NRC FORM 195 (2-76)

U.S. NUCLEAR REGULATORY COMMISSION

DOCKET NUMBER 50 - 302

FILE NUMBER

NRC DISTRIBUTION FOR PART 50 DOCKET MATERIAL

TO: Mr. John F. Stolz

FROM: Florida Power Corp. ST. Petersburg, Fla. S. A. Brandimore

DATE OF DOCUMENT 5/20/77

DATE RECEIVED 5/23/77

X LETTER X NOTORIZED PROP INPUT FORM

X ORIGINAL X UNCLASSIFIED

☐ COPY

NUMBER OF COPIES RECEIVED 3 SIGNED

DESCRIPTION

(1-P)

ENCLOSURE

Consists of requested additional info. concerning the effects of increased fission gas releases on the safety analysis......... notorized 5/20/77......

(5-P)

DO NOT REMOVE

40 encl.

PLANT NAME:
Crystal River Unit No. 3

RJL

FOR ACTION/INFORMATION

SAFETY			ENVIRO
	ASSIGNED AD:		ASSIGNED AD:
✓	BRANCH CHIEF: (5) STOLZ		BRANCH CHIEF:
✓	PROJECT MANAGER: ENGLE		PROJECT MANAGER:
✓	LIC. ASST. : HYLTON		LIC. ASST. :

INTERNAL DISTRIBUTION

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

EXTERNAL DISTRIBUTION

✓	LPDR: CRYSTAL RIVER FLA	NAT. LAB:	BROOKHAVEN NAT. LAB.	CONTROL NUMBER
✓	TIC:	REG V.IE	ULRIKSON (ORNL)	~~771440104~~
✓	NSIC:	LA PDR		
	ASLB:	CONSULTANTS:		
✓	ACRS 16 CYS ~~HOLDING~~/SENT AS CAT B			

NRC FORM 195 (2-76)

8003160265 P

Regulatory File Cy.

Florida Power
CORPORATION

May 20, 1977

Mr. John Stolz
Branch Chief
Light Water Reactors Branch #1
Division of Project Management
U.S. Nuclear Regulatory Commission
Washington, DC 20555

SUBJECT: Florida Power Corporation
Crystal River Unit 3
Docket No. 50-302

Dear Mr. Stolz:

In your letter of February 11, 1977, you requested Florida Power Corporation to evaluate the effects of increased fission gas releases on the safety analysis for Crystal River Unit 3.

Since Crystal River Unit 3 will not reach a local exposure (burnup) of 20,000 megawatt-days per metric ton of uranium prior to June 1, 1977, we were to furnish the requested information to the Commission within 90 days of receipt of your letter.

Attached for your staff's review are 3 signed originals and 40 copies of our response to the information requested in Items (a) through (d) of your February 11, 1977, letter.

If you have any questions regarding this matter, please contact us.

Very truly yours,

S.A. Brandimore
Senior Vice President
and General Counsel

Attachment

SAB:ECS:hlc 5/4

771440104

General Office 3201 Thirty-fourth Street South • P.O. Box 14042, St. Petersburg, Florida 33733 • 813—866-5151

IN WITNESS WHEREOF, the applicant has caused its name to be hereunto signed by S.A. Brandimore, Senior Vice President and General Counsel, and its corporate seal to be hereunto affixed by Betty M. Clayton, Assistant Secretary, thereunto duly authorized the 20th day of May, 1977.

FLORIDA POWER CORPORATION

By S.A. Brandimore

S.A. Brandimore
Senior Vice President
and General Counsel

ATTEST

Betty M. Clayton
Assistant Secretary

(CORPORATE SEAL)

Sworn to and subscribed before me this 20th day of May, 1977.

Notary Public

My Commission Expires:

Notary Public State of Florida at Large
My Commission Expires July 9, 1978

(NOTARIAL SEAL)

585-P

RESPONSE TO NRC QUESTIONS CONCERNING NEW NRC FISSION GAS RELEASE MODEL

Response to Item a)

It is estimated at the present time that a maximum local exposure (burnup) of 20,000 megawatt-days per metric ton of uranium (MWD/tU) will be reached by any fuel rod in Crystal River Unit 3 during the second fuel cycle. The second fuel cycle for Crystal River Unit 3 will not begin until late fall of 1978.

Response to Item b)

The results of the evaluation of the revised fission gas release model for the TAFY fuel pin analysis are provided in the attached two tables. Table 1 defines the input parameters used for the evaluation and Table 2 provides the comparison of internal pin pressure for the two fission gas release models as a function of burnup. At a maximum burnup of 38,000 MWD/MTU, the informal pin pressure, based on the NRC staff model reflects a 26% increase when compared with the results of the current model; however, in all instances, the internal pin pressure remains below system pressure. The TAFY calculations have demonstrated that the average fuel temperature at BOL conditions and 17 kw/ft is equal to 2990°F for both the NRC and B&W models.

These calculations of pin pressure and temperature, based on utilizing the TAFY Code with and without the NRC fission gas release (FGR) equation, were performed for Three Mile Island, Arkansas and the Oconee reactors as a class of power reactors. The Crystal River Unit 3 reactor belongs to the above-mentioned class of reactors and therefore the results of these calculations provided in Table 2 are equally applicable to Crystal River Unit 3.

The TAFY Analysis without the NRC FGR equation was taken from the Oconee II Cycle 1 licensing analysis. Since the input data for these analyses are identical for both the NRC and B&W FGR models, the differences reported in Table 2 can be directly attributed to the difference in the FGR models.

Response to Item c)

The TAFY Code without the NRC FGR equation is the code used in the current LOCA and safety analyses for CR #3. Use of the NRC FGR equation in TAFY will impact the LOCA analysis as the worst pin pressure would now occur at earlier burnups. Since initial inside and outside cladding surface oxide layers would be thinner at earlier burnups, the zircaloy-water (metal-water) reaction would be larger than that previously calculated. The increased energy generation in the cladding would raise the peak cladding temperature and would probably result in the present LOCA limits violating the criteria of 10 CFR 50.46. Requalification of the LOCA limits at CR #3 would then be necessary, ultimately resulting in the issuance of revised Technical Specifications.

A survey of the non-LOCA-related safety analysis accidents was performed and it was concluded that the NRC FGR model would not affect the results. The average fuel temperatures used in the transient cladding temperature calculations are not changed by the new NRC FGR model. Further, the higher pin pressures at EOL would not result in cladding rupture during these transients. Hence, the non-LOCA safety analysis accidents for CR #3 are not impacted by the NRC FGR model.

Response to Item d)

As stated above, the internal fuel rod pressure does not exceed the nominal system pressure and therefore, no discussion is given for operations with fuel cladding in tension.

ECS:hlc 5/6

TABLE 1

PIN PRESSURE ANALYSIS INPUT

(OCONEE II NSS-4)

FUEL

INITIAL MEAN DENSITY - % TD	92.5
INITIAL MEAN DIAMETER - IN	0.370
INITIAL LTL DENSITY - % TD	92.0
FINAL DESNITY - % 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
INITIAL BACKFULL PRESSURE - PSIA	375.0

RESTRICTIONS

25% REDUCTION ON H_{GAP}

NO RESTRUCTURING

SORBED GAS CONTENT - 0.01 CC/GM

USE BOL TEMPERATURES FOR ACCIDENT ANALYSIS

TABLE 2

OCONEE 1, 2, & 3; TMI-1: ANO-1 DATA

Peak Rod Burnup (MWD/MTU)	TAFY B&W FGR Model Pin Pressure (psi)	TAFY NRC FGR Model Pin Pressure (psi)
20,000	1210	1210
22,000	1235	1240
25,000	1295	1320
27,000	1[illegible]40	1410
30,000	1400	1550
32,000	1430	1615
35,000	1470	1745
37,000	1510	1865
38,000	1525	1925