure (ITDP) described in Reference 12. The LOPAR fuel analysis used the WRB-1 DNB correlation in Reference 16, while the VANTAGE 5 fuel used the WRB-2 DNB correlation in Reference 2. These DNB correlations take credit for the significant improvement in the accuracy of the critical heat flux predictions over previous DNB correlations. The WRB-2 DNB correlation also takes credit for the VANTAGE 5 fuel assembly intermediate flow mixing vane design. A DNBR limit of 1.17 is applicable for both the WRB-1 and WRB-2 correlations. Table 5.1 summarizes the pertinent thermal and hydraulic design parameters.

The design method employed to meet the DNB design basis is the ITDP, which was approved by the NRC in Reference 13. Uncertainties in plant operating, nuclear, thermal-hydraulic, and fuel fabrication parameters were considered statistically such that there is at least 95 percent probability at a 95 percent confidence level that the minimum DNBR will be greater than or equal to 1.17 for the limiting power rod. Plant parameter uncertainties were used to determine the plant DNBR uncertainties. These DNBR uncertainties, combined with the DNBR limit, establish a DNBR value which must be met in plant safety analyses. Since the parameter uncertainties were considered in determining the design DNBR value, the plant safety analyses were performed using values of input parameters without uncertainties. For this application, the minimum required DNBR values for the LOPAR fuel analysis are 1.33 for thimble cold wall cells (three fuel rods and a thimble tube) and 1.37 for typical cells (four fuel rods). The design DNBR values for the VANTAGE 5 fuel are 1.30 and 1.32 for thimble and typical cells, respectively.

In addition to the above considerations, a plant-specific DNBR margin has been considered in the analyses. In particular, safety analysis DNBR limits of 1.44 for thimble and 1.48 for typical cells for LOPAR fuel, and 1.68 and 1.71 for thimble and typical cells, respectively, for the VANTAGE 5 fuel were employed in the safety analyses. The DNBR margin between the DNBRs used in the safety analyses and the design DNBR values is subdivided as follows. A fraction of the margin is utilized to accommodate the transition core penalty (a maximum 12.5 percent for VANTAGE 5 fuel and none for LOPAR fuel) and the appropriate fuel rod bow DNBR penalty, Reference 8, which is less than 1.5 percent. The existing 7.4 percent margin in the LOPAR fuel and 22.6 percent margin in the VANTAGE 5 fuel between the design and safety analysis DNBR limits also includes a greater than 5 percent DNBR margin in the LOPAR fuel and 8.5 percent DNBR margin in the VANTAGE 5 fuel reserved for flexibility in the design. The 12.5 percent generic transition core penalty is discussed in Reference 23 and received NRC approval via Reference 24.

The LOPAR and VANTAGE 5 designs have been shown to be hydraulically compatible in Reference 2.

The major impact of thimble plug removal on the thermal-hydraulic analysis is the increase in bypass flow, which is reflected in Table 5.1. This thimble plug removal has been accounted for in the DNB analysis so that the use of thimble plugs in Cycle 4 and after is not required.

The phenomenon of fuel rod bowing, as described in Reference 8, must be accounted for in the DNBR safety analysis of Condition I and Condition II events for each plant application. Internal to the fuel rod, the IFBA and fuel pellet designs are not expected to increase the propensity for fuel rods to bow. External to the VANTAGE 5 fuel rod, the Inconel non-mixing vane and Zircaloy mixing vane grids provide fuel rod support. Additional restraint is provided with the IFM grids. Applicable generic credits for margin resulting from retained conservatism in the evaluation of DNBR and/or margin obtained from measured plant operating parameters (such as F delta h or core flow), which are more restrictive than those required by the plant safety analysis, can be used to offset the effect of rod bow. The safety analysis for the DCPP maintains sufficient margin between the safety analysis limit DNBRs and the design limit DNBRs to accommodate full-flow and low-flow DNBR penalties.

The Westinghouse transition core DNB methodology is given in References 1 and 14 and has been approved by the NRC via Reference 15. A change to the VANTAGE 5 transition core penalty is discussed in Reference 23 and the recent NRC generic approval of this change is given in Reference 24. This methodology has been extended further in Reference 26. Using this methodology, first the LOPAR fuel in a transition core is bounded by its full core analysis; second, the VANTAGE 5 fuel in a transition core is analyzed as if it were a full core of VANTAGE 5 fuel, and then by applying the appropriate transition core DNBR penalty based on the fraction of VANTAGE 5 fuel actually in the transition core.

The fuel temperatures for use in safety analysis calculations for the VANTAGE 5 fuel were evaluated using the same methods as those used to evaluate the LOPAR fuel. Westinghouse uses the PAD performance code described in Reference 4 and 5 to calculate fuel temperatures used as initial conditions in safety analyses.

TABLE 5.1

DIABLO CANYON THERMAL AND HYDRAULIC DESIGN PARAMETERS

Thermal and Hydraulic Design Parameters (Using ITDP)		Design Parameters Unit 1	Design Parameters Unit 2
Reactor Core Heat Output, MWt		3338	3411
Reactor Core Heat Output, 106 BTU/Hr		11,390	11,639
Heat Generated in Fuel, %		97.4	97.4
Core Pressure, Nominal, psia		2280	2280
Radial Power Distribution (LOPAR)		1.56 [1+0.3(1-P)]	1.56 [1+0.3(1-P)]
(V-5)		1.59 [1+0.3(1-P)]	1.59 [1+0.3(1-P)]
Limit DNBR for Design Transients			
Typical Flow Channel	(LOPAR)	1.48	1.48
	(V-5)	1.71	1.71
Thimble (Cold Wall) Flow Channel	(LOPAR)	1.44	1.44
	(V-5)	1.68	1.68
DNB Correlation	(LOPAR)	WRB-1	WRB-1
	(V-5)	WRB-2	WRB-2
HFP Nominal Coolant Conditions*			
Vessel Minimum Measured Flow+			
Rate (including Bypass), 106 lbm/hr		135.4	136.6
GPM		359,200	362,500
Vessel Thermal Design Flow+			
Rate (including Bypass), 106 lbm/hr		132.2	133.4
GPM		350,800	354,000
Core Flow Rate (excluding Bypass, based on TDF)			
106 lbm/hr		122.3	123.4
GPM		324,490	327,450
Fuel Assembly Flow Area++ for Heat Transfer, ft2	(LOPAR)	51.08	51.08
	(V-5)	54.13	54.13
Core Inlet Mass Velocity, 106 lbm/hr-ft (Based on TDF)	(LOPAR)	2.39	2.42
	(V-5)	2.26	2.28

TABLE 5.1 (Continued)

DIABLO CANYON THERMAL AND HYDRAULIC DESIGN PARAMETERS

Thermal and Hydraulic Design Parameters (Based on Thermal Design Flow)		Design Parameters Unit 1	Design Parameters Unit 2
Nominal Vessel/Core Inlet Temperature, °F		544.4**	545.1**
Vessel Average Temperature, °F		576.6	577.6
Core Average Temperature, °F		580.7	581.8
Vessel Outlet Temperature, °F		608.8	610.1
Average Temperature Rise in Vessel, °F		64.4	65.0
Average Temperature Rise in Core, °F		69.1	69.7
Heat Transfer			
Active Heat Transfer Surface Area,++ ft2	(LOPAR)	59,742	59,742
	(V-5)	57,505	57,505
Average Heat Flux, BTU/hr-ft^2	(LOPAR)	185,740	189,800
	(V-5)	192,960	197,180
Average Linear Power, kW/ft		5.33	5.45
Peak Linear Power for Normal Operation, kW/ft+++		13.06	13.34

* Based on Safety Analysis T_{in} = 548.4° F and Pressure = 2280 psia

** Safety Analysis T_{in} = 548.4° F for both units

\+ Includes 15 percent steam generator tube plugging

++ Assumes all LOPAR or VANTAGE 5 core

+++ Based on 2.45 F_Q peaking factor

6.0 SPENT FUEL POOL EVALUATION

The high density spent fuel storage racks were reanalyzed for storage of both new and spent VANTAGE 5 fuel with respect to criticality, thermal-hydraulic, and seismic effects. The results of this reanalysis are summarized below.

6.1 Criticality Analysis

A reanalysis of both Region 1, which contains Boraflex, and Region 2 of the high density spent fuel racks was performed for VANTAGE 5 fuel. This analysis included normal and abnormal accident conditions, and addressed the effects of potential Boraflex shrinkage. Results of the analysis confirm that the racks can safely accommodate VANTAGE 5 fuel up to 4.5 weight percent enrichment within the reactivity limits established by the NRC ($k_{eff} < 0.95$), including uncertainties in an allowance for credible gaps in the Boraflex which might be postulated. The current Region 2 burnup versus enrichment limits of Technical Specification 3/4.9.14, "Spent Fuel Assembly Storage," encompass the VANTAGE 5 fuel and, therefore, no change to this Technical Specification is required.

6.2 Thermal-Hydraulic Analysis

The thermal-hydraulic analysis of the spent fuel pool was reanalyzed for spent VANTAGE 5 fuel. There is no change to the calculated bulk heat load in the pool due to VANTAGE 5 fuel, since the previous analysis for the high density racks covered extended fuel cycles and conservative operating histories in calculating the decay heat from spent fuel, and these values envelope those of VANTAGE 5 fuel. Results of the local temperature analysis demonstrate those the calculated peak cladding temperature and maximum local water temperature in the vicinity of the worst case VANTAGE 5 assembly stored in the high density fuel racks at end of pool life conditions have increased. The temperatures are within acceptable limits, and do not affect the conclusions reached in PG&E's LAR 85-13 (DCL-85-333, dated October 30, 1985) and all supporting documentation regarding reracking of the spent fuel pools at Diablo Canyon Units 1 and 2 [reracking documentation]. Therefore, adequate cooling for VANTAGE 5 fuel is assured.

6.3 Seismic Analysis

The mass of the VANTAGE 5 fuel assembly is approximately 94 percent of the mass of the LOPAR fuel assembly. The envelope dimensions of the two types of fuel assemblies are essentially identical. Therefore, the change to VANTAGE 5 fuel would not be expected to significantly affect the results of the seismic analysis reported in the reracking documentation. This was confirmed by performing a seismic reanalysis for a fuel rack with VANTAGE 5 fuel under postulated Hosgri loading. The rack considered in the analysis was the one that had the highest stress factors (stresses used for rack design) and nearly the largest displacement of the cases reported in the reracking documentation. The results of this reanalysis show that the critical stress factors governing rack design are decreased over the design basis results previously reported. Rack-to-rack and fuel-to-rack loads also decreased. It is concluded that use of VANTAGE 5 fuel does not affect the structural safety

of the spent fuel racks, and the integrity of the assemblies stored in the racks is assured.

6.4 Conclusions

The storage of VANTAGE 5 fuel in the high density spent fuel storage racks does not adversely affect the conclusions reached in the reracking documentation.

7.0 ACCIDENT EVALUATION

7.1 Non-LOCA Accidents

This section addresses the impact on non-LOCA accident analyses of the following proposed changes and design safety analysis assumptions for the DCPP.

Proposed Change to the Licensing Basis:

- VANTAGE 5 Fuel Design

Design Safety Analysis Assumptions:

- Positive Moderator Temperature Coefficient
- Increased Design Nuclear Enthalpy Rise Hot Channel Factor ($F\Delta H$) and LOCA F_Q
- Increase in Allowable Steam Generator Tube Plugging Level for Certain Transients (design assumption)

VANTAGE 5

The design features of VANTAGE 5 fuel, considered in the non-LOCA analysis are:

- VANTAGE 5 Fuel Rod Dimensions
- Axial Blankets
- Integral Fuel Burnable Absorbers (IFBAs) and Wet Annular Burnable Absorbers (WABAs)
- Intermediate Flow Mixer Grids (IFMs)
- Reconstitutable Top Nozzle
- Extended Burnup Fuel Assembly Design
- Thimble Plug Deletion

Fuel Rod Dimensions

The VANTAGE 5 fuel rod dimensions that determine the safety analysis temperature versus linear power density relationship include rod diameter, pellet diameter, initial pellet-to-cladding gap size, and stack height. The non-LOCA safety analysis fuel temperature and rod geometry assumptions considered this geometry change and bound both LOPAR and VANTAGE 5 fuel.

The VANTAGE 5 fuel DNB analysis for this evaluation used the ITDP and the WRB-2 correlation which is described in Appendix A of Reference 2. The LOPAR fuel DNB analysis for this evaluation used the ITDP and the WRB-1 correlation in Reference 16.

Axial Blankets and IFBAs

Axial blankets reduce power at the ends of the fuel rod which increases axial peaking at the middle section of the rod. Used alone, axial blankets reduce DNB margin, but the effect may be offset by the presence of reduced length IFBAs which flatten the power distribution. The net effect on the axial shape is a function of the number and configuration of IFBAs in the core and time in

life. The effects of axial blankets and IFBAs on the reload safety analysis parameters are taken into account in the reload design process. The axial power distribution assumption in the safety analyses kinetics calculations have been determined to be applicable for evaluating the introduction of axial blankets in the DCPP Units 1 and 2.

IFM Grids and Reconstitutable Top Nozzle

The IFM grid feature of the VANTAGE 5 fuel design increases DNB margin. The fuel safety analysis limit DNB values contain sufficient DNB margin (see Section 5.0) to ensure that the core thermal safety limits for the VANTAGE 5 fuel with an FΔH of 1.65 are acceptable. The LOPAR fuel core limits are more restrictive than the VANTAGE 5 fuel core limits. Thus, the most restrictive core limits correspond to the LOPAR fuel design. Any transition core penalty is accounted for with the available DNBR margin.

The IFM grid feature of the VANTAGE 5 fuel design increases the core pressure drop. The control rod scram time to the dashpot design assumption is increased from 2.2 to 2.7 seconds. The increased drop time primarily affects the fast reactivity transients. These accidents were reanalyzed for this report. The revised safety analysis assumption was incorporated in all the reanalyzed events requiring this parameter and the remaining transients have been evaluated.

Core flow areas and loss coefficients were preserved in the design of the reconstitutable top nozzle. As such, no parameters important to non-LOCA safety analyses are impacted.

Extended Burnup Fuel Assembly Design

WCAP-10125-P-A, Reference 28, evaluates the impact of extended burnup on the design and operation of Westinghouse fuel. The major effect of the extended burnup rod design is on power shaping between fresh and burned assemblies.

Thimble Plug Deletion

The non-LOCA analysis performed incorporates the impact of thimble plug deletion. Thimble plug deletion affects core pressure drop and slightly increases core bypass flow.

7.1.1 Increased Design Nuclear Enthalpy Rise Hot Channel Factor (FΔH)

The FΔH for the LOPAR fuel during the transition cycles is 1.62. The FΔH for VANTAGE 5 fuel is 1.65. The non-LOCA calculations applicable for the VANTAGE 5 core have assumed a full power FΔH of 1.65. This is a conservative safety analysis assumption for this report.

The design core limits for this report incorporate the increased FΔH for the LOPAR and VANTAGE 5 fuel.

7.1.2 Increased Steam Generator Tube Plugging

All non-LOCA safety analyses reanalyzed for this report have incorporated any necessary changes to model 15 percent plant total steam generator tube plugging.

7.1.3 Increase in LOCA F_Q

The increase in the Technical Specification maximum LOCA F_Q from 2.32 to 2.45 is conservatively accounted for in the non-LOCA transients.

7.1.4 Non-LOCA Safety Evaluation Methodology

The non-LOCA safety evaluation process is described in References 1 and 2. The process determines if a core configuration is bounded by existing safety analyses in order to confirm that applicable safety criteria are satisfied. The methodology systematically identifies parameter changes on a cycle-by-cycle basis which may invalidate existing safety analysis assumptions and identifies the transients which require reevaluation. This methodology is applicable to the evaluation of VANTAGE 5 transition and full cores.

Any required reevaluation identified by the reload methodology is one of two types. If the identified parameter is only slightly out of bounds, or the transient is relatively insensitive to that parameter, a simple evaluation may be made which conservatively evaluates the magnitude of the effect and explains why the actual analysis of the event does not have to be repeated. Alternatively, should the deviation be large and/or expected to have a significantly or not easily quantifiable effect on the transients, reanalyses are required. The reanalysis approach will typically utilize the analytical methods which have been used in previous submittals to the NRC. These methods are those which have been presented in FSARs, subsequent submittals to the NRC for a specific plant, reference SARs, or report submittals for NRC approval.

The key safety parameters are documented in Reference 3. Values of these safety parameters which bound both fuel types (LOPAR and VANTAGE 5) were assumed in the safety analyses. For subsequent fuel reloads, the key safety parameters will be evaluated to determine if violations of these bounding values exist. Reevaluation of the affected transients would take place and would be documented for the cycle specific reload design, as per Reference 3.

7.1.5 Conclusions

Descriptions of the transients reanalyzed for this report, method of analysis, results, and conclusions are contained in Attachment 4. The analytical procedures and computer codes used are identified in Section 15.1. Attachment 4 has been prepared conforming to the format of the DCPP FSAR Update. The FSAR Update will be revised using the information in Attachment 4 in the next applicable revision.

For each of the non-LOCA accidents reanalyzed, it was found that appropriate safety criteria are met. For each of the non-LOCA accidents not reanalyzed, evaluations were performed which determine that the existing conclusions remain applicable for the proposed changes to the plant. These transients, descriptions, results, and conclusions are documented in the DCPP FSAR Update.

7.2 LOCA Accidents

7.2.1 Large Break LOCA For Full Core of VANTAGE 5 Fuel

7.2.1.1 Description of Analysis/Assumptions

The large break loss-of-coolant accident (LOCA) analysis for the DCPP, applicable to a full core of VANTAGE 5 fuel assemblies, was performed to develop Diablo Canyon specific peaking factor limits. This is consistent with the methodology employed in the Reference Core Report for 17x17 VANTAGE 5, Reference 2. The Westinghouse 1981 Evaluation Model + BASH model, Reference 17, was utilized and a spectrum of cold leg breaks were analyzed for DCPP Unit No. 2. Other pertinent analysis assumptions include: a core thermal power of 3411 MWt, up to 15 percent plugging in each steam generator (uniform), an $F_{\Delta H}$ of 1.65, F_Q of 2.5, and fuel data based on the new fuel thermal model, Reference 5. The most limiting break determined from the DCPP Unit 2 analysis was reanalyzed for DCPP Unit 1 at the lower power rating of 3338 MWt. The analysis results, tables, and figures are presented in Attachment 4.

VANTAGE 5 fuel features, as applied to the DCPP, result in a fuel assembly that is more limiting than a LOPAR fuel assembly with respect to large break LOCA ECCS performance, Reference 2. As such, VANTAGE 5 fuel has been analyzed herein.

7.2.1.2 Method of Analysis

The methods used in analyzing the DCPP for VANTAGE 5 fuel, including computer codes used and assumptions, are described in detail in Attachment 4, Section 15.4.1.1.2.

7.2.1.3 Results

The results of this analysis, including tabular and plotted results of the break spectrum analyzed, are provided in Attachment 4, Section 15.4.1.1.3, which has been prepared using Regulatory Guide 1.70, Revision 1 for accidents applicable to the DCPP Units 1 and 2.

Reference 17 states three restrictions related to the use of the 1981 Evaluation Model + BASH calculational model. The application of these restrictions to the plant-specific large break LOCA analysis was addressed with the following conclusions.

1. DCPP Units 1 and 2 are neither Upper Head Injection (UHI) nor Upper Plenum Injection (UPI) plants, so Restriction 1 does not apply.

2. DCPP Units 1 and 2 specific LOCA analysis analyzed both minimum and maximum ECCS cases to address Restriction 2. The C_d equal to 0.4 Double Ended Cold Leg Guillotine (DECLG) with minimum ECCS flows was found to result in the most limiting consequences.

3. Generic sensitivity studies were performed by Westinghouse for a typical 4-loop plant using different power shapes. This sensitivity study demonstrated that the chopped cosine was the most limiting power shape, (Reference 22). A chopped cosine power shape was used in the large break LOCA analysis for DCPP, thus satisfying Restriction 3.

7.2.1.4 Conclusions

The large break LOCA analysis performed for DCPP Units 1 and 2 has demonstrated that for breaks up to a double-ended severance of the reactor coolant piping, the Emergency Core Cooling System (ECCS) will meet the acceptance criteria of 10 CFR 50.46, that is:

1. The calculated peak cladding temperature (PCT) will remain below the required 2200° F.

2. The amount of fuel cladding that reacts chemically with the water or steam does not exceed one percent of the total fuel rod cladding.

3. The localized cladding oxidation limit of 17 percent is not exceeded during or after quenching.

4. The core remains amenable to cooling during and after the LOCA.

5. The core temperature is reduced and decay heat is removed for an extended period of time. This is required to remove the heat produced by the long-lived radioactivity remaining in the core.

The time sequence of events for all breaks analyzed is shown in Tables 15.4-1 and 15.4-2 of Attachment 4, Section 15.4.1.

The large break LOCA analysis for DCPP, assuming a full core of VANTAGE 5 fuel (utilizing the 1981 Evaluation Model + BASH calculational model), resulted in a PCT of 2071°F for the limiting DECLG break at a total peaking factor of 2.50. The maximum local metal-water reaction was 7.40 percent, and the total core wide metal-water reaction was less than 0.3 percent for all cases analyzed. The cladding temperature transients turn around at a time when the core geometry was still amenable to cooling.

7.2.2 Small Break LOCA For Full Core of VANTAGE 5 Fuel

7.2.2.1 Description of Analysis/Assumptions

The small break LOCA was analyzed assuming a full core of VANTAGE 5 fuel to determine the peak cladding temperature. This is consistent with the methodology employed in WCAP-10444-P-A, Reference 2, for 17x17 VANTAGE 5 transition. The currently approved NOTRUMP Model Small Break ECCS Evaluation Model, Reference 19, was utilized for a spectrum of cold leg breaks. Attachment 4, Section 15.3.1, includes a full description of the analysis and assumptions utilized for the Westinghouse VANTAGE 5 ECCS LOCA analysis. Pertinent assumptions include an $F\Delta H$ of 1.65 (for both units), total peaking factors corresponding to 2.5 at the core midplane (for both units), 15 percent steam generator tube plugging (for both units), and the core thermal power level of 3338 MWt for Unit 1 and 3411 MWt for Unit 2.

Sensitivity studies performed using the NOTRUMP small break evaluation model have demonstrated that VANTAGE 5 fuel is more limiting than OFA fuel in calculated ECCS performance. Similar studies using the WFLASH evaluation model, Reference 20, have previously shown that OFA fuel is more limiting than LOPAR fuel. For the small break LOCA, the effect of the fuel difference is most pronounced during core uncovery periods and, therefore, shows up predominantly in the LOCTA-IV calculation in the evaluation model analysis. Consequently, the previous conclusion drawn from the WFLASH studies (regarding the fuel difference) may be extended to this NOTRUMP analysis. Thus, only VANTAGE 5 fuel was analyzed since it is the most limiting of the two types of fuel residing in the core.

7.2.2.2 Method of Analysis

The methods of analysis, including the codes and assumptions used, are described in detail in Attachment 4, Section 15.3.1.

7.2.2.3 Results

The results of this analysis, including tabular and plotted results of the break spectrum analyzed, are provided in Attachment 4, Section 15.3.1.

7.2.2.4 Conclusions

The small break VANTAGE 5 LOCA analysis for DCPP Units 1 and 2, utilizing the currently approved NOTRUMP Evaluation Model, resulted in a PCT of 1358°F for the 4.0 inch diameter cold leg break for Unit 2 at 3411 MWt. The limiting break size was used in a similar analysis for Unit 1 and resulted in a PCT of 1275°F. The analysis assumed the limiting small break power shape consistent with a LOCA F_Q envelope of 2.50 at core midplane elevation and 2.31 at the top of the core. The maximum local metal-water reactions are 0.133 and 0.193 percent for Units 1 and 2, respectively, and the total core metal-water reaction is less than 0.3 percent for all the cases analyzed. The cladding temperature transients turn around at a time when the core geometry is still amenable to cooling.

The analyses presented in Attachment 4, Section 15.3, show that one centrifugal pump and one high head pump, together with the accumulators, provide sufficient core flooding to keep the calculated PCT well below the required limits of 10 CFR 50.46 (for both units). The ECCS analysis also remains in compliance with all other requirements of 10 CFR 50.46. Adequate protection is therefore afforded by the ECCS in the event of a small break LOCA.

7.2.3 Transition Core Effects On LOCA

When assessing the impact of transition cores on the LOCA analysis, it must be determined whether the transition core can have a greater calculated PCT than either a complete core of the LOPAR assembly design or a complete core of the VANTAGE 5 design. For a given peaking factor, the only mechanism available to cause a transition core to have a greater calculated PCT than a full core of either fuel is the possibility of flow redistribution due to fuel assembly hydraulic resistance mismatch. Hydraulic resistance mismatch will exist only for a transition core and is the only unique difference between a complete core of either fuel type and the transition core.

Under combined seismic and LOCA loads it was determined that during the transition from 17x17 LOPAR to 17x17 VANTAGE-5 some of the fuel assemblies experienced limited grid deformation. The maximum theoretical deformation (reduction in flow area) for a 17x17 LOPAR (Inconel) grid was calculated to be 22 percent in a grid cell, while the maximum theoretical deformation for a 17x17 VANTAGE 5 grid was calculated to be 39.3 percent in a grid cell. A large break LOCA core coolability analysis was performed to demonstrate compliance with the criteria of 10 CFR 50.46. The remaining FSAR Update LOCA accident analyses were evaluated to determine acceptability of licensing requirements.

7.2.3.1 Large Break LOCA for Transition Cores

The large break LOCA analysis was performed with a full core of VANTAGE 5 and conservatively applied the blowdown results to transition cores. The VANTAGE 5 fuel assembly differs hydraulically from the LOPAR assembly design it replaces. The assembly hydraulic resistance is approximately 10 percent higher for the VANTAGE 5 design.

An evaluation of hydraulic mismatch of approximately 10% showed an insignificant effect on blowdown cooling during a LOCA. The SATAN-VI computer code models the crossflows between the average core flow channel (N-1 fuel assemblies) and the hot assembly flow channel (one fuel assembly) during blowdown. To better understand the transition core large break LOCA blowdown transient phenomena, conservative blowdown fuel cladding heatup calculations have been performed to determine the cladding temperature effect on the new fuel design for mixed core configurations. The effect was determined by reducing the axial flow in the hot assembly at the appropriate elevations to simulate the effects of the transition core hydraulic resistance mismatch. In addition, the Westinghouse blowdown evaluation model was modified to account for grid heat transfer enhancement during blowdown for this evaluation. The results of this evaluation have shown that no PCT penalty is observed during blowdown for the mixed core. Therefore, it is not necessary to perform a blowdown calculation for the VANTAGE 5 transition core configuration because the evaluation model blowdown calculation performed for the full core of VANTAGE 5 fuel is conservative and bounding.

Since the overall resistance of the two types of fuel is essentially the same, only the crossflows during core reflood due to the smaller rod size and different grid designs need to be evaluated. The LOCA analysis uses the BASH computer code to calculate the reflood transient, Reference 17, which utilizes the BART code, Reference 18. A detailed description of the BASH code is given in Attachment 4. Fuel assembly design specific analyses have been performed with a version of the BART computer code, Reference 18, which accurately models mixed core configurations during reflood. Westinghouse transition core designs, including specific 17x17 LOPAR to VANTAGE 5 transition core cases, were analyzed. For this case, BART modeled both fuel assembly types and predicted the reduction in axial flow at the appropriate elevations. As expected, the increase in hydraulic resistance for the VANTAGE 5 assembly was shown to produce a reduction in reflood steam flow rate for the VANTAGE 5 fuel at mixing vane grid elevations for transition core configurations. This reduction in steam flow rate is partially offset by the fuel grid heat transfer enhancement predicted by the BART code during reflood. The various fuel assembly specific transition core analyses performed resulted in peak

cladding temperature increases of up to 50°F for core axial elevations that bound the location of the PCT. Therefore, the maximum PCT penalty possible for VANTAGE 5 fuel residing in a transition core is 50°F, Reference 2. Once a full core of VANTAGE 5 fuel is achieved, the large break LOCA analysis will apply without the transition core penalty.

A specific large break core coolability analysis was performed to determine compliance with 10 CFR 50.46 criteria. This analysis was performed with the 1981 + BASH ECCS Evaluation Model. The analysis for the 17x17 LOPAR fuel accounted for the grid cell theoretical maximum deformation of 22 percent by modeling an equivalent assembly-wide blockage. The equivalent assembly-wide blockage was determined as the blockage that, when applied to the assembly, resulted in the identical flow reduction seen in the subchannel having the theoretical maximum flow area reduction. The equivalent blockage used in the core coolable geometry analysis for the 17x17 LOPAR fuel was based on the maximum of four grids in one assembly deforming. The location of the grid deformation is restricted to occur above the bottom two mixing vane grids and below the top two mixing vane grids. The analysis for the 17x17 VANTAGE 5 accounted for the grid cell theoretical maximum deformation of 39.3 percent also by modeling an equivalent assembly-wide blockage. The equivalent assembly-wide blockage was determined as the blockage that, when applied to the assembly, resulted in the identical flow reduction seen in the subchannel having the theoretical maximum flow area reduction. The equivalent blockage used in the core coolable geometry analysis for the 17x17 VANTAGE 5 fuel assembly was based on the maximum of four grids deforming. Therefore, the number of deformed grids postulated to occur on a 17x17 VANTAGE 5 fuel assembly is four, with the location of the grid deformation restricted to occur above the bottom two mixing vane grids and below the top two mixing vane grids.

The core coolable geometry analysis demonstrated a PCT of 1980°F for the 17x17 LOPAR fuel assembly during the transition to 17x17 VANTAGE 5, a local zirc-oxidation of 4.67 percent, and a total cladding zirc-oxidation less than 0.3%, which are well within the limits established by 10 CFR 50.46. This result for the 17x17 LOPAR fuel assembly was based on an $F_Q(z)$ of 2.45 and a Nuclear Enthalpy Rise Hot Channel Factor of 1.62. The core coolable geometry analysis demonstrated a PCT of 2108°F for the 17x17 VANTAGE 5 assembly during the transition from 17x17 LOPAR to 17x17 VANTAGE 5, a local zirc-oxidation of 7.37 percent, and a total cladding zirc-oxidation less than 0.3 percent, which are well within the limits established by 10 CFR 50.46. The results for the 17x17 VANTAGE 5 fuel assembly were based on a $F_Q(z)$ of 2.50 and a Nuclear Enthalpy Rise Hot Channel Factor of 1.65.

Thus, it can be demonstrated that a coolable geometry exists with the postulated occurrence of limited fuel assembly grid deformation on both the 17x17 LOPAR and 17x17 VANTAGE 5 fuel assemblies as defined by 10 CFR 50.46. The presence of limited grid deformation does not result in exceeding any 10 CFR 50.46 criteria. During the transition from 17x17 LOPAR to 17x17 VANTAGE 5, $F_Q(z)$ has been limited to 2.45 for the 17x17 LOPAR fuel and 2.50 for the 17x17 VANTAGE 5 fuel, while the Enthalpy Rise Hot Channel Factor has been limited to 1.62 for 17x17 LOPAR fuel and 1.65 for 17x17 VANTAGE 5 fuel.

7.2.3.2 Small Break LOCA for Transition Cores

The NOTRUMP computer code, Reference 21, was used to model the core hydraulics during a small break event. Only one core flow channel was modeled in the NOTRUMP code, since the core flow during a small break is relatively slow, providing enough time to maintain flow equilibrium between fuel assemblies (i.e., no crossflow). Therefore, hydraulic resistance mismatch is not a factor for small break. Thus, it is not necessary to perform a small break evaluation for transition cores, and it is sufficient to reference the small break LOCA for the complete core of the VANTAGE 5 fuel design as bounding for all transition cycles.

The small break LOCA analysis for the transition from 17x17 LOPAR to 17x17 VANTAGE 5 at the DCPP was performed with the NOTRUMP Evaluation Model. This analysis resulted in a limiting break of 4-inch equivalent diameter with a PCT of 1358°F with an $F_Q(z)$ of 2.50 and an Nuclear Enthalpy Rise Hot Channel Factor of 1.65. The effects of grid deformation were evaluated based on this analysis for a postulated small break LOCA.

The effects of the grid deformation on the small break LOCA analysis calculations were evaluated. The deformation is assumed to occur in the hot assembly which changes the hydraulic resistance of that assembly. This causes a decrease in vapor mass flow rate through the hot assembly (and around the hot rod) and an increase in the temperature of the vapor exiting the assembly. Consequently, the temperature of the fuel cladding increases, particularly at the top of the core where the PCT is calculated to occur.

A conservative evaluation was performed to determine the extent of the decrease in vapor mass flow rate through the hot assembly and the increase in PCT. The equivalent assembly-wide blockage (flow reduction) was determined as the blockage that, when applied to the assembly, resulted in the identical flow reduction seen in the subchannel having the theoretical maximum flow area reduction. The equivalent blockage used in the core coolable geometry analysis for the 17x17 VANTAGE 5 fuel assembly was based on the maximum of four grids deforming.

Therefore, the number of deformed grids postulated to occur on an assembly is four with the location of the grid deformation restricted to occur above the bottom two mixing vane grids and below the top two mixing vane grids. The flow rate reduction near the top of the core through the hot assembly (around the hot rod), based on the 39.3 percent maximum theoretical deformation (area reduction) of four grids, resulted in an estimated increase to the small break PCT of 154°F at the top of the core.

Limited deformation of grids at DCPP Units 1 and 2 has been evaluated for effects on the limiting small break LOCA results. The PCT of the limiting small break has been conservatively estimated to increase by 154°F. This raises the PCT result of 1358°F to an estimated value of 1512°F. This result maintains considerable margin to the 2200°F PCT limit of 10 CFR 50.46 and demonstrates coolable geometry during the transition from 17x17 LOPAR to 17x17 VANTAGE 5. Once a full core of 17x17 VANTAGE 5 fuel has been achieved, the PCT penalty for grid deformation no longer applies since there is no grid deformation.

7.2.4 Containment Integrity Mass and Energy Releases

The containment mass and energy releases, used to determine containment peak pressure, are affected by the fuel design changes as follows:

1. The change in core fluid volume as a result of the new fuel design.

2. The increase or decrease in core stored energy.

3. The effect of the new fuel design on reflood flooding rates as a result of core flow area or hydraulic resistance changes.

The VANTAGE 5 fuel design utilizes a fuel rod of smaller diameter than the 17x17 LOPAR (standard) fuel presently used in DCPP Units 1 and 2. This smaller fuel rod diameter leads to a reduction in core stored energy which is beneficial in reducing the mass and energy releases calculated for a hypothetical LOCA. The smaller VANTAGE 5 fuel rod will slightly increase core fluid volume and the use of IFM grids will increase hydraulic resistance, but these changes are offset by the reduction in core stored energy. Thus, the implementation of VANTAGE 5 fuel at the DCPP will not result in an increase in the containment peak pressure reported in the Diablo Canyon FSAR Update or increase the offsite radiological consequences associated with high containment pressures resulting from a hypothetical LOCA. Based on this evaluation, a reanalysis of containment integrity mass and energy releases was deemed unnecessary for the implementation of VANTAGE 5 fuel at DCPP Units 1 and 2.

The analysis performed for containment integrity does not model a single fuel assembly or fuel assembly subchannel. Only a single channel model of the entire core is employed in this analysis, and this has been deemed sufficient for the purpose of determining LOCA mass and energy releases. The presence of postulated grid deformation will result in a slight increase in core hydraulic resistance which will reduce the rate of mass and energy releases during a hypothetical LOCA. A reduction in the rate of mass and energy releases would be beneficial to the calculation of containment peak pressure and the resulting determination of containment integrity. Thus, the presence of postulated limited grid deformation for the transition from 17x17 LOPAR to 17x17 VANTAGE 5 fuel at DCPP Units 1 and 2 does not result in any design or regulatory limits being exceeded.

The non-LOCA accidents reanalyzed are presented in Attachment 4. All of the Chapter 15 non-LOCA sections are updated in Attachment 4 consistent with these new analyses. There is no significant impact on the steam line break mass and energy release data due to loading VANTAGE 5 fuel. The inside and outside containment steam line break mass and energy release data were evaluated and found to be acceptable. Therefore, no revisions to the current licensing documentation are needed for the steam line break mass and energy accident analyses.

7.2.5 Steam Generator Tube Rupture

The consequences of a Steam Generator Tube Rupture (SGTR), as analyzed in the DCPP FSAR Update, are dependent upon the initial reactor and steam generator conditions of power, pressure, and temperature. The implementation of VANTAGE 5 fuel will not change the initial operating conditions at DCPP Units 1 and 2 and, therefore, the consequences of a SGTR will not be increased by the implementation of VANTAGE 5 fuel. Thus, a reanalysis of the DCPP SGTR accident was determined to be unnecessary for the implementation of VANTAGE 5 fuel, and the current SGTR analysis presented in the FSAR Update was considered to be bounding.

The SGTR accident is analyzed to calculate the offsite doses, which should remain below the 10 CFR 100 guidelines. The primary thermal hydraulic parameters affecting this conclusion are the extent of fuel failure, the primary to secondary flow through the ruptured tube, and the mass released to the atmosphere. The amount of fuel failure assumed for the DCPP FSAR Update SGTR analysis was 1 percent, which was assumed to be independent of the transient conditions. The assumed RCS coolant activities based on 1 percent fuel failure are more conservative than the Technical Specification limits. The primary to secondary break flow and the mass released to the atmosphere are primarily dependent upon RCS and secondary system thermal hydraulic parameters. Since postulation of limited grid deformation in the 17x17 LOPAR or 17x17 VANTAGE 5 fuel during transition will not change the RCS and secondary thermal hydraulic design parameters, there will be no effect on the DCPP Units 1 and 2 SGTR analysis.

A revised SGTR analysis was recently performed for DCPP Units 1 and 2 using the current NRC approved SGTR analysis methodology. The revised SGTR analysis is presented in Reference 30 and was submitted to the NRC via Reference 32. The analysis was performed assuming the most limiting parameters associated with LOPAR or VANTAGE 5 fuel such that the postulation of limited grid deformation does not affect the revised SGTR analysis for DCPP Units 1 and 2.

7.2.6 Blowdown Reactor Vessel and Loop Forces

The forces created by a hypothetical break in the RCS piping are principally caused by the motion of the decompression wave through the RCS. The strength of the decompression wave is primarily a result of the assumed break opening time, break area, and RCS operating conditions of power, temperature, and pressure. These parameters will not be affected by a change in fuel at DCPP Units 1 and 2 from 17x17 LOPAR to VANTAGE 5. The forces in the vicinity of the core are affected by the core flow area/volume. An increase in core flow area/volume will tend to more effectively dissipate the decompression wave resulting in a reduction of the forces acting on the reactor vessel internals. VANTAGE 5 fuel, having a smaller rod diameter than 17x17 LOPAR fuel, increases the core flow area and volume which is beneficial in reducing forces associated with a hypothesized LOCA. Forces acting on the RCS loop piping as a result of a hypothesized LOCA are not influenced by changes in fuel assembly design. Thus, the implementation of VANTAGE 5 fuel at Units 1 and 2 will not result in an increase of the calculated consequences of a hypothesized LOCA on the reactor vessel internals or RCS loop piping. The current FSAR Update analysis for forces on the reactor internals and RCS

piping resulting from a hypothesized LOCA are considered to be bounding to the application of VANTAGE 5 fuel at the DCPP Units 1 and 2.

The LOCA hydraulic forcing function analysis does not model either a single fuel subchannel or a single fuel assembly. Instead, a single channel model of the entire core is used in the analyses performed to generate the forces on reactor internal components. Since the primary physical mechanism for creating hydraulic loads is the propagation of a pressure wave directly from the vessel side of the break to the vessel internals, a detailed core model is not required. The peak hydraulic forces are calculated to occur within 0.5 seconds of initiation of the break and, thus, prediction of flow conditions in the core based on core physical characteristics does not participate in determining the magnitude of the forces. The slight increase in core hydraulic resistance or the localized reduction in flow area as a result of postulated grid deformation will not increase the consequences of the LOCA loads on vessel internals. The requirements of GDC-4 will still be satisfied for the DCPP Units 1 and 2 should the 17x17 LOPAR or VANTAGE 5 fuel sustain limited grid deformation during transition.

Forces created on the RCS piping as a result of the postulated break in the piping attain their peak within 0.25 seconds after the accident. The forces on the RCS loops are strongly influenced by the local conditions (i.e. pressure and temperature) existing at the break location at the time of the accident. Minor modifications to reactor internals which do not affect the steady state operating pressure or temperature will not influence the forces on the RCS piping resulting from a hypothesized LOCA. Thus, postulated limited deformation of the grids in 17x17 LOPAR or VANTAGE 5 fuel during transition at DCPP Units 1 and 2 will not result in an increase in the calculated hydraulic forcing functions on the RCS piping.

7.2.7 Post-LOCA Long-Term Core Cooling (ECCS Flows, Core Subcriticality, and Switchover of the ECCS to Hot Leg Recirculation)

The implementation of VANTAGE 5 fuel at DCPP Units 1 and 2 does not impact the assumptions for decay heat, core reactivity, or boron concentration for sources of water residing in the containment sump Post-LOCA. Thus, these licensing requirements associated with LOCA are not significantly affected by the implementation of VANTAGE 5 fuel. Additionally, an independent check on core subcriticality is performed for each fuel cycle operated at DCPP Units 1 and 2.

Calculations performed by Westinghouse to determine the post-LOCA sump boron concentration include the water volume internal to the core and are slightly sensitive to changes in fuel. The change in the fuel volume due to postulated limited grid deformation is clearly insignificant when compared with the water volumes of the RCS, RWST, accumulators, and piping associated with the pumped ECCS. The presence of deformed grids, postulated to occur when seismic and LOCA loads are combined, will not reduce the sump post-LOCA boron concentrations following a hypothetical large break LOCA. Thus, the ability to satisfy the post-LOCA long term core cooling requirement will not be affected by postulated limited grid deformation of either the 17x17 LOPAR or 17x17 VANTAGE 5 fuel during transition.

Post-LOCA hot leg recirculation switchover time is determined for inclusion in emergency procedures to ensure no boron precipitation in the reactor vessel following boiling in the core. This time is dependent on power level, and the RCS, RWST, and accumulator water volumes with their associated boron concentrations. The postulated presence of limited grid deformation does not affect the power level or the maximum boron concentrations or volumes assumed for the RCS, RWST, and accumulators. Thus, postulated limited grid deformation of 17x17 LOPAR or 17x17 VANTAGE 5 fuel as a result of combined seismic and LOCA loads at the DCPP Units 1 and 2 does not affect the calculated post-LOCA hot leg switchover time.

7.2.8 Conclusion

The effect of FSAR LOCA-related accidents as a result of postulated limited grid deformation on the 17x17 LOPAR or 17x17 VANTAGE 5 fuel at the DCPP Units 1 and 2 has been evaluated. A specific large break LOCA core coolable geometry analysis was performed for both 17x17 LOPAR and 17x17 VANTAGE 5 and the results showed that postulated grid deformation does not result in exceeding any 10 CFR 50.46 criteria. The potential effect on the calculated consequences for each remaining FSAR Update LOCA related accident was addressed and, in all cases, the evaluation judged that postulated limited grid deformation did not result in any design or regulatory limit being exceeded. Thus, based on this evaluation, there are no LOCA accidents or related considerations that are adversely affected by postulated limited grid deformation for either the 17x17 LOPAR or 17x17 VANTAGE 5 fuel.

8.0 RADIOLOGICAL CONSEQUENCES

8.1 Radiological Consequences of Reanalyzed Accidents

An evaluation of the radiological consequences of the transition to VANTAGE 5 fuel for DCPP Units 1 and 2 was performed. Three accidents were identified as having changes in results which impact the calculation of offsite doses in Section 16.5 of the FSAR Update. These accidents are: (1) Loss of Load, (2) Locked Rotor, and (3) Steamline Break.

8.1.1 Loss of Load

The atmospheric steam releases for the loss of load accident were determined to be larger than for station blackout and locked rotor accidents using current methodology, which considers pump heat in the calculation of steam release. The loss of load accident replaces the loss of offsite power to the station auxiliaries (station blackout) as the limiting accident for radiological consequences in FSAR Update Section 15.5.10. This section refers to Condition II Faults requiring atmospheric steam releases with insignificant core damage. The changes in steam release data for post-accident time periods are shown below.

Condition II Fault Steam Releases (lbm)

	0 - 2 Hours	2 - 8 Hours
FSAR Update Section 15.5.10 (Blackout)	515,000	1,100,000
New Steam Releases (Loss of Load)	656,000	964,000

The new steam releases result in the following changes in offsite doses for the design basis case for Condition II faults. The changes are as follows:

Condition II Fault Offsite Doses (Rem)

Offsite Location	FSAR Update Table 15.5-9		New Offsite Doses	
	Thyroid	Whole Body	Thyroid	Whole Body
Site Boundary	0.022	0.0018	0.028	0.0023
Low Population Zone	0.0066	0.00022	0.0067	0.00022

8.1.2 Locked Rotor

Based on the reanalysis of this accident, the number of fuel rods calculated to be in DNB is less than 10.0 percent. The current FSAR Update design basis analysis in Section 15.5.21 is based on 3.1 percent. The value of 10.0 percent and the new steam releases presented above, which bound the Locked Rotor event, result in the following changes in offsite doses for the design basis case.

Locked Rotor Accident Offsite Doses (Rem)

Offsite Location	FSAR Update Table 15.5-42		New Offsite Doses	
	Thyroid	Whole Body	Thyroid	Whole Body
Site Boundary	0.082	0.0044	0.30	0.013
Low Population Zone	0.027	0.00048	0.078	0.0011

8.1.3 Steamline Break

For this accident, new steam releases were determined based on the hot zero power inventory. The changes in steam release data for the post-accident time periods are shown below compared to current FSAR Update values.

Steamline Break Accident Steam Releases (lbm)

	0 - 2 Hours	2 - 8 Hours
FSAR Update Table 15.5-34, Ruptured Line	97,000	0
New Steam Releases, Ruptured Lines	162,784	0
FSAR Update Table 15.5-34, Intact Lines	520,000	1,100,000
New Steam Releases, Intact Lines	393,464	860,461

The new steam releases result in the following changes in offsite doses for the design basis case for the Steamline Break accident.

Steamline Break Accident Offsite Doses (Rem)

Offsite Location	FSAR Update Table 15.5-36		New Offsite Doses	
	Thyroid	Whole Body	Thyroid	Whole Body
Site Boundary	0.065	0.0018	0.088	0.0017
Low Population Zone	0.066	0.00032	0.074	0.00030

8.1.4 Conclusions

The radiological consequences for the three accidents analyzed are much less than the 10 CFR 100 regulatory guideline values of 300 rem thyroid and 25 rem whole body.

8.2 Extended Fuel Burnup Effects

The change from LOPAR fuel to VANTAGE 5 fuel for DCPP Units 1 and 2 does not involve any changes that would potentially impact the source terms utilized in the determination of radiological consequences of accidents as discussed in the FSAR Update.

The only characteristic of VANTAGE 5 fuel that would normally be reviewed for impact on the source terms is the implementation of increased fuel cycle burnup, which is generally associated with the increase in cycle length and increase in fuel enrichment. The change to VANTAGE 5 fuel for DCPP does not

involve an increase in fuel burnup since the increase in fuel burnup was incorporated previously in Cycle 3. VANTAGE 5 fuel and LOPAR fuel under the same conditions of extended fuel burnup would not have significantly different source terms for the calculation of radiological consequences (Reference 28).

The potential effects of reload fuel having an initial enrichment up to 4.5 weight percent U-235 and extended burnup to 50,000 MWD/MTU were previously evaluated for DCPP. The radiological consequences were analyzed for representative accidents and are presented in the FSAR Update Section 15.5.17.5.1 for the design basis LOCA and in Section 15.5.22.1.1 for the design basis fuel handling accident. All results were within the 10 CFR 100 values.

8.3 Steam Generator Tube Uncovery Effects

Additional analyses were performed to evaluate the radiological consequences of steam generator tube uncovery (Reference 31). The analyses addressed the potential increase in radioactivity release to the environment due to the possible uncovering of steam generator tubes. The accidents affected were Steam Generator Tube Rupture (SGTR) and other accidents having secondary system releases to the atmosphere.

PG&E addressed this concern in the DCPP analysis of the radiological consequences for an SGTR event (Reference 32).

Accidents other than the SGTR, including the three presented in this submittal, were also analyzed. To be conservative, no credit for the retention of iodine was used in these analyses for the entire accident period. The resultant doses using this conservative approach for accidents other than the SGTR did not exceed a small fraction (i.e. 10 percent) of the 10 CFR 100 values.

The Westinghouse Owners Group is sponsoring a program to study in more detail the effects of steam generator tube uncovery. The preliminary results suggest the radiological impact of steam generator tube uncovery is small. The final results are expected to be issued by April 1989.

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