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

ENCLOSURE 1

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING CYCLE 2 OPERATIONS

LOUISIANA POWER AND LIGHT COMPANY

WATERFORD STEAM ELECTRIC STATION, UNIT 3

DOCKET NO.: 50-382

1.0 INTRODUCTION

By letters dated August 29, 1986 and October 1, 1986 (Ref. 1, 2), Louisiana Power and Light Company (LPL), the licensee, submitted Parts A and B of a reload safety analysis report in support of a request to reload and operate Waterford Unit 3 for a second cycle at 100% of the rated core power of 3390 MWt. The licensee also submitted proposed modifications to the Technical Specifications for Cycle 2 which have been reviewed by the staff in separate safety evaluations.

The NRC staff has reviewed the application and the supporting documents and has prepared the following evaluation of the fuel design, nuclear design, and thermal-hydraulic design of the core as well as an evaluation of those plant transients and accidents which were reanalyzed for Cycle 2.

2.0 EVALUATION OF FUEL DESIGN

2.1 Mechanical Design

The Cycle 2 core consists of 217 fuel assemblies. Ninety-two fresh (unirradiated) Batch D assemblies will replace 73 Batch A assemblies and 19 Batch B assemblies. The remaining 61 Batch B assemblies and all Batch C assemblies in the core during Cycle 1 will be retained. The 92 Batch D assemblies will consist of 24 type D0 assemblies with 3.90 weight percent (w/o) and 3.40 w/o U-235 enriched fuel rods, 12 type D1 assemblies with 3.90 w/o and 3.40 w/o U-235 enriched rods and four burnable poison shims per assembly, 24 type D2 assemblies with 3.40 w/o and 2.78 w/o U-235 enriched rods and four burnable poison shims per assembly, and 32 type D3 assemblies with 3.40 w/o and 2.78 w/o U-235 enriched rods and eight burnable poison shims per assembly. Except for the design features listed below, the mechanical design of the Batch D assemblies is identical to that of the Cycle 1 fuel assemblies. The changes are:

a. The lowest spacer grid (Inconel) has a redesigned perimeter strip with lead-in tabs that have been changed from trapezoidal-shaped to curve-shaped. This change is intended to improve fuel handling by reducing the chance of grid hangup and is, therefore, acceptable.

b. The fuel rod overall length has been reduced by 0.15 inches and the guide tubes have been lengthened by 0.9 inches. This results in additional shoulder gap clearance. The licensee has stated that these changes do not result in the violation of any design criteria. Therefore, the changes are acceptable.

c. The control element assembly (CEA) guide tube wear sleeve modification made to the Batch D reload fuel has been reviewed and approved by the NRC (Ref. 3).

The licensee has also evaluated the criticality effects of storage of the higher enriched Batch D fuel assemblies in the Waterford 3 fuel storage facilities and has shown that the NRC acceptance criterion of k_{eff} less than or equal to 0.95 is met for all normal and abnormal conditions (Ref. 4). The staff, therefore, concludes that the Batch D fuel assemblies are acceptable for use during Cycle 2.

Attachment 5 to Reference 5 is a report, entitled "Evaluation of Interpellet Gap Formation and Clad Collapse in Modern PWR Fuel Rods," on work performed by Combustion Engineering (CE) for Electric Power Research Institute (EPRI). The report presents the results from a review of interpellet-gap formation, ovality, creepdown and clad collapse data in modern PWR fuel rods. Based on these results, CE has reformulated its creep-collapse predictor, CEPAN, to treat finite gaps and reanalyzed the flux augmentation (spiking) factor to take account of gap formation statistics data from modern fuel.

The data obtained by CE from measurements on its own fuel and the published results of such measurements for fuel from other PWR vendors was examined to obtain information on the number, axial distribution, and size of densification induced gaps formed in PWR fuel rods. Data were obtained on old fuel (unpressurized rods containing densifying fuel), intermediate fuel (pressurized but with densifying fuel) and modern fuel (pressurized rods containing non-densifying fuel). Densifying fuel is that which increases in density by about 3 percent when resintered in-reactor. Non-densifying fuel shows a density increase of less than 0.5 percent upon resintering.

The report concludes that, in modern CE fuels, the maximum gap size is less than 0.025 inches and that gaps are distributed uniformly along the fuel length. Gap density (number per unit length) and gap size are not a function of core height. Accordingly, the time to clad collapse and the flux augmentation factors have been recalculated.

It is concluded that modern CE fuel rods have a time to clad collapse far in excess of any practical residence time. It is further concluded that the maximum augmentation factor is 1.001 for gaps less than 0.025 inches, which is insignificant with respect to other power distribution uncertainties.

The staff concurs with the conclusions of the CE report as it applies to Waterford 3, Cycle 2. This concurrence is supported by similar results of analyses by another fuel vendor. Therefore, the staff concludes that no further analysis of clad collapse need be performed for Cycle 2.

License Condition 2.C.7 requires justification that sufficient shoulder gap clearance will be available in all existing fuel assemblies to be irradiated in the next cycle of operation (Cycle 2). Pursuant to this requirement, the licensee submitted the Waterford 3, Cycle 2, Shoulder Gap Evaluation report, CEN-335(C)-P, dated July 1986 (Ref. 6). The evaluation presented in this report was based on an empirical evaluation of data obtained from Arkansas Nuclear One, Unit 2 (ANO-2) and San Onofre Nuclear Generating Station, Unit 2 (SONGS-2). The staff has reviewed this report and concludes that the shoulder gaps in all the fuel are acceptable through Cycle 2 and License Condition 2.C.7, therefore, is satisfied. Details of the staff's review have been provided in a separate safety evaluation transmitted to the licensee by letter dated December 9, 1986.

2.2 Thermal Design

The thermal performance of Cycle 2 fuel was performed by analyzing a composite fuel pin that envelops the peak pins of the various fuel assemblies (fuel Batches B, C, and D) in the Cycle 2 core using the NRC approved fuel performance code FATES3A. The NRC imposed grain size restriction (Ref. 7) was included and a power history that envelops the power and burnup levels representative of the peak pin at each burnup interval from beginning-of-cycle (BOC) to end-of-cycle (EOC) was used. The maximum peak pin burnup analyzed for Cycle 2 bounds the expected EOC maximum fuel rod burnup. Based on this analysis, the internal pressure in the most limiting hot rod will not reach the nominal reactor coolant system (RCS) pressure of 2250 psia. Since this satisfies the fuel rod internal gas pressure requirement of Standard Review Plan (SRP) 4.2, Section II.A.1(f), the staff finds it acceptable and concludes that the fuel rod internal pressure limits have been adequately considered for Cycle 2 operation.

3.0 EVALUATION OF NUCLEAR DESIGN

3.1 Fuel Management

The Waterford 3 Cycle 2 core consists of 217 fuel assemblies, each having a 16 by 16 fuel rod array. A general description of the core loading is given in Section 2.1 of this SER. The highest U-235 enrichment occurs in the Batch D fuel assemblies which contain fuel rods with 3.9 weight percent U-235. The Waterford 3 fuel storage facilities have been approved for storage of fuel of maximum U-235 enrichment of 3.9 weight percent (Ref. 4).

The Cycle 2 core will use a low-leakage fuel management scheme in which the previously irradiated Batch B and C assemblies are placed on the core periphery. Most of the fresh Batch D assemblies are placed in the interior of the core and mixed with the previously irradiated fuel to min imize power peaking. With this loading and a Cycle 1 endpoint of 13,800 MWD/MTU, the Cycle 2 reactivity lifetime for full power operation is expected to be 14,750 MWD/MTU. The analyses presented by the licensee will accommodate a Cycle 2 length of up to 16,000 MWD/MTU and is applicable for Cycle 1 termination burnups of between 13,400 and 14,400 MWD/MTU.

3.2 Power Distributions

Hot full power (HFP) fuel assembly relative power densities are given in Reference 1 for BOC, middle-of-cycle (MOC), and EOC unrodded configurations. Radial power distributions at BOC and EOC are also given for rodded configurations allowed by the power dependent insertion limit (PDIL) at full power. These rodded configurations consist of part length CEAs (PLCEAs), Bank 6, and Bank 6 plus the PLCEAs. The largest radial power peak occurs at BOC for both the rodded and unrodded configurations. These expected values are based on ROCS code calculations with neutron cross sections generated by the DIT code (Ref. 8). Also, the use of ROCS and DIT with the MC fine-mesh module explicitly accounts for the higher power peaking which is characteristic of fuel rods adjacent to water holes. These methods have been approved by the NRC and, therefore, the calculated power distributions are acceptable.

3.3 Control Requirements

The value of the required shutdown margin varies throughout core life with the most restrictive value occurring at EOC hot zero power (HZP) conditions. This minimum shutdown margin of 5.15% delta k/k is required to control the reactivity transient resulting from the RCS cooldown associated with a steam line break accident at these conditions. For operating temperatures below 200°F, the reactivity transients resulting from any postulated accident are minimal and a 2.0% delta k/k shutdown margin provides adequate protection. Sufficient boration capability and net available CEA worth, including a maximum worth stuck CEA and appropriate calculational uncertainties, exist to meet these shutdown margin requirements. These results were derived by approved methods and incorporate appropriate assumptions and are, therefore, acceptable.

During the Cycle 2 reload review, the licensee submitted a request to modify the Technical Specifications governing shutdown margin when all CEAs are verified to be fully inserted in the core. In this case, the required shutdown margin and hence, the required RCS boration, can be reduced since the assumption of a stuck CEA is not made. However, the shutdown margin still accounts for the CEA of highest worth being out of the core. For example, if a CEA ejection event occurred, the core would still be subcritical by at least the amount required by the shutdown margin. This change will result in a significant savings in time and in the processing of waste water. The staff reviewed and approved the change to the Technical Specifications governing shutdown margin when all CEAs are fully inserted into the core in Amendment 11, issued by letter dated January 9, 1987.

3.4 Augmentation Factors

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

4.0 EVALUATION OF THERMAL-HYDRAULIC DESIGN

Steady-state thermal-hydraulic analysis for Cycle 2 is performed using the approved thermal-hydraulic code TORC (Ref. 9) and the CE-1 critical heat flux (CHF) correlation (Ref. 10). The core and hot channel are modeled with the approved method described in Ref. 11. The design thermal margin analysis is performed with the fast running variation of the TORC code, CETOP-D (Ref. 12). The licensee has verified that the CETOP-D model predicts minimum departure from nucleate boiling ratio (DNBR) conservatively relative to TORC.

The uncertainties associated with the system parameters are combined statistically using the approved statistical combination of uncertainties (SCU) methodology described in Refs. 13, 14, and 15. Using this SCU methodology, the engineering hot channel factors for heat flux, heat input, fuel rod pitch, and cladding diameter are combined statistically with other uncertainly factors to arrive at an equivalent DNBR limit of 1.26 at a 95/95 probability/confidence level. The fuel rod bow penalty is incorporated directly in the DNBR limit. It has been calculated using the approved method described in Ref. 16. The value used for this analysis, 1.75% DNBR, is valid for bundle burnups up to 30,000 MWD/MTU. For those assemblies with average burnup in excess of 30,000 MWD/MTU, sufficient margin exists to offset rod bow penalties.

5.0 EVALUATION OF SAFETY ANALYSES

The design basis events (DBEs) considered in the safety analyses are categorized into two groups: anticipated operational occurrences (AOOs) and postulated accidents. All events were reviewed by the licensee to assess the need for reanalysis as a result of the new core configuration for Cycle 2. Those events for which results were not bounded by the FSAR were reanalyzed by the licensee to assure that the applicable criteria are met. The AOOs were analyzed to assure that specified acceptable fuel design limits (SAFDLs) on DNBR and fuel centerline to melt (CTM) are not exceeded. This may require either reactor protection system (RPS) trips or RPS trips and/or sufficient initial steady state margin to prevent exceeding the SAFDLs.

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

5.1 Steam System Piping Failures Inside and Outside of Containment

Steam line breaks (SLBs) inside containment may have break areas up to the cross section of the largest main steam pipe (7.88 ft^2). The licensee performed a parametric analysis in both MTC and break area to determine the limiting inside containment SLB event in terms of fuel pin failure caused by the pre-trip power excursion. Since inside containment SLBs may cause environmental degradation of sensor input to core protection calculators (CPCs) and pressure measurement systems, the only credit taken for CPC action during this event is the CPC variable overpower trip (VOPT). The required input to the VOPT includes output from the resistance temperature detectors (RTDs) and the excore neutron flux detectors. These sensors have been qualified in degraded environmental conditions for a sufficient length of time to allow their use in providing input for VOPT action for this event. In addition to the VOPT, the low steam generator pressure (LSGP) trip is credited and the environmentally degraded value of the delta pressure low flow trip is used to determine the most adverse timing of a loss of offsite AC power. The results indicate that less than 4% of the fuel rods experience a DNBR below the 1.26 SAFDL and are assumed to experience cladding failure.

Break areas for outside containment SLBs are limited to the area of the main steam flow venturies (3.14 ft^2) located upstream of the containment penetrations. The outside containment SLBs, however, are not subject to the same environmental effects on the RPS as the inside containment breaks and the full array of RPS trips, including the CPC low DNBR trip, can be credited. Fewer than 1.5% of the fuel pins were predicted to fail for the outside containment SLB. However, although fuel failures are less, the limiting break in terms of radiological consequences is located outside of the containment building. This is because a steam generator iodine decontamination factor of 100, as specified in Standard Review Plan (SRP) 15.1.5, was used in the dose calculation for the inside containment SLB whereas a factor of unity was used for the outside containment SLB. The site boundary doses are a small fraction of 10 CFR 100 limits for a coincident iodine spike. For a pre-existing iodine spike or for the predicted fuel failure of less than 1.5%, the resultant doses are within the 10 CFR 100 limits. The SLB post-trip return to power was also reanalyzed due to the more adverse moderator cooldown curve and the increased inverse boron worth. The results show that no fuel failure occurs and a coolable geometry is maintained. The licensee has demonstrated conformance with the acceptance criteria stipulated SRP Section 15.1.5. As such, the staff concludes that Cycle 2 operation is acceptable with respect to accidents resulting in breaks in the steam line.

5.2 Total Loss of Forced Reactor Coolant Flow

The loss of coolant flow (LOF) event was reanalyzed by the licensee due to the change in the CPC trip. Rather than using the low DNBR trip, the LOF event for Cycle 2 was analyzed with a CPC trip based on low reactor coolant pump (RCP) shaft speed, initiated when the shaft speed drops to 96.5% of its initial speed. The results show that this event initiated from the Technical Specification LCOs in conjunction with the low RCP shaft speed trip will not exceed the DNBR limit and the maximum pressure of the reactor coolant and main steam systems will not exceed 110% of the design pressure. The acceptance criteria stated in SRP Section 15.3.1, therefore, are met and the staff concludes that the LOF event for Cycle 2 is acceptable.

5.3 Single Reactor Coolant Pump Sheared Shaft

The single reactor coolant pump sheared shaft was reanalyzed due to a change in the fuel failure pin census and the CEA worth at the point of reactor trip. A reactor trip was assumed to occur when the rapid flow reduction across the steam generator in the affected loop decreases the delta-pressure below the trip setpoint. The reactor trip produces an automatic turbine trip following which AC power is assumed to be unavailable. The loss of off-site AC power results in the coastdown of the remaining three pumps, further decreasing the reactor coolant system flow.

The amount of fuel failure calculated for Cycle 2 was less than 8.5%. The statistical convolution method (Ref. 17) was used to determine the amount of fuel failure. This method was previously reviewed by the staff and found to be acceptable for analysis of the sheared shaft event. The resultant doses were less than 30 REM thyroid and less than 2.5 REM whole body which are a small fraction of 10 CFR 100 guidelines. Additionally, the peak pressure is less than 2750 psia. The event, therefore, conforms to the requirements of SRP Section 15.3.3 and is acceptable.

5.4 Uncontrolled CEA Withdrawal from a Subcritical or Low Power Condition

The uncontrolled CEA withdrawal event from a subcritical or low power condition was reanalyzed due to an increase in the subcritical reactivity insertion rate and, for the event initiated from low power, the addition of the CPC VOPT. The events are analyzed to ensure that the DNBR and the CTM SAFDLs are not violated and to verify that the peak RCS pressure is less than the design limit of 2750 psia.

The CEA withdrawal from subcritical conditions resulted in a reactor trip on high logarithmic power with a minimum DNBR greater than the design limit of 1.26. The peak linear heat generation rate (PLHGR) was predicted to be in excess of the steady state centerline melt limit of 21 kW/ft. Since this transient value of PLHGR exceeded the steady state limit, an assessment of the resultant fuel centerline temperature was performed by the licensee based on the maximum centerline enthalpy of the fuel. The calculation assumed that no heat is transferred away from the centerline during the transient. The total enthalpy was calculated to be 103 cal/gm. The temperature corresponding to this enthalpy is 2590°F, which is well below the UO_2 melting point of 4900°F. Additionally, the peak RCS pressure is less than the design limit of 2750 psia.

For the CEA withdrawal from low power, a VOPT is generated when the power reaches 40% of full power. The results indicate that the DNBR, CTM, and RCS pressure limits will not be exceeded during the event.

The staff, therefore, concludes that Cycle 2 meets the requirements of SRP Section 15.4.1 governing CEA withdrawal events from a subcritical or low power condition.

5.5 Single Full Length CEA Drop

The single full length CEA drop event was reanalyzed to determine the initial thermal margin that must be maintained by the limiting conditions of operation (LCOs) such that the DNBR and CTM will not be violated. Since the CEA position-related penalty factors for downward single CEA deviations have been set equal to unity (no penalty) as part of the CPC improvement program, a reactor trip is not generated for a single CEA drop and, therefore, the expected margin degradation for the event is accounted for by reserving sufficient margin in the LCOs. For CEA subgroup drops, the CEA position-related penalty factors for downward deviations are still used by the CPC, as in Cycle 1, to provide a reactor trip when necessary. The event was initiated by dropping a full length

CEA over a period of one second. The turbine load was not reduced, resulting in a power mismatch between the primary and secondary systems, which leads to a cooldown of the RCS. The largest change in power peaking was obtained by examining drops involving different individual CEAs into the radial rodded configurations allowed by the PDIL transient insertion limit Technical Specification figure. This resulted in a radial peaking factor increase of 9.0%.

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

A minimum DNBR of greater than 1.26 was obtained after 900 seconds as determined from the 9.0% radial power peaking increase following the CEA drop plus 15 minutes of xenon redistribution at the final coolant conditions. If the dropped CEA has not been realigned by then, the operator will take action to reduce power in accordance with Figure 3.1-1A of the Technical Specifications. A maximum allowable initial LHR of 17.0 kW/ft could exist as an initial condition without exceeding the acceptable fuel centerline melt limit of 21.0 kW/ft during the transient. This amount of margin is assured since the LHR LCO is based on the more limiting allowable LHR for the loss-of-coolant-accident (LOCA) of 13.4 kW/ft.

The staff, therefore, concludes that Cycle 2 meets the requirements of SRP Section 15.4.3 governing control rod misoperation.

5.6 Inadvertent Boron Dilution

This event was reanalyzed due to the Cycle 2 increase in critical boron concentrations. For power operation (Modes 1 and 2), an inadvertent boron dilution event will be no more severe for cycle 2 than for Cycle 1. For sub-critical modes (Modes 3 through 6), the time required to achieve criticality due to boron dilution depends on the initial and critical boron concentrations as well as the inverse boron worth and the rate of dilution. The results show that sufficient time exists to alert the operator of a boron dilution event at least 15 minutes before criticality (30 minutes during refueling) during all modes of Cycle 2 operation, even in the absence of a boron dilution alarm. The staff concludes that Cycle 2 meets the requirements of SRP Section 15.4.6 and is acceptable with respect to inadvertent boron dilution events.

5.7 Asymmetric Steam Generator Events

The four events which affect a single steam generator are the loss of load to one steam generator (LL/1SG), the excess load to one steam generator (EL/1SG), the loss of feedwater to one steam generator (LF/1SG), and the excess feedwater to one steam generator (EF/1SG). Of these, the LL/1SG event is the limiting asymmetric event. This event is initiated by the inadvertent closure of a single main steam isolation valve (MSIV), which results in a loss of load to the affected steam generator. The

CPC high differential cold leg temperature trip serves as the primary means of mitigating this transient with the steam generator low level trip providing an additional protection. The minimum transient DNBR calculated was greater than the DNBR SAFDL limit of 1.26. A maximum allowable LHR of 17.0 kW/ft could exist as an initial condition without exceeding the fuel CTM SAFDL of 21.0 kW/ft during the transient. This amount of margin is assured by setting the LHR LCO based on the more limiting allowable LHR for LOCA of 13.4 kW/ft. The staff concludes that the calculations contain sufficient conservatism to assure that fuel damage will not result from any asymmetric steam generator event during Cycle 2 operation.

A methodology change from the reference cycle analysis of this event is the application of the HERMITE computer code to model both the effect of the temperature tilt on radial power distribution and the space-time impact of the CEA scram. HERMITE has been approved for licensing applications (Ref. 18) and uses the core parameters generated by the CESEC III code (core flow, RCS inlet temperature, RCS pressure, and reactor trip time) as input to simulate the core in two dimensions. The staff finds this improved modeling technique acceptable.

5.8 Loss of Coolant Accident (LOCA)

The emergency core cooling system (ECCS) performance evaluation for both the large break and the small break LOCA must show conformance with the acceptance criteria required by 10 CFR 50.46. A Waterford 3 specific analysis was performed for Cycle 2, primarily to implement a new axial power shape (Ref. 19), update the containment heat sink and containment free volume data, and employ the CE flow blockage model (Ref. 20). Except for these factors and the assumption of no single failure, the Cycle 2 methodology is the same as that used for Cycle 1. The staff requested confirmation that the assumption of no single failure is the worst case in terms of peak clad temperature. In response, the licensee stated that an assumption of no single failure (i.e., no low pressure safety injection (LPSI) pump failures) yields a higher peak clad temperature because of the lower containment back pressure caused by increased spillage from the safety injection system.

The large break LOCA analysis was performed for the 0.8 double-ended guillotine at pump discharge (DEG/PD) break and resulted in a peak clad temperature of 2150°F, a peak local clad oxidation percentage of 7.8% and a peak core wide clad oxidation percentage of less than 0.80%. These results meet the 10 CFR 50.46 acceptance criteria for peak clad temperature (2200°F), peak local clad oxidation percentage (17.0%), and core wide clad oxidation percentage (1.0%). The staff, however, requested verification that the assumption of no single failure, a different axial power shape, revised containment data and the new flow blockage model did not require a new break sensitivity study per Appendix K to 10 CFR 50. Although the licensee stated their belief that the break analyzed would remain the most limiting break (Ref. 32), they have committed to reanalyzing the large break LOCA (including a new break spectrum) (Ref. 33).

The staff concludes that the information provided in References 32 and 33 provides sufficient justification for relying upon the previous break spectrum analyses until May 1987 as stated in Reference 33.

6.0 CPC/CEAC SOFTWARE MODIFICATIONS

The Waterford 3 CPC system is provided by the reactor vendor, CE. The system is designed to provide the necessary reactor trips (low DNBR and high local power density) to ensure that the SAFDLs on DNBR and CTM are not exceeded during AOOs. The CPC system is also designed to aid in limiting the consequences of certain postulated accidents.

The CPC software for Cycle 2 operation is an updated version of the CE CPC software which has been previously approved for use in CESSAR 80 plants. By letters dated August 30, 1985 (Ref. 21), October 18, 1985 (Ref. 22), February 10, 1986 (Ref. 23), April 8, 1986 (Ref. 24), and May 21, 1986 (Ref. 25), CE submitted documents which describe additional CPC software modifications, algorithm change procedures, and reload data block (RDB) constants applicable to Waterford 3 as well as to SONGS 2 and 3, ANO-2, and Palo Verde 1, 2, and 3, and are intended to be implemented at each plant at the appropriate time. These documents have been reviewed and approved by the staff (Refs. 26, 27, 28, 29, and 30) and have been implemented at Waterford 3.

7.0 STATISTICAL COMBINATION OF UNCERTAINTIES (SCU)

In Cycle 1, the uncertainties of relevant parameters were treated deterministically. As mentioned in Section 4.0, Cycle 2 will statistically combine these uncertainties to generate a new DNBR limit of 1.26, which is applied to the safety analyses, the CPC trip setpoints and the COLSS required overpower margin calculations. The SCU analysis performed for Waterford 3, Cycle 2 demonstrates that there will be at least a 95% probability with at least 95% confidence (95/95 probability/confidence) that the limiting fuel pin will avoid departure from nucleate boiling (DNB) as long as the minimum DNBR found with the best estimate design CETOP-D model remains at or above 1.26. The system parameter uncertainties included in the Cycle 2 SCU analysis are:

1. Inlet flow distribution uncertainties
2. Enthalpy rise factor
3. Systematic pitch uncertainties
4. Systematic clad outer diameter uncertainties
5. Heat flux factor
6. CE-I CHF correlation uncertainties
7. TORC Code uncertainty
8. Fuel rod bow penalty on DNBR
9. HID-1 grid penalty.

The first seven system parameter uncertainties were combined statistically whereas the last two were applied deterministically.

SCU will also be applied to the determination of LCOs and limiting safety system settings (LSSS) on linear heat rate (LHR) and DNBR beginning with Cycle 2. The methodology used for Waterford 3 is similar to that used for SONGS Units 2 and 3 (Refs. 13, 14, and 15). Minor changes were made to the SCU methodology as part of the CPC improvement program but involve no substantial difference in basic methodology or results (Ref. 31). The staff, therefore, finds the application of SCU to Cycle 2 of Waterford 3 acceptable.

8.0 TECHNICAL SPECIFICATION CHANGES

The staff has reviewed the proposed modifications to the Technical Specifications for Cycle 2 submitted by the licensee. These changes which have been reviewed in separate safety evaluations.

9.0 EVALUATION FINDINGS

The staff has reviewed the fuels, physics and thermal-hydraulics information presented in the Waterford 3 Cycle 2 reload report. Based on the evaluations given in the preceding sections, the staff finds the proposed reload report acceptable.

Principal Contributor: L. Kopp

Dated: January 16, 1987

10.0 REFERENCES

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6. "Waterford Unit 3, Cycle 2 Shoulder Gap Evaluation," CEN-335(C)-P, Combustion Engineering, July 1986.

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15. "Statistical Combination of Uncertainties, Part 3, Uncertainty Analysis of Limiting Conditions for Operation for SONGS Units 2 and 3," CEN-183(S)-P, Combustion Engineering, October 1984.

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

19. Letter from A. E. Scherer (CE) to C. O. Thomas (NRC), Enclosure, "Supplemental Material for Inclusion in CENPD-132, Supplement 3-P," November 5, 1985.

20. Topical Report, Enclosure 1-P to LD-81-095, "CE ECCS Evaluation Model Flow Blockage Analysis, December 1981.

21. Letter from A. E. Scherer (CE) to G. W. Knighton (NRC), transmitting "CPC/CEAC Software Modifications for the CPC Improvement Program," CEN-308-P, Rev.00-P, August 30, 1985.

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23. Letter from A. E. Scherer (CE) to G. W. Knighton (NRC), transmitting "Reload Data Block Constant Installation Guidelines," CEN-323-P, Rev.00-P, and "CPC Protection Algorithm Software Change Procedure," CEN-39(A)-P, Rev. 03-P, February 10, 1986.

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27. Letter from D. M. Crutchfield (NRC) to A. E. Scherer (CE), Acceptance for Referencing of Licensing Topical Report CEN-323-P, Rev. 00-P, "Reload Data Block Constant Installation Guidelines," August 5, 1986.

28. Letter from D. M. Crutchfield (NRC) to A. E. Scherer (CE), Acceptance for Referencing of Licensing Topical Report CEN-39(A)-P, Rev. 03-P, "CPC Protection Algorithm Software Change Procedure," September 29, 1986.

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31. "Statistical Combination of Uncertainties for Waterford-3," CEN-343(C)-P, Combustion Engineering, October 1986.

32. Letter from K. W. Cook (LPL) to George W. Knighton (NRC) "Response to RAI on Waterford 3 Cycle 2 Reload Analysis Report," November 26, 1986.

33. Letter from K. W. Cook (LPL) to George W. Knighton (NRC) "Waterford SES Unit 3, Docket No. 50-382; Waterford 3 Cycle 2 Large Break LOCA Analysis," December 23, 1986.