NRC FORM 195 (2-76)

U.S. NUCLEAR REGULATORY COMMISSION

NRC DISTRIBUTION FOR PART 50 DOCKET MATERIAL

DOCKET NUMBER: 50-269/270/287

FILE NUMBER

TO: Mr. Benard C. Rusche

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

DATE OF DOCUMENT: 1/4/77

DATE RECEIVED: 1/11/77

☒ LETTER ☐ NOTORIZED
☒ ORIGINAL ☒ UNCLASSIFIED
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PROP

INPUT FORM

NUMBER OF COPIES RECEIVED: One signed

DESCRIPTION

Ltr. w/attached....re our 11/29/76 ltr... concerning fission gas release for Oconee Units 1-2-3.

(4-P)

PLANT NAME:

Oconee Units 1-2-3

ENCLOSURE

ACKNOWLEDGED

DO NOT REMOVE

SAFETY — FOR ACTION/INFORMATION — ENVIRO 1/11/77 RJL

	SAFETY			ENVIRO
X	ASSIGNED AD:	Goller		ASSIGNED AD:
X	BRANCH CHIEF:	Schwencer		BRANCH CHIEF:
X	PROJECT MANAGER:	Zech		PROJECT MANAGER:
X	LIC. ASST. :	Sheppard		LIC. ASST. :

INTERNAL DISTRIBUTION

X	REG FILE		SYSTEMS SAFETY		PLANT SYSTEMS		SITE SAFETY &
X	NRC PDR		HEINEMAN		TEDESCO		ENVIRO ANALYSIS
X	I & E (2)		SCHROEDER		BENAROYA		DENTON & MULLER
X	OELD				LAINAS		
X	GOSSICK & STAFF		ENGINEERING		IPPOLITO		ENVIRO TECH.
	MIPC		MACARRY		KIRKWOOD		ERNST
	CASE		KNIGHT				BALLARD
	HANAUER		SIHWEIL		OPERATING REACTORS		SPANGLER
	HARLESS		PAWLICKI		STELLO		
							SITE TECH.
	PROJECT MANAGEMENT		REACTOR SAFETY		OPERATING TECH.		GAMMILL
	BOYD	X	ROSS	X	EISENHUT (LTR.)		STEPP
	P. COLLINS		NOVAK	X	SHAO		HULMAN
	HOUSTON		ROSZTOCZY	X	BAER (2)		
	PETERSON	X	CHECK	X	BUTLER		SITE ANALYSIS
	MELTZ	X	R. Meyer	X	GRIMES		VOLLMER
	HELTEMES		AT & I	X	F. Coffman		BUNCH
	SKOVHOLT		SALTZMAN	X	J. Guibert	X	J. COLLINS
			RUTBERG				KREGER

EXTERNAL DISTRIBUTION

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X	TIC:		REG V.IE	ULRIKSON (ORNL)
X	NSIC:		LA PDR	
	ASLB:		CONSULTANTS:	
X	ACRS 16 CYS ~~HOLDING~~/SENT: Cat. B. (1/11/77)			

CONTROL NUMBER: MM4 324

NRC FORM 195 (2-76)

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WILLIAM O. PARKER, JR.
VICE PRESIDENT
STEAM PRODUCTION

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373-4083

DOCKETED USNRC
NUCLEAR REGULATORY COMMISSION Mail Section

RECEIVED JAN 10 1977 U.S. NUCLEAR REGULATORY COMMISSION Mail Section

January 4, 1977

Mr. Benard C. Rusche, Director
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

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

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

Regulatory Docket File

Dear Mr. Rusche:

With reference to your November 29, 1976 letter concerning fission gas release (FGR), it has been determined that the three (3) Oconee Nuclear Station power reactors have already reached a burnup exposure in excess of 20,000 megawatt-days per metric ton of uranium in some fuel rods.

Please note that since the new fission gas release correlation has been submitted strictly for evaluation purposes at this time, the results should be reviewed accordingly. In particular, we have strong reservations concerning the applicability of this model to Oconee's low enriched UO2 fuel rod design.

As requested by your letter, we have evaluated the impact of using the new fission gas release correlation in our thermal performance code (TAFY). Calculations of pin pressure and temperature have been made. These calculations were based on utilizing the TAFY code with and without the NRC FGR equation.

The TAFY analysis without the NRC FGR equation was taken from the Oconee II Cycle 1 licensing analysis. The input parameters and TAFY NRC restrictions are listed in the attached Table 1. This input is representative of all three Oconee units. Results of pin pressure and fuel temperature calculations are shown in Table 2. Since the input to the analysis using the two models are identical, the differences in pressures between the two models can be directly attributed to the difference in FGR models. The fuel temperatures remain the same because of the NRC restriction that BOL temperatures be used for accident analysis.

324

The results indicated that the internal fuel rod pressure remains below the nominal system pressure for fuel burnup up to and including the maximum expected burnup.

The TAFY code without the NRC FGR equation is the code used in the safety analyses of LOCA and other accidents. A survey of the safety analyses of all accidents was performed, and it was concluded that the NRC fission gas release model would not affect the results of the safety analyses of non-LOCA transients. The average fuel temperatures used in the transient cladding temperature calculations for these transients are not changed by the new NRC FGR model. The higher pin pressures at end-of-life would not result in cladding rupture during these transients.

In case of the LOCA analysis, if the NRC FGR Model is used in TAFY, then the LOCA analysis will be impacted in an unfavorable manner. This is the result of higher pin pressures occurring at earlier burnups. Since initial inside and outside cladding surface oxide layers would be thinner at earlier burnups, the zircaloy-water (metal-water) reaction could possibly be larger than that previously calculated. The increased energy generation in the cladding could possibly result in a peak cladding temperature increase in excess of $20^{o}F$. The evaluation of the exact impact of this presumed higher fission gas release upon LOCA results would require an extensive analysis.

Duke Power Company believes that the staff's suggested fission gas release correlation is not applicable for the type of fuel and extent of burnup utilized in Oconee reactors and that further analysis is not warranted. B&W has made an independent review of the literature and the available experimental data to determine the appropriateness of the staff's new model. Although it appears that there is some enhancement of fission gas release at very high burnups, B&W's evaluation of the available $UO2$ data indicates that the increase in release rates with high burnups occurs later in life and to a lesser extent than one would predict using the staff's suggested FGR correlation. B&W's current approved analytical fuel pin model (TAFY), used in calculating fuel temperatures and pin pressures, contains significant margins to ensure sufficient conservatism. The additional margin provided for in the NRC staff's FGR model becomes overly conservative when compared with current evaluation techniques.

In accordance with your specific request, three signed originals and 40 copies of this letter are submitted.

Very truly yours,

William O. Parker, Jr.

William O. Parker, Jr. by

TABLE 1

PIN PRESSURE ANALYSIS INPUT

(OCONEE 11 NSSS-4)

FUEL

INITIAL MEAN DENSITY - % TD	92.5
INITIAL MEAN DIAMETER - IN	0.370
INITIAL LTL DENSITY - % TD	92.0
FINAL DENSITY - % TD	96.5
DISH RADIUS - IN	0.150
DISH FACTOR	0.0170
INITIAL STACK LENGTH - IN	144

CLAD

CLAD ID - IN	0.377
CLAD OD - IN	0.430
CLAD LENGTH - IN	153
INITIAL PLENUM VOLUME - IN^3	0.75

RESTRICTIONS

25% REDUCTION ON H_{GAP}

NO RESTRUCTURING

SORBED GAS CONTENT - 0.01 CC/GM

USE BOL TEMPERATURES FOR ACCIDENT ANALYSIS

TABLE 2

PIN PRESSURE AND FUEL TEMPERATURE FOR OCONEE 1, 2, & 3

Peak Rod Burnup (MWD/MTU)	TAFY B&W FGR Model Pin Press. (psi)	TAFY B&W FGR Model Avg. Fuel Temp.* at 17 KW/FT (°F)	TAFY NRC FGR Model Pin Press. (psi)	TAFY NRC FGR Model Avg. Fuel Temp.* at 17 KW/FT (°F)
20,000	1210	2990	1210	2990
22,000	1235	2990	1240	2990
25,000	1295	2990	1320	2990
27,000	1340	2990	1410	2990
30,000	1400	2990	1550	2990
32,000	1430	2990	1615	2990
35,000	1470	2990	1745	2990
37,000	1510	2990	1865	2990
38,000	1525	2990	1925	2990

*BOL Values