50-286

MAY 16 1973

R. C. DeYoung, Assistant Director for Pressurized Water Reactors, L

NUCLEAR DESIGN SECTION FOR INDIAN POINT 3 SER

Plant Name:	Indian Point 3
Docket No.:	50-286
Licensing Stage:	OL
Responsible Branch and Project Manager:	PWR-1 H. Specter
Requested Completion Date:	May 1, 1973
Technical Review Branch Involved:	Core Performance Branch
Description of Request:	OL Review
Review Status:	90% Complete

Attached is the Core Performance Branch input on nuclear safety, Section 4.3, for the Indian Point 3 Safety Evaluation Report.

V. Stello, Assistant Director
for Reactor Safety
Directorate of Licensing

cc: S. Hanauer
J. Hendrie
D. Vassallo
H. Specter
D. Ross
M. Dunenfeld

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DATE ▶	5/15/73	5/15/73	5/16/73			

NINE MILE POINT NUCLEAR STATION - UNIT 2

Docket No. 50-410

SAFETY EVALUATION

The fuel assembly and reactor core design for the initial loading of the Nine Mile Point 2 reactor will be very similar to those to be used in Browns Ferry 1, 2 and 3 (Dockets 50-259, 50-260 and 50-296), Peach Bottom 2 and 3 (Dockets 50-277 and 50-278), Duane Arnold (Docket 50-331), and Brunswick 1 and 2 (Dockets 50-324 and 50-325).

The fuel assemblies will contain pins with ^{235}U contents ranging from 0.71 to about 2.45 percent. In addition, selected fuel pins will contain low concentrations of gadolinia (full and partial length) for burnable poison reactivity control and for shaping the axial power distribution.

The final ^{235}U and gadolinia concentrations in the fuel pins and the ^{10}B concentration in the control rods are expected to have values such that the power distributions, peaking factors and other nuclear parameters will be quite similar to those of those previously reviewed BWR reactors. Results on the effects of changes in the significant reactivity coefficients on the consequences of the more important transients (Amendment 35 to the Browns Ferry SAR) are applicable to the Nine Mile Point 2 design. These studies showed little change in transient behavior when the reactivity coefficients were varied over a wider range than is expected to occur because of the small expected differences in the Nine Mile Point 2 core and other previously reviewed cores.

We asked the applicant to provide information (a) to demonstrate that monitoring of the power distribution will be adequate to ensure that operating limits for linear power density and critical heat flux ratio (CHFR) are not exceeded, and (b) to provide information on the errors associated with the determination of linear power density and CHFR. After reviewing the response by the applicant and subsequent discussions with General Electric and applicant personnel, we have concluded that satisfactory resolution can be achieved through a generic review of GE methods. An additional topical report discussing the power distribution aspects of BWRs (including comparisons of measured with calculated power distributions) is to be issued by GE in the fall of 1973. We conclude that sufficient information exists for purposes of a construction permit. We expect to continue our review of the nuclear design; should changes be required, they can be effected.

This review has not considered effects of fuel densification.