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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 45 TO FACILITY OPERATING LICENSE NO. NPF-3

TOLEDO EDISON COMPANY

AND

CLEVELAND ELECTRIC ILLUMINATING COMPANY

DAVIS-BESSE NUCLEAR POWER STATION, UNIT 1

DOCKET NO. 50-346

1. Introduction

By letters dated March 5, 1982 (Ref. 1) March 23, 1982 (Ref. 2), June 1, 1982 (Ref. 3) and June 21, 1982 (Ref. 4) the Toledo Edison Company (TECo) made application to modify the Technical Specifications for the Davis-Besse Nuclear Power Station, Unit No. 1, to permit operation for a third cycle. The safety analysis for the previous, second cycle of operation at Davis-Besse 1 is being used by the licensee as a reference for the proposed, third cycle of operation. Where conditions are identical or limiting in the second cycle analysis, our previous evaluation (Ref. 5) of that cycle continues to apply.

1.1 Description of the Cycle 3 Core

The Davis-Besse Unit 1 core consists of 177 fuel assemblies, each of which is a 15x15 array containing 208 fuel rods, 16 control rod guide tubes, and one incore instrument guide tube. Cycle 3 will operate in a feed-and-bleed mode with core reactivity control supplied mainly by soluble boron in the reactor coolant and supplemented by 53 full length control rod assemblies composed of silver-indium-cadmium alloy clad in stainless steel. In addition to the full length control rods, eight axial power shaping rods (APSRs) are provided for additional control of the axial power distribution. The licensed core full power level is 2772 MWt.

2. Evaluation of Fuel System Design

2.1 Fuel Assembly Mechanical Design

The forty-eight Babcock and Wilcox Mark (B&W) B-4A 15x15 fuel assemblies loaded as Batch 5A and 5B at the end of Cycle 2 (EOC 2) are mechanically interchangeable with with Batches 1C, 3, and 4 fuel assemblies previously loaded at Davis-Besse Unit 1. The cladding stress, strain and collapse analyses are bounded by conditions previously analyzed for Davis-Besse Unit 1 or were analyzed specifically for Cycle 3 using methods and limits previously reviewed and approved by the NRC.

2.2 Fuel Rod Design

Although all batches in the Davis-Besse 1 Cycle 3 core utilize the same Mark B-4A fuel, most of the Batch 5 assemblies incorporate a slightly different initial fuel density (95 percent of theoretical density) as compared to the Batch 1C (96 percent), Batch 3 (96 percent), and Batch 4 (94 percent) fuel assemblies. The change is a consequence of using a modified fuel fabrication process. The stability (densification resistance) of all fuel types is similar and the initial active fuel lengths are virtually the same. We regard such changes as within the range of expected fuel design variation and, therefore, find them acceptable.

2.2.1 Rod Internal Pressure

Section 4.2 of the Standard Review Plan (Ref. 6) addresses a number of acceptance criteria used to establish the design bases and evaluation of the fuel system. 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)) is that fuel rod internal gas pressure should remain below normal system pressure during normal operation unless otherwise justified.

Toledo Edison has stated 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 (Ref. 7) rather than a newer B&W code called TACO (Ref. 8). 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 (Ref. 9) 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/MtU. Although we have not examined the comparison, we note that the analyses exceed the expected exposure (36,579 MWd/MtU) in Davis-Besse 1 at EOC 3 for all assemblies. We conclude that the rod internal pressure limits have been adequately considered for Cycle 3 operation.

2.3 Fuel Thermal Design

There are no major changes between the thermal design of the new Batch 5 fuel and previous batches reinserted in the Cycle 3 core. The change in initial fuel density results in a slightly altered nominal value of the linear heat rating (LHR) for the fuel and the maximum LHR value based on centerline melt. We regard these changes as minor and acceptable.

2.3.1 Loss of Coolant Accident Initial Conditions

In addition to the steady-state conditions, the average fuel temperature as a function of linear heat rate and lifetime pin pressure data used in the LOCA analysis (Section 7.2 of the reload submittal) are also calculated with the TAFY code (Ref. 7). Babcock and Wilcox has stated that the fuel temperature and pin pressure data used in the generic LOCA analysis (Ref. 10) are conservative compared to those calculated for Cycle 3 at Davis-Besse Unit 1.

As previously mentioned in Section 2.2.1 of this evaluation, B&W currently has two fuel performance codes, TAFY (Ref. 7) and TACO (Ref. 8), 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 (Ref. 11) indicates that the TAFY code predictions do not produce higher peak cladding temperatures than TACO for all Cycle 3 conditions as suggested in Reference 9. 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 (Ref. 12) a method of resolving this issue which was accepted by the NRC staff (Ref. 13). The method involves the use of reduced LOCA kW/ft limits at low core elevations during the first 50 effective full power days (EFPD) of operation. The licensee has incorporated these changes into the Davis-Besse Unit 1 Technical Specifications to support the operation of the plant at full rated power during Cycle 3. These changes have been reviewed by the NRC staff and have been found acceptable. We conclude that the initial thermal conditions for LOCA analysis have been appropriately considered for Cycle 3 operation.

2.4 Operating Experience

Babcock and 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 as of December 31, 1981 is given on page 4-2 of BAW-1707 (Ref. 1).

2.4.1 Holddown Spring Failures

The upper end fitting of the B&W Mark B-4A fuel assembly contains a holddown spring to accommodate length changes due to thermal expansion and irradiation growth while providing a positive holddown force for the assembly. On May 14, 1980, a failed holddown spring was discovered by remote video inspection at Davis-Besse Unit 1 (Ref. 14). Further examination ultimately identified a total of 19 failed springs in the Cycle 1 fuel assemblies. Subsequent examination of spent fuel assemblies at other B&W reactors revealed a small number of similar failures.

This issue was previously considered in our safety evaluation (Ref. 5) of the Cycle 2 reload. On the basis of the licensee's analysis of the consequences of operating with failed holddown springs, the replacement of all failed and suspect springs, and the licensee's commitment to continued surveillance of the fuel assemblies, we concluded that there was reasonable assurance that the holddown spring issue had been correctly analyzed and did not result in a safety concern for Cycle 2 operation.

An inspection (Ref. 15) of all Cycle 2 assemblies during the current refueling outage revealed a broken holddown spring in a single assembly due to be discharged. Although we do not consider the failed holddown spring issue completely resolved at this time, we conclude that the licensee's continued attention in this area alleviates any safety concern for the proposed Cycle 3 operation.

2.5 Fuel Rod Bowing

The licensee has calculated a fuel rod bowing penalty with a method similar to that approved in Reference 16. The rod bowing magnitude correlation used in that method is approved (Ref. 17), and we conclude that it adequately accounts for gap closure as a function of burnup in the Mark B fuel design. The remaining input assumptions for the rod bowing analysis, and the manner in which the resultant rod bowing penalty is offset, are described in the Thermal-Hydraulic Design section of this evaluation.

3. Nuclear Design

A core loading diagram for Cycle 3 of Davis-Besse Unit 1 is presented in the reload report (Ref. 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 beginning-of-cycle (BOC) are also given. Two sets of parameters are presented; one set for the base design (230 EFPD) and the other set for the alternate design which includes an APSR pull at 200 EFPD and a power coastdown to EOC at 268 EFPD.

The core physics calculations are performed with the PDQ07 code which has been reviewed and approved by the NRC staff. This code has been used for analyses of the previous cycles of Davis-Besse Unit 1. The results of the analyses show small differences between Cycle 2 and Cycle 3 values, occasioned by the difference in design cycle lengths (306 ±10 EFPD for Cycle 2 vs. 230 EFPD for the base design or 268 EFPD for the alternate design for Cycle 3) and by the fact that the core is not yet in its equilbrium configuration. The analysis of shutdown margin shows that 1.53% k/k exists at EOC compared to the required 1.0% 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 physics parameters to be acceptable.

4. Thermal-Hydraulic Design

The thermal-hydraulic performance for Cycle 3, in which the fresh Batch 5 fuel is hydraulically and geometrically similar to the other fuel in the Cycle 3 core, is identical to that of Cycle 2. The thermal-hydraulic design evaluation supporting Cycle 3 operation is based on the methods and models previously used in Cycle 2 as described in References 18 and 19. The design conditions are given in Table 1 of this evaluation and are identical for Cycles 2 and 3.

A rod bow penalty was calculated for each fuel batch using the approved procedure given in Reference 16. The fresh Batch 5 fuel with a rod bow penalty of less then 0.5% was found to be the most restrictive on Departure from Nucleate Boiling Ration (DNBR). However, by taking a 1% credit for the use of a flow area reduction hot channel factor in DNBR calculations, the resulting rod bow penalty is eliminated.

The flux/flow trip setpoint for Cycle 3 operation has been established as 1.07. This setpoint and other plant operating limits are based on criteria that meet the design minimum DNBR limit of 1.30 calculated using the BAW-2 correlation.

The minimum DNBR at 112 percent of full power is 1.79 for Cycle 3 which is the same as for Cycle 2. We find that the thermal-hydraulic design is acceptable since the Cycle 3 and Cycle 2 (previously approved) design conditions are identical and acceptable design methods have been used in the analysis.

5. Accident Analyses

The important kinetics parameters for Cycle 3 have been compared with those from the Final Safety Analysis Report (FSAR) and previous cycles. The initial conditions of the transients in Cycle 3 are bounded by those assumed in the FSAR, the fuel densification report, or by previous cycles. The transient evaluation of Cycle 3 is, therefore, considered to be bounded by previously accepted analyses.

Two sets of bounding values for allowable LOCA peak linear heat rates (LHRs) are given as a function of core height. The first set, which covers the first 50 EFPD, includes reduced LOCA kW/ft limits at low core elevations and are based on the interim LOCA LHR limits. The second set, which covers the balance of the cycle and are identical to the values used in the previous cycle, are the Final Acceptance Criteria LOCA LHR limits. These limits are satisfactorily incorporated into the Technical Specifications for Cycle 3 through the operating limits on rod index, APSR position, and axial power imbalance.

6. Technical Specification Changes

We have reviewed the proposed Technical Specifications for Cycle 3 operation which include the following changes:

(a) The high flux trip setpoint has been increased slightly to 105.1% of rated thermal power from 104.9% with four pumps operating. This eliminates the dual inclusion of the instrument calibration error in the trip setpoint as explained in Ref. 6. This is acceptable since the safety analyses conservatively assume a high flux trip at 112% of rated power which includes calibration and instrument errors. Likewise, the trip setpoint readjustment to 79.6% of rated thermal power with three pumps operating is acceptable.

(b) Two sets of operating limits for the period after 200 EFPD are included depending on whether the APSRs are removed (completely withdrawn) or not. These sets of limits are reflected in the regulating rod insertion limits (3.1.3.6 and Figures 3.1-2a to 3.1-3e), APSR insertion limits

(3.1.3.9 and Figures 3.1-5a to 3.1-5i), and the axial power imbalance limits (3.2.1 and Figures 3.2-1a to 3.2-2e). We have approved two operating schemes and associated parameters; one for the base design (230 EFPD) and the other for the alternate design which includes an APSR withdrawal at 200 EFPD and a power coastdown to EOC at 268 EFPD. Based on the acceptability of either one of these two schemes, and on the fact that previously approved techniques and models were used to derive the revised Technical Specification parameters, we conclude that the new specifications are acceptable.

(c) The power level cutoff at 92% full power has been eliminated from the regulating group position limits of Figures 3.1-2a through 3.1-2e. This has been done and approved for other recent B&W operating reactors and is, therefore, acceptable.

7. Environmental Consideration

We have determined that the amendment 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.

8. Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of an accident of a type different from any evaluated previously, and does not involve a significant reduction in a margin of safety, 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: July 28, 1982

The following NRC personnel have contributed to this Safety Evaluation: Laurence I. Kopp, Harry Balukjian and John Voglewede.

TABLE 1

DAVIS-BESSE CYCLES 2 AND 3
THERMAL-HYDRAULIC DESIGN CONDITIONS

Design power level, MWt	2772
System pressure, psia	2200
Reactor coolant flow, gpm	387,200[b]
Reactor coolant flow, % design	110
Vessel inlet/outlet coolant temp., 100% power, F	557.7/606.3
Ref design radial-local power peaking factor	1.71
Ref design axial flux shape	1.5 cosine with tails
Hot Channel factors	
Enthalpy rise (F_q)	1.011
Heat flux (F''_q)	1.014
Flow area	0.98
Avg heat flux, 100% power, Btu/h-ft^2	1.89×10^5[a]
Max heat flux, 100% power, Btu/h-ft^2	4.85×10^5[a]
CHF correlation	BAW-2
Minimum DNBR (at 112% power)[b]	1.79

(a) With thermally expanded fuel rod OD of 0.43075 inch.

(b) Telecon, L. Young, Toledo Edison, to A. DeAgazio, NRC, June 7, 1982.

References

1. R. P. Crouse (Toledo Edison) letter dated March 5, 1982 (No. 787) transmitting Davis-Besse Nuclear Power Station, Unit 1 Cycle 3 Reload Report (BAW-1707) dated February 1982.

2. R. P. Crouse (Toledo Edison) letter dated March 23, 1982 (No. 789) transmitting revision to BAW-1707.

3. R. P. Crouse (Toledo Edison) letter dated June 1, 1982 (No. 822) transmitting revision to BAW-1707.

4. R. P. Crouse (Toledo Edison) letter dated June 21, 1982 (No. 829) submitting additional information.

5. W. V. Johnston (NRC) memorandum for R. W. Reid (NRC) on "Davis-Besse Unit 1 Reload" dated August 19, 1980.

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

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

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

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

10. W. L. Bloomfield, et. al., "ECCS Analysis of B&W's 177-FA Raised-Loop NSS," Babcock and Wilcox Company Report BAW-10105, Revision 1, June 1975.

11. 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.

12. J. H. Taylor (B&W) letter to L. S. Rubenstein (NRC) dated September 5, 1980.

13. L. S. Rubenstein (NRC) letter to J. H. Taylor (NRC) dated October 28, 1980.

14. T. D. Murray (Toledo Edison) letter to J. G. Keppler (NRC/Reg. III) dated May 23, 1980.

15. R. P. Crouse (Toledo Edison) letter to J. F. Stolze (NRC) dated June 28, 1982 (No. 832).

16. 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.

17. R. O. Meyer (NRC) memorandum for D. F. Ross (NRC) on "Revised Coefficients for Interm Rod Bowing Analysis," dated March 2, 1978.

18. Davis-Besse Unit 1 Fuel Densification Report, BAW-1401, Babcock and Wilcox, Lynchburg, Virginia, April 1975.

19. Attachment 1 to Application to Amend Operating License for Removal of Burnable Poison Rod and Orifice Rod Assemblies, BAW-1489, Rev. 1, Babcock and Wilcox, Lynchburg, Virginia, May 1978.