INTERIM REPORT ON FUEL DENSIFICATION

FOR THE

METROPOLITAN EDISON THREE MILE

ISLAND NUCLEAR STATION UNIT 1

APRIL 1973

1. INTRODUCTION

At the present time, the Regulatory Staff is evaluating topical reports BAW-10055, "Fuel Densification Report", and BAW-1388, "Fuel Densification Report for Oconee Unit 1". It is anticipated that by May 1, 1973, the evaluation will be completed of the application of B&W analytical models in accordance with the guidelines set forth in the "Technical Report on Densification of Light Water Reactor Fuels", USAEC, of November 14, 1972.

The purpose of this interim report is to: (1) provide the Staff with sufficient information to compare Three Mile Island - Unit 1 (TMI-1) with Oconee 1 and to describe any changes in methods or application thereof in the final report for TMI-1, and (2) present a preliminary evaluation of the rated power capability of TMI-1 even considering fuel densification effects.

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2. SUMMARY

Based on the analysis performed for TMI-1 using the methods outlined in BAW-10055, including a change in the power spike model and a change in input to this model (see Section 3), the following conclusions are made assuming that the fuel pellets densify up to 96.5% theoretical density.

1. Cladding collapse will not occur prior to 21,000 effective full power hours of operation (greater than one fuel cycle).
2. The mechanical performance of B&W fuel rods will not be impaired.
3. The interim acceptance criteria for ECCS are met.
4. The reactor can be safely operated at a power level of 2535 MWt with minor modifications to the reactor protection system setpoints.
    a. A reduction in the overpower trip setpoint from 114 to 112% of 2535 MWt.
    b. A reduction in allowable imbalance limits.
5. The increase in power peaking from gaps in fuel rods due to the assumed densification can be compensated for by base load operation as opposed to load following.

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3. RESULTS

3.1 Power Spike Model

The power spike model described in BAW-10055 will be used as described, except for one minor modification to the model and one change to input as described in the following subsections which are revised versions of the same subsections in BAW-10055.

2.3.4. Calculations of the Maximum Void Size in Axial Interval

The equation for the determination of the maximum gap size as a function of axial position has been modified from

$$\Delta L = \left[\left[\frac{0.965 - \delta_i + 2\sigma(z)}{2} \right] + 0.004 \right] L(z)$$

to the following

$$\Delta L = \left[\frac{0.965 - \delta_i}{2} + 0.004 \right] L(z)$$

The term $\sigma(z)$ has been eliminated from the determination of maximum axial gap as a result of discussions with the staff and their evaluation of the Point Beach 2 Nuclear Plant Analysis.

2.4.1 Power Spikes Due to a Single Gap and

2.4.2 Power Spikes Due to Co-Planer Gaps

The calculational methods described in BAW-10055 will be changed to reflect the use of the DOT, Sn transport theory code in both X-Y and R-Z geometries. The power spike in the adjacent rods will be obtained using DOT in X-Y geometry and the variation with gap length will be obtained

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using R-Z geometry. The change redefines the power spike as the power increase in the adjacent pin rather than that used in the third sentence in subsection 2.4.1., "(i.e., the percentage increase at the edge of the surrounding fuel region)". B&W has known that this definition of the power spike due to a single gap was conservative and unrealistic and the true power increase in the adjacent pin has been determined. The results have been compared to those reported by Brookhaven National Laboratory in their December 1972 report, "Peaking Factors in Pressurized Water Reactors with Fuel Densification". The last figure in this report (un-numbered), "Correction Factor F on Finite Gap Lengths", shows BNL results that are less than those values calculated by B&W for the same fuel enrichment. Therefore, the B&W method is conservative compared to the BNL method.

The DOT transport theory code in X-Y geometry will also be used instead of the PDQ07 diffusion theory code to determine the power spike in a center rod due to more than one gap at the same axial height in a surrounding array of fuel rods. The use of the transport calculation provides a more accurate description of the physical system than does the diffusion theory calculation.

For the TMI-1 reactor these correction factors will be calculated for the second zone fuel enrichment (2.75 w/o as-built), since this zone experiences the highest power peaking of the three fuel enrichments in Core 1.

The preliminary data indicate that the power spike factor shown in Figure 2-12 for the TMI-1 reactor will be about 1.11 to 1.20 at 140 inches.

The following discussion on several additional sections of BAW-10055 is presented to indicate the manner in which the final analysis for TMI-1 will be performed.

3.2 Thermal Analysis

The application of the power spike factor in the analysis of fuel temperatures and DNBR analysis will be that approved by the staff for BAW-10055. The as-built data for TMI-1 will, of course, be used in the evaluation.

The preliminary evaluation has shown that the DNBR loss due to densification from about 92.5 to 96.5% T.D. results in a power peaking margin reduction of about 4 to 5%.

The fuel melting criterion will be about 21 kW/ft based on a maximum first cycle burnup of 20,800 MWd/mtU, which includes a 10% uncertainty margin.

These effects will be compensated for by reducing of the overpower trip setpoint from 114 to 112% of 2535 MWt and changing the operational philosophy from load following to base loaded operation.

3.3 Nuclear Analysis

The change in operation from load following to a base loaded philosophy will reduce power peaks more than enough to compensate for the effects derived in section 3.2. This change in operation will allow greater imbalance trip limits, for TMI-1 than for Oconee I in spite of the fact that the average density is 92.5% compared to 93.5% T.D. and the fuel enrichment considered in the determination of the power spike factor is 2.75 w/o compared to 2.15 w/o for Oconee I.

3.4 Safety Analysis

The ground rules and acceptance criteria adopted for the Oconee 1 report will be adhered to in the TMI-1 analysis.

3.5 Mechanical Analysis

The application of the B&W clad collapse model will be the same for this application as presented for Oconee 1. The only change will be in the as-built

data. Therefore, clad collapse is not predicted prior to 21,000 effective full power hours of operation.

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