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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 80 TO FACILITY OPERATING LICENSE NO. NPF-3

TOLEDO EDISON COMPANY

AND

THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

DAVIS-BESSE NUCLEAR POWER STATION, UNIT NO. 1

DOCKET NO. 50-346

1.0 INTRODUCTION

By letter dated July 20, 1984 (Ref. 1), Toledo Edison Company made application to modify the Davis-Besse Nuclear Power Station Technical Specifications to permit operation for a fifth cycle. The analysis performed and the resulting modifications to the Technical Specifications are described in the Cycle 5 reload report (Ref. 2). The licensee has used the fourth cycle of operation at Davis-Besse as the reference cycle for the proposed fifth cycle of operation. Where conditions are identical or limiting in the fourth cycle analysis, our previous evaluation (Ref. 3) of that cycle continues to apply.

1.1 Description of the Cycle 5 Core

The Davis-Besse Cycle 5 core will consist 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. The length of Cycle 5 is expected to be 390 effective full power days (EFPD) of operation. The reference cycle for the nuclear and thermal-hydraulic design of Cycle 5 is Cycle 4 which was scheduled for 280 EFPD. The Cycle 5 design is characterized by only eight fuel assemblies being cross core shuffled so as to minimize any carryover effects from tilts encountered in previous cycles. The licensed power level remains at 2772 MWt.

Cycle 5 will operate in bleed-and-feed mode with core reactivity control supplied mainly by soluble boron in the reactor coolant and supplemented by 53 full length control rod assemblies (CRAs). In addition, eight axial power shaping rods (APSRs) are provided for additional control of the axial power distribution. The Cycle 5 loading will require 64 new fuel assemblies (Batch 7) and the reinsertion of one previously discharged fuel assembly. The 64 new fuel assemblies are fabricated by Babcock and Wilcox (B&W) but contain fuel pellets manufactured by General Electric (GE). Due to the increased length of Cycle 5, additional core reactivity is necessary. This increased reactivity will be controlled in part by 64 burnable poison rod assemblies (BPRAs) located in the fresh fuel. Batch 7 is comprised of the Mark-B5 design which is identical in concept to the Mark-B4 currently used. The only change is to the upper end fitting which has the retention mechanism built in for BPRA holddown. This change is to eliminate the need for retainer assemblies.

2.0 EVALUATION

2.1 Fuel System Design

The 64 BPRA B&W Mark-B5 fuel assemblies loaded as Batch 7 at end of Cycle 4 (EOC 4) are mechanically interchangeable with type Mark-B4 Batches 1E, 4B, 5B and 6 fuel assemblies previously loaded at Davis-Besse. The Mark-B5 upper end fitting provides four open slots that align and guide the movement of the holddown spring, spring retainer, and a new Mark-B5 BPRA spider. The Mark-B5 design also contains a redesigned holddown spring made from Inconel 718 material which provides added margin over the Mark-B4 spring design made from Inconel 750 (Ref. 3). The licensee stated that the Mark-B5 upper end fitting has been tested extensively, both in air and in over 1000 hours of simulated reactor environment, to determine analytical input and to assure good incore performance. The licensee intends to continue visual inspection programs on the new fuel holddown springs (Ref. 4).

The cladding stress, strain and collapse analyses are bounded by conditions previously analyzed for Davis-Besse or were analyzed specifically for Cycle 5 using methods and limits previously reviewed and approved by the NRC. End-of-life fuel rod internal pressures have also been analyzed using previously-approved methods and limits.

The licensee stated (Ref. 4) that there is no change in analysis methodology for fuel rod pin pressure calculations from Cycle 4 to Cycle 5. The licensee further stated that the calculated results show that the fuel rod pressure remains below system pressure at rod exposure up to 45,000 MWd/MTU. We find this acceptable.

For the LOCA analysis (Section 7.2 of Ref. 2) the volume-averaged fuel temperature and fuel rod internal pressure were calculated for Cycle 5 as a function of linear heat rating. The licensee has stated that these conditions are bounded by those used in the generic LOCA analysis for Davis-Besse.

The licensee has stated that the analytical methods which were used and accepted for Cycle 4 reload have also been used to support the proposed amendment. These methods (Ref. 5), including the TACO-2 fuel performance code and the revised cladding models in the Emergency Core Cooling System (ECCS) code package, do not differ from the analytical methods used and accepted for previous cores to demonstrate conformance to acceptance criteria and NRC regulations. The approved TACO-2 code is used to determine the margin for centerline melting and other design calculations for fuel Batches 5B, 6 and 7. The ECCS analysis utilizes the TACO-2 code and incorporates cladding rupture, strain, and flow blockage models based upon data presented in NUREG-0630 (Ref. 6).

2.2 Nuclear Design

To support Cycle 5 operation of Davis-Besse, the licensee has provided analyses (Ref. 2) using analytical techniques and design bases established in B&W reports that have been approved by the NRC staff. The validity of the methods also has been reinforced through predictions of a number of cycles for this and other reactors. The licensee has provided a comparison of the core physics parameters (Ref. 2) for Cycles 4 and 5 as calculated with these techniques. We reviewed the characteristics compared to previous cycles, and find them acceptable for use in the Cycle 5 accident and transient analysis, as described in Section 2.4 of this evaluation.

The Cycle 5 design cycle length is 390 days, whereas the Cycle 4 design length was 280 days. The licensee stated that the analytical methods are the same for Cycle 5 as for the reference Cycle 4. The changes in the Cycle 5 physics parameters reflect the change in core loading philosophy. In going to 18-month cycles, the transition to a low leakage core was incorporated. This scheme loads the fresh fuel in a checkerboard pattern with the twice burned fuel in the core interior and loads the once burned fuel on the core periphery. This scheme and the use of the BPRAs produces a flatter radial power distribution causing the changes in reactivity when compared to Cycle 4. No significant operational or procedural changes exist for Cycle 5 with regard to axial or radial power shape, xenon, or tilt control. The Cycle 5 exposure dependent Quadrant Power Tilt limit as presented in Table 8-2 of Ref. 2 was used in the analysis. This shows that the Beginning of Cycle (BOC) steady state Quadrant Power Tilt limit using the incore detector system must be updated at 50±10 EFPD.

Due to the differences in design cycle lengths, the critical boron concentrations for Cycle 5 differ from those of Cycle 4. Because of different isotopic distributions, Cycle 5 control rod worths, ejected rod worths, and stuck rod worths differ from those of Cycle 4. The licensee took into account ejected rod worths and their adherence to shutdown margin requirements in the development of rod position limits for Cycle 5. The licensee presented an analysis of shutdown margin adequacy as a function of predicted control and stuck rod worths. This analysis allowed for a 10 percent uncertainty on net rod worth and for flux redistribution. It shows margin in excess of requirements.

We, therefore, conclude that the licensee has demonstrated adequate provision of shutdown margin for Cycle 5. In addition, control rod worth measurements are made during startup tests. These will confirm the adequacy of predicted control rod worths.

We find the nuclear design of Cycle 5 to be acceptable.

2.3 Thermal-Hydraulic Design

The thermal-hydraulic performance for Cycle 5, in which the fresh Batch 7 fuel is hydraulically and geometrically similar to the other fuel in the Cycle 5 core, is identical to that of Cycle 4. The introduction of the Mark-B5 upper end fitting does not affect either the core flow rate or the thermal-hydraulic performance. The introduction of BPRAs increases the core flow available for heat transfer by reducing the core bypass flow rate from 10.7 to 8.1%. This reduced bypass flow rate conservatively has been neglected for Cycle 5. The thermal-hydraulic design evaluation supporting Cycle 5 operation is based on the methods and models previously used in Cycle 4 as described in References 8 and 9. The design conditions are given in Table 1 and are identical for Cycles 4 and 5.

A rod bow topical report (Ref. 7) was submitted and approved (Ref. 8) before the last fuel cycle. This report addressed the mechanisms and resulting local conditions of the rod bow. The conclusion was that the rod bow penalty is insignificant and is offset by the reduction in power production capability of the fuel assemblies with irradiation. Therefore, there is no resulting rod bow penalty for Cycle 5.

The flux/flow trip setpoint for Cycle 5 has been established as 1.068 and was 1.069 for Cycle 4. This setpoint and other plant operating limits are based on the design minimum DNBR limit of 1.30 calculated using the BAW-2 correlation. It is noted that the design flow for the reload analysis is 387,200 gpm which is 110% of the design reactor coolant system flow. The latest measured reactor coolant system flow was 404,308 gpm (Ref. 4) which provides a 4.4% margin of flow.

The minimum DNBR at 112 percent of full power is 1.79 for Cycle 5 which is the same as for Cycle 4. We find that the thermal-hydraulic design is acceptable since the Cycle 5 and Cycle 4 (previously approved) design conditions are identical and acceptable design methods have been used in the analysis.

2.4 Accident and Transient Analysis

Acceptability of core thermal, thermal-hydraulic, and kinetics parameters, including the reactivity feedback coefficients and control rod worths, was discussed in Sections 2.2 and 2.3. The licensee concluded, by examination of the Cycle 5 values of these parameters with respect to acceptable previous cycle values, that transients and accidents for Cycle 5 are bounded by previously accepted analyses.

The licensee stated that each FSAR accident analysis was examined with respect to changes in the Cycle 5 parameters to determine the effects of the Cycle 5 reload and to ensure that thermal performance during anticipated transients is not degraded. A generic loss-of-coolant accident (LOCA) analysis for B&W 177 fuel assembly raised-loop nuclear steam systems (NSSs) was performed by the licensee using the Final Acceptance Criteria

ECCS Evaluation Modes (Ref. 9). The combination of average fuel temperature as a function of linear heat rate (LHR) and the lifetime pin pressure data used in the LOCA limits analysis was found to be conservative compared to those calculated for this reload.

The licensee's accident and transient analysis, reported in Section 7 of Ref. 2, was reviewed and found to have no significant differences from the previously accepted analysis presented for Cycle 4, with the exception of considering the effect of higher burnup on rod internal pressure changes and release of volatile fission products into the pellet-clad gap.

To assess the effect of higher burnup, we evaluated, independently and in accordance with the methodology of Regulatory Guide 1.25, the doses from a postulated fuel handling accident inside containment. Even though the conditions at the end of Cycle 5 will be beyond the bases stated in the Guide, this methodology continues to be conservative if the effect of higher burnup on the rod internal pressure and on the fraction of volatile radioactive fission products in the pellet-clad gap of the highest power assembly is considered appropriately. Ref. 2 shows that the highest power assembly is a freshly exposed Batch 7 assembly. Therefore, the case to be considered is an assembly at about 13,100 MWd/MTU at the highest allowable linear heat generation rate, 18.4 KW/ft. The assumptions used by the NRC staff and the results of the calculation are given in Table 2. The results show that the fuel handling delay to 72 hours from shutdown and site related parameters are adequate to mitigate the consequences of this accident.

The licensee and the NRC staff have considered the factors dependent upon power level (2772 MWt) and burnup (peak assembly discharge exposure of 41,000 MWd/MTU) that impact the radiological consequences of accidents. We find that operation for Cycle 5 with the extended burnup described in the licensee's application is acceptable.

2.5 Technical Specification Modifications

The pertinent Technical Specifications have been revised for Cycle 5 operation to account for changes in power peaking and control rod worths as discussed in Sections 2.2 and 2.4. We have reviewed these changes as proposed in Reference 2 and find them acceptable.

3.0 ENVIRONMENTAL CONSIDERATION

This amendment involves a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. We have determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

4.0 CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) 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: December 13, 1984

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REFERENCES

1. Letter R. P. Crouse (Toledo Edison Company) to J. F. Stolz (NRC) dated July 20, 1984.

2. "Davis-Besse Nuclear Power Station Unit 1, Cycle 5 - Reload Report," Babcock & Wilcox Company, BAW-1827 (April 1984). Attachment to Reference 1 above.

3. Letter J. F. Stolz (NRC) to R. P. Crouse (Toledo Edison Company), "Amendment No. 61 to Facility Operating License No. NPF-3; Cycle 4 Operation," dated September 21, 1983.

4. Letter R. P. Crouse (Toledo Edison Company) to J. F. Stolz (NRC) dated October 19, 1984.

5. Letter R. P. Crouse (Toledo Edison Company) to J. F. Stolz (NRC) dated May 6, 1983 transmitting "Bounding Analytical Assessment of NUREG-0630 on LOCA and kW/ft Limits," B&W Document No. 77-1142162-00.

6. D. A. Powers and R. O. Meyer, "Cladding Swelling Models for LOCA Analysis," U.S. Nuclear Regulatory Commission Report NUREG-0630 (April 1980).

7. "Fuel Rod Bowing in Babcock and Wilcox Fuel Design," Babcock and Wilcox Company, BAW-10147P (Proprietary) (April 1981).

8. Letter C. O. Thomas (NRC) to J. H. Taylor (B&W) dated February 15, 1983.

9. "ECCS Evaluation of B&W's 177-FA Raised-Loop NSS", BAW 10105, Rev. 1, Babcock and Wilcox Company (July 1975).

TABLE 1

DAVIS-BESSE CYCLES 4 AND 5

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
Average 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, G. Bradley, Toledo Edison, to A. De Agazio, NRC, September 1, 1983.

TABLE 2

CALCULATION OF THE FUEL HANDLING ACCIDENT INSIDE CONTAINMENT

Power level	2772 MW_t
Peaking factor	2.8
Fuel failures	1 module of 177
Fractional release of volatiles to environment before containment isolation	20 percent
Shutdown time	72 hours

Atmospheric Diffusion and Transport Relative Concentration, X/Q (sec/m^3)

Exclusion Area Boundary	0-2 hours	2.2×10^{-4}
Low Population Zone	0-8 hours	9.6×10^{-6}

Doses (Rem)	Thyroid	Whole Body
EAB	21	.4
LPZ	0.9	.1