

50-302

NRC DISTRIBUTION FOR PART 50 DOCKET MATERIAL

FILE NUMBER

TO: Mr. John F. Stolz		FROM: Florida Power Corp. ST. Petersburg, Fla. S. A. Brandimore		DATE OF DOCUMENT 5/20/77	
<input checked="" type="checkbox"/> LETTER <input checked="" type="checkbox"/> ORIGINAL <input type="checkbox"/> COPY		<input checked="" type="checkbox"/> NOTORIZED <input checked="" type="checkbox"/> UNCLASSIFIED	PROP	INPUT FORM	DATE RECEIVED 5/23/77
NUMBER OF COPIES RECEIVED 3 SIGNED					

DESCRIPTION

PLANT NAME:
Crystal River Unit No. 3

RJL

(1-P)

ENCLOSURE

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

DO NOT

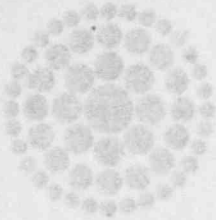
40 encl.

(5-P)

SAFETY		FOR ACTION/INFORMATION		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
& E (2)	SCHROEDER	BENAROYA	DENTON & MULLER
OELD		LAINAS	
GOSSICK & STAFF	ENGINEERING	IPPOLITO	ENVIRO TECH.
MIPC	MACARRY	KIRKWOOD	ERNST
CASE	BOSNA		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
	RUTEERG		KREGER

EXTERNAL DISTRIBUTION			CONTROL NUMBER
LPDR: CRYSTAL RIVER	NAT. LAB:	BROOKHAVEN NAT. LAB.	771440104 MA 4 (1)
TIC:	REC V. IE	ULRIKSON (DRNL)	
NSIC:	LA PDR		
ASLB:	CONSULTANTS:		
ACRS 16 CYS HOLDING SENT AS CAP B			



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 N o. 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

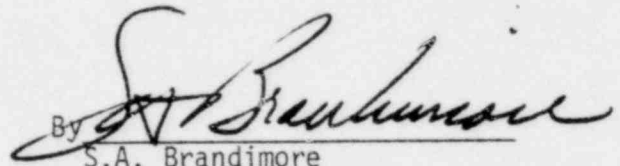
Attachment

SAB:ECS:hlc 5/4

271440104

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
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 internal 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 11 NSS-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
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

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