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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 46 TO FACILITY OPERATING LICENSE NO. NPF-73

DUQUESNE LIGHT COMPANY

OHIO EDISON COMPANY

THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

THE TOLEDO EDISON COMPANY

BEAVER VALLEY POWER STATION, UNIT 2

DOCKET NO. 50-412

## 1.0 INTRODUCTION

By letter dated October 15, 1991, Duquesne Light Company (DLC) proposed a change to the Beaver Valley Power Station, Unit No. 2 Appendix A Technical Specifications (TS). The proposed change would increase the allowable control rod drop time specified in Limiting Condition for Operation (LCO) 3.1.3.4 to 2.7 seconds from 2.2 seconds. This change would allow the use of the VANTAGE 5 Hybrid (VANTAGE 5H) fuel design which incorporates a smaller thimble tube diameter. The smaller thimble tube diameter results in a slightly greater rod drop time. In addition, DLC proposed changes to certain Bases sections to reflect the modified DNB design basis which uses the new Westinghouse correlation, WRB-1, for predicting critical heat flux and the MINI Revised Thermal Design Procedure (MINI-RTDP).

Additional supporting information was submitted by letters dated January 27, 1992, and February 25, 1992. The additional information did not change the initial proposed no significant hazards consideration determination.

## 2.0 BACKGROUND

The VANTAGE 5H fuel design evolved from the VANTAGE 5, Optimized Fuel Assembly (OFA), and Standard (STD) fuel assembly designs. The features of the VANTAGE 5H fuel assembly include Zircaloy Grids, Reconstitutable Top Nozzles, Debris Filter Bottom Nozzles (DFBNs), Snag Resistant Grids and Standardized Fuel Pellets. In addition, the VANTAGE 5H fuel uses Integral Fuel Burnable Absorber and Axial Blanket design features.

The VANTAGE 5H features were previously reviewed and approved by the NRC in Westinghouse topical report WCAP-10444-P-A, "Reference Core Report VANTAGE 5 Fuel Assembly, Addendum 2." During the review of VANTAGE 5 fuel design described in WCAP-10444-P-A, the staff identified conditions to be resolved for the licensees using the VANTAGE 5 fuel design. Since the VANTAGE 5H fuel

9205070150 920501
PDR ADOCK 05000412
P PDR

design adopts some features from the VANTAGE 5 fuel design, the staff evaluation, provided below, of the Beaver Valley Unit 2 Cycle 4 reload with the VANTAGE 5H fuel design and the associated TS changes, addresses each of those conditions listed in the safety evaluation of WCAP-10444-P-A that affect the Beaver Valley 5H fuel. This approach is the same as that used by the staff in their review of the Beaver Valley Unit 1 Cycle 8 reload with VANTAGE 5H fuel.

## 3.0 EVALUATION

The conditions from WCAP-10444-P-A that were found to be applicable to Beaver Valley Unit 2 Cycle 4 reload with the VANTAGE 5H fuel design were reviewed by the staff. The conditions reviewed were Statistical Convolution, Irradiation Demonstration Programs, MINI-Revised Thermal Design Procedure, Transient Analysis, Reactor Coolant Pump Shaft Seizure accident, and Loss Of Coolant Accident (LOCA) Analysis.

### 3.1 Statistical Convolution

In the safety evaluation (SE) for WCAP-10444-P-A, it was stated that the statistical convolution method was not approved for evaluating the uncertainties associated with the axial fuel rod to nozzle gap for the VANTAGE 5 fuel design. In the case of VANTAGE 5H fuel, the convolution method also would also not be an approved method for evaluating fuel rod to nozzle gap. The staff's review confirmed that the convolution method was not used in the Cycle 4 analysis.

The VANTAGE 5H rod gap has been increased in Cycle 4 by the use of the Reconstitutable Top Nozzle which has a reduced nozzle plate thickness. The thickness reduction results in additional space for fuel rod growth. Section 2.1.1 of WCAP-10444-P-A indicates that the fuel rod gap growth is acceptable for the VANTAGE 5 fuel design. It is therefore acceptable for the Unit 2 Cycle 4 reload.

### 3.2 Irradiation Demonstration Program

The SE for WCAP-10444-P-A required that an irradiation demonstration program be performed to confirm the VANTAGE 5 fuel performance. DLC has done this in that the evaluation of the VANTAGE 5H grid performance is based on: (1) the extensive design and irradiation experience with previous grid designs, and (2) full grid testing conducted with the VANTAGE 5H grid design. DLC indicated that there were fuel assembly demonstration programs that entailed inserting optimized fuel assemblies (OFA) containing Zircaloy grids into 14x14, 15x15, and 17x17 cores. The satisfactory performance of these demonstration assemblies resulted in OFAs with Zircaloy grids being used in reloads and operating successfully since the early 1980s in many Westinghouse cores.

3.3 MINI - Revised Thermal Design Procedure (MINI RTDP)

The existing thermal-hydraulic analysis of the 17x17 STD fuel used in the Beaver Valley Unit 2 plant is based on standard thermal and hydraulic methods and the W-3 (R-Grid) DNB correlation as described in the Beaver Valley Unit 2 Updated Final Safety Anaslysis Report (UFSAR). The DNB analysis for the mixed core (17x17 STD and VANTAGE 5H fuel assemblies) has been modified to incorporate the WRB-1 DNB correlation and a conservative application of the Revised Thermal Design Procedure (MINI-RTDP).

With MINI-RTDP methodology, peaking factor uncertainties are combined statistically with the DNB correlation uncertainties to obtain the overall DNBR uncertainty factor. The uncertainty factor is then used to define the design limit DNBR that satisfies the DNB design criterion. This criterion states that there is at least a 95% probability at a 95% confidence level that DNB will not occur on the most limiting fuel rod for any Condition I or II event. MINI-RTDP excludes the uncertainties on primary system parameters (reactor power, flow, temperature and pressure) from the statistical combination process. These uncertainties will be used to offset the nominal values of these parameters in transient analyses, resulting in more adverse initial conditions.

The MINI-RTDP methodology was previously reviewed and approved by the NRC for the Westinghouse core reload application and, therefore, is acceptable for the Beaver Valley Cycle 4 reload calculations.

3.4 Transient Analysis

The reanalysis and evaluation of non-LOCA transients summarized below takes into consideration the effects of upgrading to VANTAGE 5H fuel and the deletion of thimble plugs. The evaluation bounds the case where some or all of the thimble plugs are present and supports up to 20% steam generator tube plugging.

The major effect of changing from STD 17x17 fuel to VANTAGE 5H fuel is the increased design Rod Control Cluster Assembly (RCCS) drop time. The slower drop time (from 2.2 to 2.7 seconds) is due to the decrease in the inner diameter of the VANTAGE 5H fuel thimble tube, as compared to the STD 17x17 fuel thimble, by 0.008 inches.

The impact of removing the thimble plugs is an increase in the core bypass flow from 4.5% to 6.5%. This increase in bypass flow results in an equivalent decrease in the flow through the core active fuel region.

The DNB limited events have either been re-analyzed incorporating the decrease in core flow and increase in rod drop time, or have been evaluated such that the results of previous analyses remain valid. In addition, the reactor core thermal limit curves have not changed and the current Technical Specification Overtemperature $\Delta T$ and Overpower $\Delta T$ setpoints remain valid.

For the events that are not DNB related, or for which the prevention of DNB is not the only safety criterion, the effects of decreased core flow and increased rod drop time have been evaluated with respect to the applicable acceptance criteria; e.g. core ex[illegible]lant temperature. The evaluation demonstrated that acceptance crit[illegible]ontinue to be met.

DLC has reanalyzed those events that are sensitive to increased design RCCA drop time. The events re analyzed include loss of forced reactor coolant flow, locked rotor, control rod bank withdrawal from subcritical, and rod ejection. The WRB-1 correlation and the MINI-RTDP methodology were used to evaluate transient DNBRs for both the STD and VANTAGE 5H fuels. The transient reanalyses and evaluations demonstrate that the applicable safety analysis acceptance criterion continue to be met for the intended fuel design and are therefore acceptable.

3.5 Reactor Coolant Pump Shaft Seizure

The evaluation for locked rotor event is similar to the evaluation on Locked Rotor for Beaver Valley Unit 1. DLC has evaluated the reactor coolant pump shaft (locked rotor) accident based on the failure of the peak cladding temperature of 2700 degrees F. DLC has concluded that there is no fuel failure and the cooling was maintained since the calculated peak clad temperature (1870 degrees F) remained much less than 2700 degrees F and the amount of Zirconium-water reaction was small.

DLC found that 18% of the fuel rods could experience DNB with minimum DNBRs less than the safety analysis DNBR limit. This was calculated based on a fuel rod power census which is conservative for Cycle 4 operations and is expected to bound future cycles. It was concluded that the integrity of the primary coolant system is not endangered and the core remains intact with no consequential loss of core cooling capability.

The amount of fuel failure is used to assess the radiological consequences of the Locked Rotor event. Since the acceptable fuel failure criterion of 95/95 DNBR limit is used for DNBR analysis, it is concluded that the reactor coolant pump shaft seizure accident is satisfactorily addressed for VANTAGE 5H fuel.

3.5.1 Radiological Consequences of a Reactor Coolant Pump Shaft Seizure

DLC performed calculations to demonstrate that the offsite dose consequences of a locked rotor accident for the new fuel would still be within the NRC Standard Review Plan acceptance criteria (25% of 10 CFR Part 100 dose reference values) and that the control room operator doses would still meet the requirements of General Design Criterion (GDC) 19 of 10 CFR Part 50, Appendix A. The October 15, 1991 submittal indicated that the initial recalculation of the control room operator doses resulting from the locked-rotor event showed doses exceeding GDC 19. However, a subsequent evaluation, dated January 27, 1992, performed by DLC showed that the control room operator doses are within the dose limits of GDC 19. This analysis of the control room operator doses relies upon a modified analysis of the atmospheric diffusion associated with the various release points at Beaver Valley 2. On February 25, 1992, DLC submitted a report prepared by Halliburton NUS Environmental Corporation which provides the basis for the revised meteorological parameters used in the analysis. Also on February 25, 1992, DLC submitted an analysis of the offsite radiological consequences of the locked-rotor event.

The staff has completed its review of the offsite consequences of the locked-rotor event. The staff considered that radioactivity would be released from the steam generator safety valves and/or the power operated relief valves under the following conditions:

1) radioactivity in the steam and secondary coolant of the steam generators at the technical specification (TS) concentrations;

2) a primary to secondary leak rate at the TS limit;

3) primary coolant at the TS limit for radioactivity concentration; and

4) a concurrent iodine spike at the time of the locked-rotor event.

The staff has evaluated the offsite consequences based upon an accident duration of 8 hours. The staff has determined independently that the dose consequences at the exclusion area boundary (EAB) and the low population zone (LPZ) are acceptable.

The staff, however, has not completed reviewing the control room operator dose analysis. DLC's analysis for the control room utilized X/Q values that were derived using the methodology presented by J. V. Ramsdell in NUREG/CR-5055, "Atmospheric Diffusion for Control Room Habitability Assessment" and in Proceedings of the 21st DOE/NRC Air Cleaning Conference (Conference 900813, NUREG/CP-0016, Vol 2, pp 714-729, 1990) "Alternatives to Current Procedures Used to Estimate Concentration in Building Wakes." (The Proceedings contained a typographical error but Mr. Ramsdell provided corrected information by letter to Halliburton NUS).

The staff has reviewed the Halliburton NUS report and has determined that insufficient information is included to support an independent staff evaluation of the X/Q values that were derived. Therefore, additional meteorological information is required; DLC shall submit, not later than July 15, 1992, the site-specific hourly meteorological data that were used to determine the X/Q values used in the calculation of doses to the control room operators. Following submittal of this information, the staff will complete the evaluation of the revised X/Q values and the resulting control room doses and will issue a supplemental safety evaluation presenting the results of their conclusions.

However, based upon a preliminary review, the staff has concluded that the approach DLC used to calculate the doses to the control room operators is generally-acceptable. Therefore, DLC's evaluation of the radiological consequences to the control room operators is accepted for Cycle 4 only.

3.6 LOCA Analysis

The LOCA analysis was evaluated to determine whether the thimble plug removal and the VANTAGE 5H low pressure drop Zircaloy grid fuel feature had an effect on the results.

The following LOCAs were evaluated:

| | |
|---|---|
| 15.6.5 | Large Break LOCA |
| 15.6.5 | Small Break LOCA |
| 15.6.3 | Steam Generator Tube Failure |
| [illegible] 4 | Blowdown Reactor Vessel and Loop Forces |
| [illegible].6.5 | Post LOCA Long-Term Cooling, Subcriticality Evaluation |
| 6.3.2.5 | Hot Leg Switchover to Prevent Potential Boron Precipitation/Long-Term SI Verification (UFSAR Table 6.3-7) |
| 15.3.4 | LOCA Containment Integrity |

DLC indicated that in most cases the Cycle 4 modifications are supported by the existing licensing basis safety analyses. In these cases it was concluded that specific safety analyses are not sensitive to the fuel and thimble plug removal upgrades, or have otherwise incorporated bounding analyses assumptions. In all cases, the LOCA evaluation and reanalysis demonstrate that the applicable safety criteria are met and, therefore, the staff finds the evaluation acceptable.

3.7 Overall

The NRC staff has reviewed DLC's proposed Technical Specification change to support operation of Beaver Valley Unit 2 with the VANTAGE 5H fuel design. Based on the approved generic topical reports and plant specific analysis, the NRC staff finds the use of the VANTAGE 5H fuel design acceptable for cycle 4 only. The staff will evaluate the acceptability of operation beyond cycle 4 based upon review of the additional meteorological information requested. The staff's findings will be presented in a supplemental safety evaluation. Therefore, the proposed change to TS LCO 3.1.3.4, which would increase the allowable control rod drop to 2.7 seconds from 2.2 seconds, is acceptable for cycle 4 only, and a footnote noting this limitation has been added to TS page 3/4 1-23. The footnote also notes that approval for operation with the stated allowable control rod drop time beyond cycle 4 is pending. This footnote has been discussed with a DLC representative, and it is acceptable with DLC. DLC should provide the requested meteorological information no later than July 15, 1992.

4.0 <u>STATE CONSULTATION</u>

In accordance with the Commission's regulations, the Pennsylvania State official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has 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 radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding (57 FR 2592). Accordingly, the 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 the amendment.

6.0 CONCLUSION

The Commission has 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, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: S. Brewer
J. Hayes

Date: May 1, 1992