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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 32 TO FACILITY OPERATING LICENSE NO. DPR-72

FLORIDA POWER CORPORATION, ET AL

CRYSTAL RIVER UNIT NO. 3 NUCLEAR GENERATING PLANT

DOCKET NO. 50-302

1.0 Introduction

By letters dated March 21, 1980, April 14 and 30, 1980, June 6 and 13, 1980, and July 22 and 31, 1980, Florida Power Corporation (FPC or licensee) requested amendment of Appendix A to the Crystal River-3 (CR-3) Operating License to permit power operation during Cycle 3.

Reference 5-2 includes a Babcock & Wilcox Company (B&W) report BAW-1607, Rev. 1, dated April 1980 to support the CR-3 Cycle 3 operation at an upgraded power level of 2544 MWt as opposed to Cycle 2 power level of 2452 MWt. The report describes the fuel system design, nuclear design, thermal-hydraulic design, accident analysis, and startup test program. The report supports Cycle 3 operation for 335 effective full power days (EFPD) at the upgraded power level.

Our letter of April 4, 1980 stated we couldn't consider a power increase to 2544 MWt until we resolve our concerns about sensitivity to secondary side transients and other post TMI-2 concerns. By a letter dated April 30, 1980 (Reference 4-4), FPC informed the NRC of their decision to operate at the currently licensed power level of 2452 MWt when Cycle 3 is started. Also, in Reference 4-4, FPC submitted modified technical specifications that reflect Cycle 3 operation without the power upgrade. The reason given by FPC for not going to the higher power level at this time is the unavailability of the reactor coolant pump power monitors (RCPPMs). The RCPPMs are required for the upgraded power operation so that in a postulated loss of coolant flow (LOCF) event, the RCPPMs will trip the reactor sooner than the existing flux/flow comparator. The faster RCPPM response (0.62 sec as compared to 1.40 sec for the flux/flow comparator) decreases the time during which the reactor is operating at 2544 MWt with degrading flow conditions (LOCF), so that the departure from nucleate boiling ratio (DNBR) criterion is not violated.

Even though no power increase will be implemented when the plant resumes operation in Cycle 3, this safety review and accident analyses evaluation are based on the higher power level of 2544 MWt. Sections 2.0, 3.0, 4.0, and 5.0 below evaluate the Fuel System Design, Nuclear Design and Startup Test Program, Thermal-Hydraulic, and Accidents and Transients respectively. The technical specifications for Cycle 3 operation at the lower power level of 2452 MWt are evaluated at the end of sections 3.0 and 4.0 below. References are listed at the end of each section.

8008140489

2. Evaluation of Fuel System Design

2.1 Fuel Assembly Mechanical Design

The fresh Babcock and Wilcox Mark B-4 fuel assemblies loaded as Batch 5 at the end of Cycle 2 (EOC 2) are mechanically interchangeable with Batches 2 and 3 (Mark B-3) and Batch 4 (Mark B-4) fuel assemblies previously loaded at Crystal River Unit 3. Fifty-two Batch 2 assemblies and four Batch 4 assemblies have been discharged and fifty-six Batch 5 assemblies will be loaded for Cycle 3. This reload scheme is a revision (2-1) to that originally proposed (2-2) by the licensee. The change allows the replacement of one Batch 4 assembly with a broken hold-down spring and three additional, symmetric, Batch 4 assemblies. Our evaluation of the broken hold-down spring is further discussed in Section 2.4.2.

The Mark B-4 fuel assembly has been previously approved (2-3) by the NRC staff and is utilized in other B&W nuclear steam supply systems. The new assemblies have modified end fittings, mainly to reduce the coolant flow pressure drop. The Mark B-4 assemblies also incorporate some modification to the spacer grid corner cells to reduce wear during fuel handling. Two assemblies will contain primary neutron sources and two assemblies will contain regenerative neutron sources in Cycle 3. The justification (2-4) for the design of the retainer is applicable to the neutron sources used in Cycle 3.

2.2 Fuel Rod Design

Although Crystal River 3 Batch 4 and Batch 5 utilize the same Mark B-4 fuel, the Batch 5 assemblies incorporate a slightly higher initial fuel density. The change, from 94 to 95 percent of theoretical density, is a consequence of using a more stable (densification resistant) fuel material. This change results in a shorter initial, but almost identical densified, active fuel length.

The fuel pellet end configuration has also changed to a truncated cone dish for Batch 5 as opposed to a spherical dish for the previous four batches. This minor change facilitates manufacturing and does not significantly alter the performance characteristics of the fuel.

2.2.1 Cladding Collapse

Due to the cumulative nature of cladding deformation, creep collapse analyses were performed for the previous two cycles as well as the proposed third cycle of operation. Batches 2 and 3 are more limiting than Batches 4 and 5 due to their previous incore exposure time. That analysis was performed for the most limiting fuel assembly power history using the CROV computer code and procedures described in the topical report BAW-10084PA, Rev. 2 (2-5). The analysis conservatively determined a creep collapse time of 25,000 effective full power hours (EFPH) of operation. Since the collapse time is greater than the estimated residence time for the most limiting assembly at EOC3 (22,800 EFPH), we conclude that cladding creep collapse has been adequately considered.

2.2.2 Cladding Stress

The licensee stated (2-2) that the Batch 2 and 3 reinserted fuel assemblies are the limiting batches from a cladding stress point of view because of their lower density and longer previous exposure time.

We have examined the mechanical analysis section of the CR-3 Fuel Densification Report and find the cladding stress analyses were performed for both beginning-of-life and end-of-life (EOC-3) conditions for first cycle fuel. The results, shown in Table 3.5-1 of the report, compare cladding circumferential stress levels with the yield and ultimate strength of Zircaloy under a variety of conditions. The cladding stress levels are strongly dependent on the pressure differential across the cladding wall and are limiting (maximum) for beginning-of-life when the rod internal pressure is minimum. This is contrary to the exposure dependence cited in Reference 2-6. In addition, we find no evidence that the lower density fuels are limiting in the analyses.

We agree that the pressure differential across the cladding wall is a major contributor to the cladding stress level. The external system pressure remains relatively constant (2200 psia) during normal operation. The differential across the cladding wall is the greatest, therefore, when the rod internal pressure is much less, or much greater, than the coolant pressure. As discussed in Section 2.2.4, the rod internal pressure does not exceed system pressure during normal operation. Therefore, limiting cladding stress conditions based on rod internal gas pressure exist at beginning-of-life. The licensee has informed us that Batches 4 and 5 fuel have a higher initial fill gas pressure than Batches 2 and 3. As a result, the analyses presented in the CR-3 densification report may be applied to Cycle 3 operation.

We also note, however, that fuel swelling, cladding creep, and fuel-cladding mechanical interaction may also contribute to the effects of internal gas pressure on cladding stress levels. In general, these effects are localized and the licensee's design bases (CR-3 FSAR 3.1.2.4.2 (2-7)) state that such "stresses relieved by small material deformation are permitted to exceed the yield strength." We do not believe that the design criterion for cladding stress will limit the operational flexibility of CR-3. Therefore, we conclude that cladding stress limits, will not be exceeded during normal operation of Cycle 3 fuel at CR-3.

2.2.3 Cladding Strain

The fuel design criteria (CR-3 FSAR Section 3.1.2.4.2 (2-7)) specify a 1% limit on cladding plastic strain due to diameter increases resulting from fuel swelling, thermal ratcheting, creep and internal gas pressure. Strain limits were established on the basis of low-cycle fatigue techniques, not to exceed 90% of material fatigue life. The design evaluation, discussed in Section 3.2.4.2.1 of the CR-3 FSAR (2-7) and Section 3.5.2 of the CR-3 Fuel Densification Report (2-6), was performed for design pellet burnup and heat generation rate as well as limiting dimensional tolerances. These conditions are considerably beyond those expected for Cycle 3 at Crystal River Unit 3. The results show circumferential plastic strain is less than 1% at design EOL burnup, and cumulative fatigue damage after three cycles of operation is less than 90% of material fatigue life. We conclude that the cladding strain and fatigue limits have been adequately considered for Cycle 3 operation.

2.2.4 Rod Internal Pressure

Section 4.2 of the Standard Review Plan (2-8) addresses a number of acceptance criteria used to establish the design bases and evaluation of the fuel system. Not all of these have been addressed in the licensee's reload application or previous reports. Among those which may affect the operation of the fuel rod is the internal pressure limit. Our current criterion (SRP 4.2, Section II.A.1(f)) states that fuel rod internal gas pressure should remain below normal system pressure during normal operation unless otherwise justified. Meeting this criterion is also a condition of acceptance as discussed previously in Section 2.2.2 (cladding stress).

Although the CR-3 FSAR states that "at end-of-life, fission gas pressure does not exceed system pressure" (2-7), it also describes the use of an internal gas pressure of 3,300 psi to determine fuel cladding internal design conditions. It is not clear whether the limit of rod internal pressure on system pressure is a design criterion or simply an analytical result. The analysis is not described in the reload submittal. Furthermore, we believe (2-9) that some of the analytical methods utilized by Babcock and Wilcox may be deficient at high burnups.

In response to a question of this criterion, Florida Power has stated (2-10) that fuel rod internal pressure will not exceed nominal system pressure during normal operation for Cycle 3. This analysis is based on the use of the B&W TAFY code (2-11) rather than a newer B&W code called TACO (2-12). Although both of these codes are currently approved for use in safety analyses, we believe that only the newer TACO code is capable of correctly calculating fission gas release (and therefore rod pressure) at very high burnups. Babcock and Wilcox has responded (2-13) to this concern with an analytical comparison between both codes. In this response, they have stated that the internal fuel rod pressure predicted by TACO is lower than that predicted by TAFY for fuel rod exposures of up to 42,000 MWD/TL. Although we have not examined the comparison, we note that the analyses exceed the expected exposure in CR-3 Cycle 3 by a large margin. We conclude that the rod internal pressure limits have been adequately considered.

2.3 Fuel Thermal Design

There are no major changes between the new Batch 5 fuel and previous batches reinserted in the Cycle 3 core. The increase in initial fuel density (95% T.D.) results in a slightly higher linear heat rating for the fuel based on centerline melt. The rating was established with the TAFY code (2-11). We have performed an independent check of the Batch 5 fuel design parameters and agree that fuel melting will not occur for the linear heat ratings given in Table 4.2 of Ref. 2-2.

The average fuel temperature as a function of linear heat rate and lifetime pin pressure data used in the LOCA analysis (Section 7.15 of the Reload submittal) are also calculated with the TAFY code (2-11). Babcock and Wilcox has stated (2-2) that the fuel temperature and pin pressure data used in the generic LOCA analysis (2-14) are conservative compared to those calculated for Cycle 3 at Crystal River 3.

As previously mentioned in Section 2.2.4 of this evaluation, B&W currently has two fuel performance codes, TAFY (2-11) and TACO (2-12), which could be used to calculate the LOCA initial conditions. The older code TAFY has been used for the Cycle 3 LOCA analysis. Recent information (2-15) indicates that the TAFY code predictions do not produce higher peak cladding temperatures than TACO for all Cycle 3 conditions as suggested in Ref. 2-13. The issue involves calculated fuel rod internal gas pressures that are too low at beginning of life. The rod internal pressures are used to determine swelling and rupture behavior during LOCA. Babcock and Wilcox has proposed (Attachment 3 of Ref. 2-13) a method of resolving this issue which has not yet been accepted by the staff. While we have not yet completed the review, we believe the Cycle 3 LOCA initial conditions are acceptable as submitted.

2.4 Operating Experience

Babcock & Wilcox has accumulated operating experience with the Mark B 15x15 fuel assembly at all of the eight operating B&W 177-fuel assembly plants. A summary of this operating experience is given on page 4-3 of Ref. 2-2.

2.4.1 Guide Tube Wear

Significant wear of Zircaloy control rod guide tubes has been observed in facilities designed by Combustion Engineering. Similar wear has also been reported in those facilities designed by Westinghouse. In a letter dated June 13, 1978, we requested information from Babcock and Wilcox on the susceptibility of the facilities designed by B&W to guide tube wear. The information provided by B&W in a letter dated January 12, 1979, was insufficient for us to conclude that guide tube wear was not a significant problem in B&W plants. This was documented in our letter to B&W dated August 22, 1979.

Because significant guide tube wear could impede the control rod scram capability, and also affect the required coolable geometry of the reactor core, we consider this wear phenomenon a potential safety concern. Therefore, we requested (2-16) additional information from the licensee on the wear characteristics of the control rods on the guide tubes at CR-3. The response to this request has not yet been received. The licensee has stated (2-10) that a generic response to this request has been prepared by Babcock and Wilcox. The report, B&W Control Rod Guide Tube Wear Generic Report (BAW-1623), has been concurred with by the licensee but has not been received by the NRC.

We have, however, received preliminary information on post-irradiation examinations of identical guide tubes for wear in Rancho Seco spent fuel (2-17). The results of these measurements indicate that through-wall wear or excessive wall degradation will not likely occur during anticipated fuel residence time under control rod assemblies. On the basis of this preliminary information and the imminent documentation of a complete generic evaluation, we conclude that guide tube wear has been adequately addressed for Crystal River 3 during Cycle 3.

2.4.2 Hold-Down Spring Damage

Davis-Besse Unit 1, another B&W designed reactor, reported fuel assembly hold-down spring damage in late May of this year. Due to the similarity of the reactor and fuel assemblies used at Crystal River Unit 3, all in-core and discharged fuel assemblies were examined for hold-down spring damage. A broken hold-down spring was discovered in assembly NJ018E, a Batch 4 assembly that had been in core location N-14 during Cycle 2. This assembly and three symmetric assemblies were replaced with Batch 5 fuel. The resulting changes to the Crystal River Unit 3 Cycle 3 Reload Report (2-2) are discussed in Reference 2-1.

2.5 Rod Bow

The licensee has stated that a rod bow penalty has been calculated according to the procedure approved in reference 2-17. The burnup used is the maximum fuel assembly burnup of the batch that contains the limiting (maximum radial x local peak) fuel assembly. For Cycle 3, this burnup is 31,358 MWD/MTU in a Batch 3 assembly. The resultant net rod bow penalty after inclusion of the 1% flow area reduction factor credit is 2.8% reduction in DNBR. However, this rod bow penalty is offset by the 10.2% DNBR margin included in trip setpoints and operating limits.

2.6 Densification Power Spike

The densification power spike was eliminated from DNBR evaluations based on the NRC approval of this change in reference 2-19.

2.7 Cladding Strain and Flow Blockage

The licensee has responded (2-20) to our request for information concerning the new fuel cladding strain and fuel assembly flow blockage models described in NUREG-0630.

Florida Power Corporation has reviewed all of the subject information supplied by Babcock & Wilcox and is in agreement with the results that calculated peak fuel cladding temperature will remain unchanged or lowered with the use of the new NRC ramp-rate-dependent correlations, and that compliance with 10 CFR 50.46 is assured for Crystal River Unit 3.

2.8 References

2-1 P. Y. Baynard (Florida Power) letter to Director, Office of Nuclear Reactor Regulation (NRC), dated June 6, 1980.

2-2 Crystal River Unit 3 - Cycle 3 Reload Report, Babcock and Wilcox Company Report BAW-1607, February 1980.

2-3 R. W. Reid (NRC) letter to W. P. Stewart (Florida Power), dated July 3, 1979.

2-4 BPRA Retainer Design Report, Babcock and Wilcox Company Report BAW-1496, May 1978.

2-5 Program to Determine In-Reactor Performance of B&W Fuels - Cladding Creep Collapse, Babcock and Wilcox Company Report BAW-10084P-A, October 1978.

2-6 Crystal River Unit 3 Fuel Densification Report, Babcock and Wilcox Company Report BAW-1397, August 1973.

2-7 Crystal River Unit 3 Nuclear Generating Plant Final Safety Analysis Report, Docket No. 50-302, Florida Power Corporation, March 1977.

2-8 Standard Review Plan, Section 4.2 (Rev. 1), "Fuel System Design," U.S. Nuclear Regulatory Commission Report NUREG-75/087.

2-9 J. Stolz (NRC) letter to Florida Power Corporation, dated February 11, 1977.

2-10 R. M. Bright (Florida Power) letter to Office Director of Nuclear Reactor Regulation (NRC), dated June 13, 1980.

2-11 C. D. Morgan and H. S. Kao, "TAFY-Fuel Pin Temperature and Gas Pressure Analysis," Babcock and Wilcox Company Report BAW-10044, May 1972.

2-12 "TACO-Fuel Pin Performance Analysis, Babcock and Wilcox Company Report BAW-10087P-A, Rev. 2, August 1977.

2-13 J. H. Taylor (B&W) letter to P. S. Check (NRC), dated July 18, 1978.

2-14 R. C. Jones, J. R. Biller, and B. M. Dunn, "ECCS Analysis of B&W's 177-FA Lowered-Loop NSS," Babcock and Wilcox Company Report BAW-10103A, Rev. 3, July 1977.

2-15 R. O. Meyer (NRC) memorandum to L. S. Rubenstein (NRC) on "TAFY/TACO Fuel Performance Models in B&W Safety Analyses," dated June 10, 1980.

2-16 R. W. Reid (NRC) letter to W. P. Stewart (Florida Power) dated November 23, 1979.

2-17 J. J. Mattimoe (Sacramento Municipal Utility District) letter to R. W. Reid (NRC) on "Guide Tube Wear Measurements--Preliminary Results," February 15, 1980.

2-18 L. S. Rubenstein (NRC) letter to J. H. Taylor (B&W) on "Evaluation of Interim Procedure for Calculating DNBR Reductions due to Rod Bow," dated October 18, 1979.

2-19 S. A. Varga (NRC) letter to J. H. Taylor (B&W) on "Update of BAW-10055, Fuel Densification Report," December 5, 1977.

2-20 J. A. Hancock (Florida Power) letter to D. G. Eisenhut (NRC) dated December 20, 1979.

3.0 Evaluation of Nuclear Design and Startup Test Program

3.1 General

A core loading diagram for Cycle 3 is presented in the reload report (BAW 1607, Revision 1) along with enrichment and burnup distributions. The nuclear parameters for Cycle 3 are compared to those for Cycle 2 including reactivity coefficients, boron worths and rod group worths. An analysis of the shutdown margin capability and a radial power map at BOC are also given.

The core physics calculations are performed with PDQ07 code (Reference 3-1) which has been reviewed and approved by the staff. This code has been used for analysis of the previous cycles of CR-3. The results of the analysis show small differences between Cycle 3 and Cycle 2 values, occasioned by the difference in cycle lengths (335 EFPD for Cycle 3 vs. 275 EFPD for Cycle 2) and by the fact that the core is not yet in its equilibrium configuration. The analysis of shutdown margin shows that 1.84% Δ k/k exists at end of cycle compared to the required 1.0% Δ k/k for hot shutdown. The calculated radial power distribution at BOC shows adequate margin to limits.

Based on the fact that approved methods have been used to obtain the core characteristics, that margin exists to limiting values of the parameters, and that startup testing will be used to obtain measured values of important parameters, we find the analysis of core parameters to be acceptable.

3.2 Evaluation of Fuel Loading Error and Rod Misoperation Transients

The accident and transient analyses presented in the FSAR have been examined to determine the effect of increasing core power level to 2544 MWt. The results of this examination were evaluated as part of the review of the Cycle 2 reload submittal. The results of that evaluation are presented in Table 1 of the Safety Evaluation (Reference 3-2) for that submittal. The conclusions reached in that table still apply to the rod withdrawal error, rod misoperation, fuel loading error, and rod ejection events.

3.3 Startup Test Program

The physics startup test program as submitted by the licensee in BAW 1607 has been revised (Reference 3-4). The final program is identical to that used for Cycle 2 with the exception of a revised review criterion applied to the power distribution measurements.

The program consisted of zero-power test and power escalation test. The zero-power test consisted of (a) critical boron concentration, (b) temperature reactivity coefficient, (c) control rod group reactivity worth, (d) ejected control rod reactivity worth measurements, and (e) a symmetry test involving swapping of symmetrical rods.

The power escalation tests consisted of (a) core power distribution verification at 40%, 75% and 100% full power, (b) incore vs. excore detector imbalance correlation verification, (c) temperature reactivity coefficient, and (d) power Doppler reactivity coefficient measurements.

The staff has reviewed the complete physics startup test program including review and acceptance criteria and remedial actions and finds this program acceptable.

3.4 Evaluation of Power Distribution and Reactivity Technical Specification Changes

We have reviewed Figures 2.1-2, 2.1-1, 3.1-1, 3.1-2, 3.1-3, 3.1-4, 3.1-9, 3.1-10, 3.2-1 and 3.2-2 and Tables 2.2-1 and 3.2-2 of the proposed Technical Specifications (Reference 3-3). The same procedures and techniques were employed to derive these curves and tables as have been used for previous cycles. The changes from the previous cycle curves are not large and are consistent with the changes in core parameters. On these bases we find the above cited changes to the Technical Specifications to be acceptable.

3.5 References

3.1 PDQ07 Users Manual, BAW-10117 PA, January, 1977.
3.2 Letter, R. W. Reid, NRC to W.P. Stewart, FPC, July 3, 1979 with attachments.
3.3 Letter, R. M. Bright, FPC, to Director, NRR, NRC, April 30. 1980.
3.4 Letter, R. M. Bright, FPC, to Director, NRR, NRC, June 13, 1980.

4.0 Evaluation of Thermal Hydraulic Design

4.1 DNBR Evaluations

A comparison between the thermal hydraulic design conditions for Cycles 1, 2 and 3 (Reference 4-1) is listed in Table 4.1. The design power level for Crystal River 3 Cycle 3 reload is 2544 MWt even though it will actually operate at the licensed core power level of 2452 MWt. However, the thermal hydraulic design calculations in support of Cycle 3 operation assumed a power level of 2568 MWt (same as for Cycle 2) for consistency with other B&W plants. A summary of our evaluation follows:

(a) Critical Heat Flux - The B&W-2 critical heat flux (CHF) correlation in conjunction with the TEMP thermal hydraulic code (Reference 4-2) was used for DNBR evaluation instead of the W-3 correlation used for Cycle 1. The B&W-2 correlation has been approved by the staff and is currently used to license all operating B&W plants with Mark-B fuel assembly cores including the Crystal River 3, Cycle 2. (Reference 4-3).

(b) Reactor Coolant Flow - The assumed system flow for Cycle 3 analyses is 106.5% of the design flow (88,000 gpm/pump) and is the same as the low flow limit included in the Technical Specifications and analyses for Cycle 2. The flow rate from measurements at Crystal River 3 indicate a system flow capability of 109.5% of design flow rate, including measurement uncertainty.

(c) Rod Bow - As discussed in Section 2.5, a net rod bow penalty of 2.8% reduction in DNBR has been calculated by approved methods. This is acceptable for Cycle 3 operation since a 10.2% DNBR margin, exclusive of the penalty is available.

(d) Peaking Factor - The licensee has stated that a reference design radial x local power peaking factor ($F_{\Delta H}$) of 1.71 was used for Cycle 2 and 3 evaluations. The Cycle 1 $F_{\Delta H}$ of 1.78 was reduced to 1.71 in conjunction with orifice rod assembly and burnable poison rod assembly removal.

4.2 Pressure-Temperature Limit Analysis

The licensee presented pressure-temperature limit curves for four and three pump operation. The most limiting of these curves (four-pump) provides the basis for the reactor protection system variable low-pressure trip function. The curves are based on a minimum DNBR of 1.433, which provides 10.2% margin to the CHF correlation limit and allows flexibility for future cycle designs.

4.3 Loss-of-Coolant-Flow-Transients

The flux/flow trip is designed to protect the plant during pump coastdowns from four-pump operation or to act as a high flux trip during partial-pump operation. Redundant pump monitors will be installed for each Crystal River 3 pump prior to operation at 2544 MWt in order to trip the reactor immediately upon the loss of power to two or more pumps. The flux/flow trip setpoint will then serve only to protect the plant during a one-pump coastdown from four-pump operation. The licensee stated that the margin for flux/flow setpoint was determined with a transient analysis initiated from 108% instead of 102% of 2544 MWt. This margin allows for uncertainties in power measurements and heat balance error. While the analyses are acceptable for operation at 2544 MWt, the indicated power measurement uncertainties (6%) imposed on a 102% real power level infers an operating mode that would be unacceptable. The licensee will not be permitted to operate the plant with the sustained indicated power level above the license limit (100%). Operation at 100% is acceptable if the power measurement uncertainties are not greater than ± 2%, giving a maximum real power level of 102%.

For the analysis, actual measured one-pump coastdown data were used and maximum additive trip delays were used between the time trip conditions were reached and actual control rod motion started. Once a flux/flow trip limit was found to be adequate by thermal-hydraulic analysis, error adjustments were made to account for flow measurement noise and instrument error before the actual trip setpoint was determined. The staff finds these analyses methods to be acceptable. The recommended Cycle 3 thermal-hydraulic flux/flow trip limit of 1.10 (actual in-plant setpoint of 1.07) resulted in a transient minimum DNBR of 1.75 (B&W-2) during the pump coastdown. This represents >34% DNBR margin to the correlation limit of 1.30, and is therefore acceptable.

The four-pump coastdown and locked rotor transients were analyzed at a power level of 102% of 2568 MWt. The results are discussed in Section 5, "Evaluation of Accidents and Transients".

Table 4.1 Cycle 1, 2 and 3 Thermal-Hydraulic Design Conditions

	Cycle 1 <268.8 EFPD	Cycle 1 >268.8 EFPD	Cycles 2 & 3 2544 MWt
Design power level, MWt	2452	2452	2568
System pressure, psia	2200	2200	2200
Reactor coolant design flow, gpm % design	352,000	352,000	352,000
Reactor Coolant Flow, % design	105	105	106.5
Ref design radial x local power peaking factor, $F_{\Delta H}$	1.78	1.71	1.71
Ref design axial flux shape	1.5 cosine	1.5 cosine	1.5 cosine
Hot channel factors			
Enthalpy rise	1.011	1.011	1.011
Heat flux	1.014	1.014	1.014
Flow area	0.98	0.98	0.98
Densified active length, in.	141.12	140.2[b]	140.2[b]
Avg heat flux at 100% power, Btu/h-ft^2	167×10^3	168×10^3	176×10^3
Max heat flux at 100% power, Btu/h-ft^2	446×10^3[a]	431×10^3	452×10^3
CHF correlation	W-3	B&W-2	B&W-2
Minimum DNBR (% power)	1.61 (114) 1.92 (102)	2.14 (112) 2.27 (108) 2.49 (102)	1.98 (112) 2.12 (108) 2.33 (102)

[a] The maxmium heat fluxes shown are based on reference peaking and average flux. For Cycle 1, thermal hydraulic calculations also included the densification spike factor in the DNBR calculations. B&W no longer considers this spike factor in DNBR calculations, as described in Section 2.6.

[b] 140.2 inches is a conservative (minimum) value used in Cycle 2 and 3 analyses; it is the minimum densified length for any B&W fuel.

4.4 Evaluation of Thermal Hydraulic Technical Specification Changes

The staff has reviewed hot leg temperatures and low flow limits in Table 3.2-1 of the proposed technical specification for Cycle 3 operation (Reference 4-4). The same procedures and techniques were employed to derive the values in this table for this cycle as were used for previous cycles. These values are the same or slightly different from previous cycles and are consistent with the changes in the parameters. The minimum reactor coolant flow rates of 139.7×10^6 lbs/hr for four pump operation and 104.4×10^6 lbs/hr for three pump operation reflect the values used in the analyses. Therefore we conclude that these changes are acceptable.

References

4-1 Crystal River Unit 3, Cycle 3 Reload Report, BAW-1607, Rev. 1, Babcock and Wilcox, Lynchburg, Virginia, April 1980.

4-2 Correlation of Critical Heat Flux in Bundle Cooled by Pressurized Water, BAW-10000A, Babcock and Wilcox, Lynchburg, Virginia, May 1976.

4-3 Crystal River Unit 3, Cycle 2 Reload Report, BAW-1521, Babcock and Wilcox, Lynchburg, Virginia, February 1979.

4-4 Letter R. M. Bright, FPC, to Director, NRR, NRC, April 30, 1980.

4-5 Crystal River Unit 3, Fuel Densification Report, BAW-1397, Babcock and Wilcox, Lynchburg, Virginia, August 1973.

4-6 S. A. Varga (NRC) to J. H. Taylor (B&W), Letter, "Update of BAW-10055, Fuel Densification Report", December 5, 1977.

4-7 Letter R. M. Bright, FPC, to Director, NRR, NRC, April 30, 1980.

parameters discussed below is to lower the MDNBR obtained during the course of the transient.

a. Design initial power level is 102% of 2544 MWt, while the value used in the analysis is 102% of 2568 MWt.

b. The Cycle 3 flow rate is 109.5% of 352,000 gpm, while the value used in the analysis is 106.5%.

c. Core flow rate as a function of time following a LOCF event is expected to be larger than shown in the FSAR Figure 14-17 for four-pump coastdown and larger than shown in Figure 14-19a for the locked-rotor. However, the analysis used the flow rates shown in the above mentioned figures.

d. Expected values at the beginning of Cycle 3 (the worst time during the cycle life for a LOCF event) of the Doppler coefficient, the moderator temperature coefficient, and the design radial x local power peaking factor ($F_{\Delta H}$) are -1.52×10^{-5} Δk/k.°F, -0.30×10^{-4} Δk/k.°F, and 1.47 respectively. The values used for the above parameters in the LOCF analysis are -1.27×10^{-5} Δk/k.°F, 0.0 Δk/k.°F, and 1.71 respectively.

The minimum DNBR obtained during this transient is 2.10 which is well above the 1.45 FSAR value. It is noteworthy that two principal differences between the FSAR analysis and the latest Cycle 3 analysis are that the FSAR analysis used W-3 CHF correlation and a reactor protection system (RPS) flux/flow trip delay time of 1.40 sec before the control rods start to move into the core, while the Cycle 3 analysis used the BAW-2 CHF correlation and an RPS RCPPM trip delay time of 0.62 sec.

5.2.2 Locked-Rotor

The locked-rotor event is analyzed using the same conservative assumptions used in the four-pump coastdown transient discussed above. An additional assumption for the locked-rotor event was to initiate film boiling at a DNBR of 1.43 instead of the 1.3 limit.

The licensee concluded that less than 0.5% of the fuel pins in the core will experience a DNBR less than 1.43, and no pins will experience a DNBR less than 1.00. Even if the 0.5% of the fuel pins which experienced a DNBR less than 1.43 were to fail, the offsite dose releases resulting from a locked rotor event are expected to be a small fraction of 10 CFR 100 limits (see Table 5.2).

5.2.3 Conclusion

Based on the above we conclude that the accident and transient analysis is acceptable.

TABLE 5.1

Comparative Review of FSAR and Cycle 3 Parameters for Some Key Events

Event, Parameter	FSAR	Cycle 3
Rod Withdrawal		
BOL: Doppler ($\Delta k/k.°F$)	-1.17×10^{-5}	-1.52×10^{-5}
MTC ($\Delta k/k.°F$)	0.0	-0.30×10^{-4}
max. rod worth ($\%\Delta k/k$)	up to 12.9	< 9.37
Mod. Dilution		
BOL: Boron Conc. (ppm)	1150	1185
Boron worth (ppm/$\%\frac{\Delta k}{k}$)	100	108
MTC ($\Delta k/k.°F$)	$+0.5 \times 10^{-4}$	-0.3×10^{-4}
Dilution rate (gpm)	up to 500	< 100
Cold Water (2-RCP start)		
EOL: Doppler ($\Delta k/k.°F$)	-1.3×10^{-5}	-1.6×10^{-5}
MTC ($\Delta k/k.°F$)	-4.0×10^{-4}	-2.63×10^{-4}
Rod Drop		
EOL: Doppler ($\Delta k/k.°F$)	-1.3×10^{-5}	-1.61×10^{-5}
MTC ($\Delta k/k.°F$)	-3.0×10^{-4}	-2.63×10^{-4}
max. rod worth ($\% \Delta k/k$)	0.40	0.20
MSLB		
EOL: MTC ($\Delta k/k.°F$)	-3.0×10^{-4}	-2.63×10^{-4}
Ejected Rod		
BOL: Doppler ($\Delta k/k.°F$)	-1.17×10^{-5}	-1.52×10^{-5}
MTC ($\Delta k/k.°F$)	0.0	-0.3×10^{-4}
max. rod worth ($\% \Delta k/k$)	0.65	0.49
MFWLB		
BOL: Doppler ($\Delta k/k.°F$)	-1.17×10^{-5}	-1.52×10^{-5}
MTC ($\Delta k/k.°F$)	0.0	-0.3×10^{-4}

5.0 Evaluation of Accidents and Transients

5.1 General

The licensee examined each FSAR accident and transient with respect to changes in Cycle 3 parameters to determine the effect of upgrading the reactor power from 2452 to 2544 MWt. All FSAR accidents and transients with the exception of the loss-of-coolant flow (LOCF), i.e., the four-pump coastdown and the locked-rotor transients, were analyzed during the FSAR stage at 2568 MWt. This power level is higher than the requested power upgrade of 2544 MWt. Except for the LOCF, the licensee examined all FSAR accidents and transients relative to Cycle 3 operation by comparing input parameters as stated in the FSAR and as calculated for Cycle 3 and concluded that they are bounded by the FSAR and the fuel densification report analyses (References 5-1 and 5-3). The four-pump coastdown and the locked-rotor transients were reanalyzed at 102% of 2568 MWt for consistency with other B&W reactors. The LOCF event is discussed in Section 5.2 below.

A comparative review of reactivity parameters of FSAR accidents and transients except the LOCF is shown in Table 5.1. The applicability of the FSAR and reload report analyses to Cycle 3 operation is summarized in Table 5.2.

5.2 Loss-of-Coolant Flow (LOCF)

At the Cycle 2 power level of 2452 MWt, the reactor protection system depends on the flux-to-flow comparator to trip the reactor to avoid a MDNBR less than 1.3 for a 1-, 2-, 3-, or 4-pump coastdown transient. However, at the requested power upgrade of 2544 MWt, the flux-to-flow comparator, which has a trip delay time of 1.40 sec, is too slow to avoid violating the DNBR criterion for 2-, 3-, or 4-pump coastdown events. Therefore, the licensee has submitted for NRC's review and approval a proposal to add a reactor coolant pump power monitor (RCPPM)* which will continuously monitor each RCP power supply and upon power interruption to two or more RCPs will send a trip signal to the control rods with a total trip delay time of 0.62 sec. This faster trip response decreases the time during which the reactor flux-to-flow ratio exceeds the operating values and maintains the MDNBR above the 1.3 criterion during the course of the event.

5.2.1 Four-Pump Coastdown

The four-pump coastdown transient was reanalyzed at 102% of 2568 MWt assuming conservative input parameters as compared to Cycle 3 expected parameters. The effect of using the conservative

*RCPPM is being reviewed currently by NRC.

TABLE 5.2 Applicability of FSAR and Reload Report Analyses Power Level to Cycle 3

Accident/Transient	Analysis Reference	Analysis Power level, MWt	Status of Analysis Relative to Cycle 3: Percent of 2544 MWt	Remarks
Rod Withdrawal	FSAR	100% of 2568	101%	see footnote 1
Moderator Dilution	FSAR	100% of 2568	101%	see footnote 2
Cold Water (2-pump start)	FSAR	50% of 2568	50.5%	see footnote 3
4-PCD	Reload Report	102% of 2568	103%	Bounding
Locked-Rotor	Reload Report	102% of 2568	103%	Bounding
Stuck-in, Stuck-out, Rod Drop	FSAR	100% of 2568	101%	see footnote 4
Loss of Electrical Power	FSAR	100% of 2568	101%	see footnote 5
SLB	FSAR	100% of 2568	101%	see footnote 1
S.G. Tube Rupture	FSAR	100% of 2568	101%	see footnote 5
Fuel Handling	FSAR	100% of 2568	101%	see footnote 5
Rod Ejection	FSAR	100% of 2568	101%	see footnote 1
Max. Hypothetical Accident	FSAR	100% of 2568	101%	see footnote 5
Waste Gas Tank Rupture	FSAR	100% of 2568	101%	see footnote 5
LOCA	References 4, 5, 6	100% of 2772	109%	Bounding
MFWLB	FSAR	100% of 2568	101%	see footnote 2
Letdown Line Rupture Outside Containment	Reload Report	100% of 2603	102%	Bounding

Footnotes (Table 5.2)

1. The FSAR analysis assumed the reactor power before the accident to be 100% of 2568 MWt and the reactor is assumed to trip at 112% of 2568 MWt. This is more conservative than starting from 102% of 2544 MWT and tripping at 110% of 2544 MWt since more energy is added to the system for the FSAR analysis assumptions.

2. The FSAR analysis assumed the reactor power before the accident to be 100% of 2568. While the effect of a higher initial power of 102% of 2544 MWt (2595 MWt) is to cause the pressure trip to occur slightly sooner and the peak pressure to be slightly higher, the peak pressure is expected to be lower than the code safety limit of 2750 psia.

3. If the two pumps are started from 52% of 2544 MWt, the transient will produce a slightly higher neutron power, thermal power, and peak pressure. Since the FSAR analysis (at 50.5% of 2544 MWt) produced maximum neutron power of 75%, maximum thermal power of 65%, and a 150 psi increase over steady-state pressure of 2200 psi, the steady-state power increase is not expected to produce peak thermal power or peak pressure higher than the overpower safety limit of 112% or the code pressure limit of 2750 psia.

   FPC has been operating the Crystal River 3 plant since Cycle 2 with a modified Technical Specification that does not allow plant operation with less than three RCPs on. Therefore, the cold water accident presented in the FSAR is not directly applicable to the CR-3 Cycle 3 operation. However, an inadvertent one-pump start would decrease the RCS T_{ave} by 2°F to 3°F as compared to 7°F for two-pump start. The power and pressure surges due to a one-pump restart would be proportional to the degree of T_{ave} decrease. Therefore, a one-pump start with three-pump operation is not expected to exceed the overpower safety limit of 112% or the code pressure limit of 2750 psia.

4. Starting the transient at 102% of 2544 MWt would yield about 2350 psia peak pressure during the transient, which is much less than the code limit of 2750 psia.

5. The primary concern for this event is the radioactivity releases. The licensee has analyzed these consequences and states that they are well below the 10 CFR 100 limits.

5.3 References

5-1 CR-3, FSAR, Docket 50-302, FPC

5-2 CR-3, Cycle-3 Reload Report, BAW-1607, Rev. 1, April 1980.

5-3 CR-3, Fuel Densification Report, BAW-1397, August 1973.

5-4 ECCS Analysis of B&W's 177-FA Lowered-Loop NSS, BAW-10103A, Rev. 3, July 1977.

5-5 Letter, J. H. Taylor (B&W) to R. L. Baer (NRC), "LOCA Analysis for B&W's 177-FA Plants With Lowered Loop Arrangement (Category 1 plants) Utilizing a Revised System Pressure Distribution," July 8, 1977.

5-6 Letter, W. P. Stewart (FPC) to R. W. Reid (NRC), "Crystal River-3, Docket No. 50-302, Operating License No. DPR-72, ECCS Small Break Analysis," January 12, 1979.

6.0 Conclusions

We have evaluated the reloading of CR-3 for Cycle 3 operation and the proposed Technical Specification modifications that reflect the new cycle parameters. In the original submittal, the licensee had intended to start Cycle 3 operation at an upgraded power level of 2544 MWt. Consequently, normal operation, transients and accidents have been reanalyzed and reviewed for this increased power level. However, Cycle 3 will start at the same Cycle 2 power level of 2452 MWt.

After evaluating the FPC submittals, we conclude that CR-3 operation at or below 2452 MWt is acceptable.

We have determined that the amendment for Cycle 3 operation at 2452 MWt does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an Environmental Impact Statement, or Negative Declaration and Environmental Impact Appraisal need not be prepared in connection with the issuance of this amendment for Cycle 3 operation at 2452 MWt. We will, however, prepare an Environmental Impact Appraisal in connection with the licensee's request to allow operation of CR-3 at increased power levels up to 2544 MWt. This document will be issued concurrently with any further Commission action concerning operation at this increased power level.

We have concluded, based on CR-3 Cycle 3 operation at 2452 MWt and the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of an accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: August 1, 1980