James A. FitzPatrick
Nuclear Power Plant
P.O. Box 41
Lycoming, New York 13093

315-342-3840

NewYorkPower
Authority

Michael J. Colomb
Site Executive Officer

October 6, 2000
JAFP-00-0232

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Mail Station P1-137
Washington, D.C. 20555

SUBJECT: James A. FitzPatrick Nuclear Power Plant
Docket No. 50-333
Information Regarding Power Uprate SER for Amendment 239

References:
1. NYPA Letter, H. P. Salmon, Jr. to the NRC, "Discrepancies in NRC Safety Evaluation Report Associated with Technical Specification (TS) Amendment 239," (JPN-99-041), dated November 30, 1999

2. NRC Letter, K. Cotton to W. J. Cahill, Jr., Regarding Issuance of Amendment 239 for James A. FitzPatrick Nuclear Power Plant (TAC No. M92781), dated December 6, 1996

Dear Sir:

This letter provides information requested by the NRC staff during a telephone conference held on September 28, 2000. The staff requested this information as a result of their review of discrepancies identified by the Authority (Reference 1) in the Safety Evaluation Report associated with FitzPatrick Technical Specification Amendment 239 (Reference 2). Specifically, the staff requested information regarding which edition and addenda of the ASME code was used to evaluate FitzPatrick's CRDMs (Control Rod Drive Mechanisms) in support of the FitzPatrick power uprate license amendment application.

In Reference 1, the Authority stated that pressure containing portions of the CRDMs are designed and fabricated in accordance with the requirements of Section III of the 1965 ASME Boiler and Pressure Vessel Code with Winter 1966 Addenda. Reference 1 also stated that several CRDMs have been modified and comply with the stress requirements of ASME III 1971 edition, with Winter 1972 Addenda, and including the 1974 Edition with Winter 1975 Addenda. (Under ASME Section XI, utilizing the original or later version of the ASME Section III code is allowed.)

Technical Specification Amendment 239 (Reference 2) stated that the Authority has evaluated the adequacy of the Control Rod Drive Mechanism (CRDM) in accordance with the code of record, (ASME Code Section III, 1965 Edition and Addenda through Winter 1966). This Amendment did not, however, mention that some of the drives had been modified and had been qualified to a later edition of the ASME code.

A047

The NRC requested during a September 28, 2000 telephone call, that the Authority submit the nuclear safety evaluation (10 CFR 50.59) and General Electric Certified Stress reports associated with this modification for confirmation of the code used. Attachments 1 and 2 provide the two documents requested.

There are no new commitments contained in this letter. If you have any questions, please contact Mr. George Tasick at (315) 349-6572.

Very truly yours,

Michael J. Colomb
Site Executive Officer

STATE OF NEW YORK
COUNTY OF OSWEGO
Subscribed and sworn to before me this 6 day of Oct 2000.

Nancy B. Czerow
Notary Public

NANCY B. CZEROW
Notary Public, State of New York
Qualified In Oswego County #4884611
Commission Expires 1-26-01

cc: Regional Administrator
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

Office of the Resident Inspector
U.S. Nuclear Regulatory Commission
P.O. Box 136
Lycoming, New York 13093

Mr. Guy Vissing, Project Manager
Project Directorate I
Division of Licensing Project Management
U.S. Nuclear Regulatory Commission
Mail Stop 8C2
Washington, DC 20555

Attachment I to JAFP-00-0232

Nuclear Safety Evaluation for Control Rod Drive Mechanism Upgrade (BWR/4 to BWR/6)

JAF-SE-87-158

New York Power Authority

JAMES A. FITZPATRICK NUCLEAR POWER PLANT
Docket No. 50-333
DPR-59

NEW YORK POWER AUTHORITY
JAMES A. FITZPATRICK NUCLEAR POWER PLANT

NUCLEAR SAFETY EVALUATION
NO. JAF-SE-87-158

TITLE: CONTROL ROD DRIVE MECHANISM UPGRADE (BWR/4 TO BWR/6) QA CLASS: I

___ Plant Modification F1- -
_X_ Minor Modification M1-86-102
___ TEST NO.
___ EXPERIMENT
___ OTHER (Describe) ________

A. The proposed change, test or experiment:

1. ( ) Does
(X) Does Not - Increase the probability of occurrence or consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR.

2. ( ) Does
(X) Does Not - Create the possibility of an accident or malfunction of a type other than any evaluated previously in the FSAR.

3. ( ) Does
(X) Does Not - Reduce the margin of safety as defined in the basis for Technical Specifications.

4. ( ) Does
(X) Does Not - Involve a change in the Technical Specifications (nuclear or environmental). Para/Sec. N/A

5. ( ) Does
(X) Does Not - Involve an unreviewed safety question (1, 2, 3 and/or 4).

6. ( ) Does
(X) Does Not - Contain Security Safeguards Information.

John Erkan
Prepared by: JOHN ERKAN
Title: SR. PLANT ENG. SUPV.
Date: 12/22/87
Reviewed by: Charles J Brown
(normally Tech. Serv.)
Title: Plant Engineer
Date: 12/23/87

(MINOR MOD AUTH. ONLY)
Dept. Supt: [signature] 12/22/87 Date
Tech. Svc. Supt. [signature] 12/22/87 Date
Supt. of Power [signature] 12/21/ Date

PORC MTG. NO. & DATE 87-112 12/2?/?

SRC MTG. NO. & DATE 03-88 3/5/88

NEW YORK POWER AUTHORITY
JAMES A. FITZPATRICK NUCLEAR POWER PLANT
NUCLEAR SAFETY EVALUATION

JAF-SE-87-158

Attachment to Minor Modification M1-86-102

A. DESCRIPTION OF MOD/CHANGE:

The existing BWR/4 control rod drive mechanisms (CRD) shall be replaced by, or modified to, the General Electric BWR/6 type design. The BWR/6 CRD incorporates an improved cooling water orifice to reduce plugging and utilizes the redesigned piston tube and hydraulic buffer. Other modifications include dimensional changes and material upgrades for the index tube, spud and cylinder, inner filter, and drive piston assembly. Finally, the uncoupling rod was redesigned as a one piece design to prevent the inadvertent installation of the rod in the wrong spud hole leading to coupling difficulties.

B. SAFETY/ENGINEERING EVALUATION AND BASIS:

The updated BWR/6 CRD's were designed for a faster scram performance primarily accomplished by a larger scram capability (i.e. higher pressure and larger nitrogen volume). They were analyzed and tested for the increased pressure loads and dynamic forces to support the BWR/6 fast scram requirements. These units have accumulated more than 2000 reactor years of successful operating experience and provides improved operating reliability and reduced maintenance. As applied to the BWR/4 plants, the BWR/6 device will be subjected to significantly lower operating loads than its inherent design capability, resulting in a significant increase in the drive component safety design margin. Therefore, the probability of occurrence or magnitude of the consequences of an accident or malfunction of equipment important to safety previously analyzed in the FSAR is not increased.

The possibility of an accident or malfunction of a different type than any evaluated previously in the FSAR is not created because the improved CRD assemblies have been evaluated and tested by the original NSSS supplier (G.E.) and have demonstrated reliable performance throughout the expected operating conditions. The BWR/6 drives are QA Class I, nuclear safety-related, and were designed to be fully compatible with the existing equipment interfaces, resulting in no changes to the JAF supporting hydraulic system requirements.

In the event that BWR/6 conversion parts were inadvertently interchanged with existing BWR/4 parts, the scram capability would not be impaired, however, several drive functional anomalies would result. Plant procedures shall be revised or written, as appropriate, to prevent this installation error.

NEW YORK POWER AUTHORITY
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NUCLEAR SAFETY EVALUATION

JAF-SE-87-158

Attachment to Minor Modification M1-86-102

The margin of safety as defined in the basis of any Technical Specification is not reduced. No safety or licensing concerns are presented by the replacements, nor are any changes to the JAF Technical Specifications required. Section 3.5.5.1 of the FSAR shall be updated to reflect the new CRD installation.

Attachment (1) provides the completed General Electric modification safety evaluation describing all modifications in detail, performance/acceptance testing, and interfaces with CRD System operation. Additionally, Reference a provides verification that the BWR/6 and BWR/4 drives operate with negligible differences.

Based on the discussions presented above and in the attached safety evaluation, it is determined that the CRD conversion does not involve an unreviewed safety issue.

E. REFERENCES

a) GENSNYPA-87-0813-2, Letter from J. Silva to R. Weise, dated August 13, 1987

b) FSAR Section 3.5.5.1, Control Rod Drive Mechanisms

c) FSAR Section 12.2.3, Classification of Structures and Equipment (Class I Equipment)

d) JAF Technical Specifications

e) GEK-784A, Control Rod Drive System

Attachment 2 to JAFP-00-0232

General Electric Control Rod Drive Certified Stress Report

New York Power Authority

JAMES A. FITZPATRICK NUCLEAR POWER PLANT
Docket No. 50-333
DPR-59

REVISION STATUS SHEET

23A4920	REV. 2
CONT ON SHEET 2	SH NO. 1

GENERAL ELECTRIC

NUCLEAR ENERGY BUSINESS OPERATIONS

DOCUMENT TITLE CONTROL ROD DRIVE

TYPE CERTIFIED STRESS REPORT

LEGEND OR DESCRIPTION OF GROUPS

FMF N/A

MPL ITEM NO. N/A

| - DENOTES CHANGE

REVISIONS			C
0	DMH-2067 5/2/86		
1	D. DENHAM 1/21/87 NH26696 CHKD BY: L. AMARAL	GAB	
2	D. Denham D. DENHAM 27 APR 1987 NH26824 CHKD BY: G. Baylis G. BAYLIS	G.B. GB	

PR				
765				
771HIR				
SD-2				

PRINTS TO

MADE BY	APPROVALS	DEPT	LOCATION
SA KENRECK 1/30/86	JC CARRUTH 5/2/86	NEBO	SAN JOSE
CHKD BY	ISSUED		
L AMARAL 1/30/86	GA BAYLIS 5/2/86	CONT ON SHEET 2	SH NO. 1

NEO 805 (REV 1/82)

S-190-VEND-DW

F1

The purpose of this document is to provide the ASME Boiler and Pressure Vessel Code, Section III, Division 1, Nuclear Power Plant component required stress report for the components covered in this report. This document reconciles any differences between the Design Specification Code requirements for BWR/2-5 (21A8781 Rev. 2 and 22A6253 Rev. 0) and BWR/6 (22A5556 Rev. 2) so that the components covered by this report can be used as replacements for those covered by the BWR/2-5 stress reports (22A2017 Rev. 3 and 22A6254 Rev. 0). Demonstration that functional requirements are met will be done elsewhere.

REFERENCE DOCUMENT

DOCUMENT NUMBER	REVISION NUMBER	TYPE OF DOCUMENT	TITLE
DC22A4912	0	DESIGN CERTIFICATION	CONTROL ROD DRIVE

I certify that, to the best of my knowledge and belief, this stress report reconciliation complies with all requirements of the Design Specification 21A8781 Rev. 2 and 22A6253 Rev. 0, Paragraph NA3350, and Subsections IWA* and IWB* of the ASME Boiler and Pressure Vessel Code, Section III and XI, Division 1, Nuclear Power Plant Components, 1971 Edition, Winter 1972 Addenda to and including the 1974 Edition, Winter 1975 Addenda and as such the components covered by it may be used as replacement appurtenances for items previously covered by stress reports 22A2017 Rev.3 and 22A6254 Rev. 0. (*Subsections IWA and IWB did not exist for this edition of the Code. Since rules governing replacement parts are better defined in later editions we will use 1983 rules to determine adequacy of the replacements but certify to the 1972-75 Code to match the existing stress reports and certifications) Code cases N207 and 1361 apply to BWR/6 components. BWR/2-5 components only rely on Code case 1361. Prior to application of Code case N207, the owner (utility) must be notified.

LISTED DOCUMENTS

Type of Document	Title	Document Number	Revision Number
Stress Report	Control Rod Drive	22A4912	2
Product Drawing	Cylinder, Tube and Flange	919D258G3	14-16
Product Drawing	Plug	159A1176P1	9
Product Drawing	Cylinder and Flange	919D254G1	15-22
Product Drawing	Flange	719E474G1	1-13
Product Drawing	Plug	175A7961P1	9
Product Drawing	Flange (Forging)	919D610P1	8-9
Product Drawing	Piston Tube	105D6495G1	2-4

NUCLEAR ENERGY BUSINESS OPERATIONS — GENERAL ELECTRIC

23A4920 REV 2	SH NO. 3

Type of Document	Title	Document Number	Revision Number
Product Drawing	Piston Tube, Lower	105D6476G1	2-3
Product Drawing	Indicator Tube	166B9275G1	1
Product Drawing	Pipe	166B9313P1	1
Product Drawing	Cap	166B9274P1	2
Product Drawing	Base	137C5311P1	2-3
Product Drawing	Nut	137C5934P1	3
Product Drawing	Ring Flange	114B5122P2 or P3	15-16
		OR 137C8151P1	0
Product Drawing	Cap Screw	117C4516P2	8-9

REGISTERED PROFESSIONAL ENGINEER JOHN C. CARRUTH M21193 MECHANICAL STATE OF CALIFORNIA

Certified By: J C Carruth
Registered Professional Engineer

P.E. Number: M21193

State: California

Date: April [illegible]

Documentation of Review of this Stress Report

This documents that this reconciliation has been reviewed in accordance with the requirements of Paragraph NA3260 of the ASME Boiler and Pressure Vessel Codes, Section III, and satisfies the requirements of the certified design specifications 21A8781 Rev 2 and 22A6253 Rev. 0.

Review By: General Electric Company
Nuclear Energy Systems Division
San Jose, California

Acting as Owner's Designee

H. C. Russell — Signature

APRIL 21, 1987 — Date