

NRC DISTRIBUTION FOR PART 50 DOCKET MATERIAL

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
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INPUT FORM

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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 |
| <input checked="" type="checkbox"/> ASSIGNED AD: | Goller | | | ASSIGNED AD: | | |
| <input checked="" type="checkbox"/> BRANCH CHIEF: | Schwencer | | | BRANCH CHIEF: | | |
| <input checked="" type="checkbox"/> PROJECT MANAGER: | Zech | | | PROJECT MANAGER: | | |
| <input checked="" type="checkbox"/> LIC. ASST.: | Sheppard | | | LIC. ASST.: | | |

| INTERNAL DISTRIBUTION | | | |
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| <input checked="" type="checkbox"/> REG FILE | SYSTEMS SAFETY | PLANT SYSTEMS | SITE SAFETY & |
| <input checked="" type="checkbox"/> NRC PDR | HEINEMAN | TEDESCO | ENVIRO ANALYSIS |
| <input checked="" type="checkbox"/> I & E (2) | SCHROEDER | BENAROYA | DENTON & MULLER |
| <input checked="" type="checkbox"/> OELD | | LAINAS | |
| <input checked="" type="checkbox"/> 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 | ROSS | EISENHUT (171) | STEPP |
| P. COLLINS | NOVAK | SHAO | HULMAN |
| HOUSTON | ROSZTOCZY | BAER (2) | |
| PETERSON | CHECK | BUTLER | SITE ANALYSIS |
| MELTZ | R. Meyer | GRIMES | VOLLMER |
| HELTEMES | AT & I | F. COFFMAN | BUNCH |
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| EXTERNAL DISTRIBUTION | | |
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| <input checked="" type="checkbox"/> NSIC: | LA PDR | |
| ASLB: | CONSULTANTS: | |
| <input checked="" type="checkbox"/> ACRS 16 CYS HOLDING/SENT: | Cat. B. (1/11/77) | |

CONTROL NUMBER

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DUKE POWER COMPANY

POWER BUILDING

422 SOUTH CHURCH STREET, CHARLOTTE, N. C. 28242

WILLIAM O. PARKER, JR.
VICE PRESIDENT
STEAM PRODUCTION

TELEPHONE: AREA 704
373-4083

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

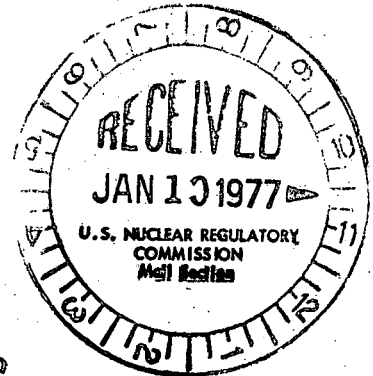
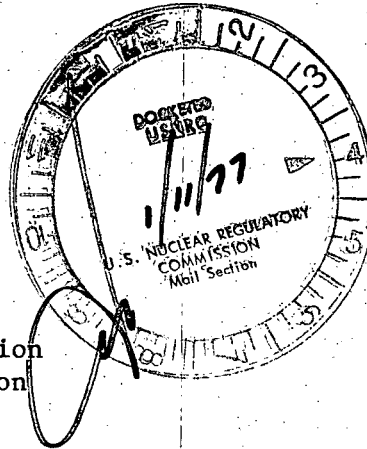
Dear Mr. Rusche:

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.



Regulatory Docket File

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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°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 JWS*

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 |

RESTRICTIONS25% 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

| <u>Peak Rod Burnup (MWD/MTU)</u> | <u>TAFY B&W FGR Model</u> | | <u>TAFY NRC FGR Model</u> | |
|--------------------------------------|-------------------------------|--|-----------------------------|--|
| | <u>Pin Press. (psi)</u> | <u>Avg. Fuel Temp.* at 17 KW/FT (°F)</u> | <u>Pin Press. (psi)</u> | <u>Avg. Fuel Temp.* at 17 KW/FT (°F)</u> |
| 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 | 1450 | 2990 | 1615 | 2990 |
| 35,000 | 1470 | 2990 | 1745 | 2990 |
| 37,000 | 1510 | 2990 | 1865 | 2990 |
| 38,000 | 1525 | 2990 | 1925 | 2990 |

*BOL Values