NRC FORM 195 (2-76)

U.S. NUCLEAR REGULATORY COMMISSION

DOCKET NUMBER 50-269/270/287

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TO: Mr. Edson G. Case

FROM: Duke Power Company
Charlotte, North Carolina
William O. Parker, Jr.

DATE OF DOCUMENT 10/24/77

DATE RECEIVED 10/27/77

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DESCRIPTION
Re our 6-8-76 ltr
Consists of info. concerning the study relative to the Oconee Station reactor vessel support system.....

(2-P)

ENCLOSURE

PLANT NAME: Oconee Units 1-2-3
RJL 10/28/77

DISTRIBUTION FOR REACTOR VESSEL SUPPORT INFO FOR OPERATING REACTORS PER MR. TRAMMELL 7-12-76

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WILLIAM O. PARKER, JR.
VICE PRESIDENT
STEAM PRODUCTION

October 24, 1977

TELEPHONE: AREA 704
373-4083

RECEIVED
OCT 27, 1977
U.S. NUCLEAR REGULATORY COMMISSION
Mail Section

Mr. Edson G. Case, Acting Director
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Mr. A. Schwencer, Chief
Operating Reactors Branch #1

Reference: Oconee Nuclear Station
Docket Nos. 50-269, -270, -287

Dear Sir:

Your letter of June 8, 1976 requested that a schedule be submitted concerning our response to questions relative to the Oconee Nuclear Station reactor vessel support system. In our letter dated September 14, 1976, we stated that it was our belief that the probability of occurrence of a pipe rupture within the reactor vessel cavity was acceptably remote and no further study was required. However, in order to evaluate and quantify this probability, we commissioned a study of the probability of a failure of the reactor coolant system in the annular region between the reactor vessel and the cavity shield wall.

By letter dated September 27, 1977 from Mr. D. O. Harris, Science Applications Incorporated, report SAI-050-77-PA, "An Analysis of the Probability of Pipe Rupture at Various Locations in the Primary Cooling Loop of a Babcock and Wilcox 177 Fuel Assembly Pressurized Water Reactor - Including the Effects of a Periodic Inspection", was submitted. This report evaluates the probability of a pipe rupture in the primary cooling loop of a B&W designed NSSS and evaluates the relative probability of a rupture within the reactor cavity. The effects of required in-service inspections are included.

The results of this study show that the probability of a large pipe rupture in the primary coolant loop is indeed very low. Further, if the postulated large rupture were to occur within the primary loop when a representative code required inservice inspection program had been utilized, only one in every 20-20,000 such ruptures would occur within the boundries of the reactor cavity.

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Therefore, this study confirms our position that the probability of occurrence of the postulated event is sufficiently low that no further study is warranted.

Very truly yours,

William O. Parker, Jr.

MST:ge