123 Main Street
White Plains, New York 10601
914 681.6200

New York Power Authority

J. Phillip Bayne
Executive Vice President
Nuclear Generation

June 29, 1984
JPN-84-42

Director of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Attention: Mr. Domenic B. Vassallo, Chief
Operating Reactors Branch No. 2
Division of Licensing

Subject: James A. FitzPatrick Nuclear Power Plant
Docket No. 50-333
Generic Letter No. 83-28
Required Actions Based on Generic
Implications of Salem ATWS Events

References: 1. NRC Generic Letter No. 83-28, D. G. Eisenhut to All Licensees, dated July 8, 1983.

2. NYPA letter, J. P. Bayne to D. G. Eisenhut, dated September 6, 1983 (JPN-83-80) regarding same subject.

3. NYPA letter, J. P. Bayne to D. B. Vassallo, dated November 9, 1983 (JPN-83-92) regarding same subject.

Dear Sir:

Reference 1 requested information, plans and schedules related to the generic implications of the February, 1983 Salem ATWS events. Reference 2 provided the Authority's preliminary schedule for submitting the requested information.

A055
1/1

In Reference 3, the Authority responded to the extent practical and committed to further supplement our response.

Attached is our final response to Generic Letter No. 83-28. It describes those programs and procedures currently in effect at FitzPatrick. It also describes our plans and schedules for items not addressed by current programs.

Some of the information included in the attachment duplicates prior submittals. This was done to consolidate all elements of our response in a single document.

If you have any questions or require additional information, please contact Mr. J. A. Gray, Jr. of my staff.

Very truly yours,

Robt A Burns for

J. P. Bayne
Executive Vice President
Nuclear Generation

State of New York
County of Westchester

Subscribed and Sworn to before me
this 29 day of June 1984.

Jeanne La Luna
Notary Public

JEANNE LA LUNA
NOTARY PUBLIC, STATE OF NEW YORK
NO. 60-4614305
QUALIFIED IN WESTCHESTER COUNTY
TERM EXPIRES MARCH 30th 19.85...

cc: Office of the Resident Inspector
U.S. Nuclear Regulatory Commission
Lycoming, New York 13093

NEW YORK POWER AUTHORITY
James A. FitzPatrick Nuclear Power Plant (JAFNPP)

RESPONSE TO GENERIC LETTER 83-28

Attachment 1 to JPN-84-42

This document describes the current procedures and programs for each item identified in NRC Generic Letter 83-28. A description of the Authority's position or plans is also provided for those items not currently addressed in procedures or programs. An estimated completion date is also provided, where sufficient information exists to project a date.

Response to Item 1.1 - Post Trip Review (Program Description and Procedure)

The JAFNPP post trip review program and procedure is contained in Operations Department Standing Order (ODSO) 23 titled "Post Trip Review." This ODSO addreses Items 1.1.1, 1.1.2, 1.1.4, 1.1.5 and 1.1.6 of Generic Letter 83-28. A copy of ODSO 23 is included as Appendix A.

1.1.3 The qualifications and training of the personnel responsible for completing ODSO 23 is described below:

Shift Technical Advisor (STA)
The STA (or the individual fulfilling the STA role) has been trained and is qualified to the requirements of NUREG 0737. This includes training in the analysis and determination of the causes of off-normal conditions such as mitigating core damage.

Operations Superintendent and Shift Supervisors
The Operations Superintendent and Shift Supervisors are qualified in accordance with the requirements of ANSI 18.1 (1971) with respect to training and experience and hold Senior Reactor Operator (SRO) licenses. In addition, training is currently being conducted to upgrade the Shift Supervisor's level of training to meet or exceed the "Draft Comission Policy for Engineering Expertise on Shift" (Federal Register Volume 46, No. 143, July 25, 1983).

Other Shift Personnel
Other shift personnel are, as a minimum, qualified in accordance with the requirements of ANSI 18.1 (1971) with respect to training and experience. In addition, a number of these personnel hold Reactor Operator (RO) or SRO licenses.

Response to Item 1.2 - Post-Trip Review, Data and Information Capability

1.2.1 Capability for Assessing Sequence of Events

1.2.1.1 The sequence of events function is provided by the plant process computer's Sequence of Events log. The Sequence of Events log is initiated (printed) by a change in state of any of 116 digital (on-off) inputs.

1.2.1.2 Currently 161 points are available of which 45 are designated as spares. The 116 points in use at this time are listed below:

TABLE 1

Process Computer Points Currently In-Use

POINT ID	DESCRIPTION
D000	RFP A SUCTION PRESSURE
D001	RFP B SUCTION PRESSURE
D002	RFP A DISCHARGE PRESS
D003	RFP B DISCHARGE PRESS
D013	115KV SOUTH BUSS UV RLY
D014	115KV NORTH BUSS UV RLY
D015	115KV BRKR 10012 OPEN
D016	115KV BRKR 10012 CLOSED
D017	115KV BRKR 10022 OPEN
D018	115KV BRKR 10022 CLOSED
D019	345KV BRKR 10042 OPEN
D020	345KV BRKR 10042 CLOSED
D021	BREAKER FAILURE 10012
D022	BREAKER FAILURE 10022
D023	BREAKER FAILURE 10042
D024	BREAKER FAILURE 10052
D025	345KV NMPT LINE #10
D026	345KV EDIC LINE #1
D027	345 BUSS
D028	115KV LGHT HSE HLL LINE
D029	115KV LINE TO NINE MILE
D030	345KV BRKR 10052 OPEN
D031	RESERVE STATION TFR T2 UV
D032	RESERVE STATION TFR T3 UV
D033	345KV BRKR 10052 CLOSED
D034	MOIST SEP A HI LVL TRP
D035	MOIST SEP B HI LVL TRP
D036	SONIC DETECTOR RV-2-71A
D037	SONIC DETECTOR RV-2-71B
D038	SONIC DETECTOR RV-2-71C
D039	SONIC DETECTOR RV-2-71D
D040	SONIC DETECTOR RV-2-71E
D042	SONIC DETECTOR RV-2-71F
D041	SONIC DETECTOR RV-2-71G
D042	SONIC DETECTOR RV-2-71H
D043	SONIC DETECTOR RV-2-71J
D044	SONIC DETECTOR RV-2-71K
D045	SONIC DETECTOR RV-2-71L
D052	TBN EHC PANEL 24 VDC PWR
D054	MAIN TURBINE TRIP
D055	TBNE BACK UP OVERSPD TRIP

D056	TBNE LOSS OF 125 VDC TRP
D057	TURBINE MANUAL TRIP
D058	TBNE EXH HOOD HI T TRIP
D059	LOW COND VACUUM A TRIP
D060	LOW TBNE BRG OIL PRESS
D062	TBNE TRIP HI VIBRATION
D063	TBNE TRIP-LOSS STAT CLNT
D064	TBNE TRP-THRST BRG WEAR
D065	TBNE TRP-SHAFT PMP PRES
D066	TBN TRP-EMERG TRP FLUID
D067	TBN TRP-HYD FLUID PRESS
D068	TBN TRP-LOSS SPEED FDBK
D500	SDIV A1 W LEVEL SW SCRAM
D501	SDIV B1 W LEVEL SW SCRAM
D502	SDIV A2 W ANLG TRP SCRAM
D503	SDIV B2 W ANLG TRP SCRAM
D504	MAIN STEAM LINE CHNL A1
D505	MAIN STEAM LINE CHNL B1
D506	MAIN STEAM LINE CHNL A2
D507	MAIN STEAM LINE CHNL B2
D508	CONTMT HIGH PRESS CH A1
D509	CONTMT HIGH PRESS CH B1
D510	CONTMT HIGH PRESS CH A2
D511	CONTMT HIGH PRESS CH B2
D512	REACTOR CHNL A1 HI PRESS
D513	REACTOR CHNL B1 HI PRESS
D514	REACTOR CHNL B2 HI PRESS
D515	REACTOR LO WTR LVL CH A1
D517	REACTOR LO WTR LVL CH B1
D518	REACTOR LO WTR LVL CH A2
D519	REACTOR LO WTR LVL CH B2
D520	MSL A-1 HIGH RADIATION
D521	MSL B-1 HIGH RADIATION
D522	MSL A-2 HIGH RADATION
D523	MSL B-2 HIGH RADIATION
D524	NEUT MON SYSTEM CHNL A1
D525	NEUT MON SYSTEM CHNL A2
D526	NEUT MON SYSTEM CHNL B1
D527	NEUT MON SYSTEM CHNL B2
D528	SDIV A1 E LEVEL SW SCRAM
D529	SDIV B1 E LEVEL SW SCRAM
D530	MANUAL SCRAM CHANNEL A
D531	MANUAL SCRAM CHANNEL B
D532	REACTER SCRAM CHANNEL A
D533	REACTOR SCRAM CHANNEL B
D534	BOTH SCRAM CHANNELS A&B
D535	SDIV A2 E ANLG TRP SCRAM
D536	SDIV B2 E ANLG TRP SCRAM
D538	TSV FAST CLOSURE CHNL A1
D539	TSV FAST CLOSURE CHNL B1
D540	TSV FAST CLOSURE CHNL A2
D541	TSV FAST CLOSURE CHNL B2
D542	TCV FAST CLOSURE CHNL A1
D543	TCV FAST CLOSURE CHNI B1
D544	TCV FAST CLOSURE CHNL A2

D545	TCV FAST CLOSURE CHNL B2
D546	APRM CHNL A UPSCALE LVL
D547	APRM CHNL B UPSCALE LVL
D548	APRM CHNL C UPSCALE LVL
D549	APRM CHNL D UPSCALE LVL
D550	APRM CHNL E UPSCALE LVL
D551	APRM CHNL F UPSCALE LVL

1.2.1.3 Time discrimination between events is one millisecond; (i.e., for two events occurring within less than one millisecond of each other, their sequence cannot be determined.)

1.2.1.4 The Sequence of Events log is printed on a printer in the control room. A facsimile of an actual printout of data point D532 (which was generated as a result of part of a routine surveillance test) is shown below:

TIME (hour, minutes and seconds)	MILLISECONDS	POINT ID	DESCRIPTION	STATUS (on-off trip-reset etc.)
093247	840	SEQ D532	REACTOR SCRAM	CHANNEL A TRIP
093250	719	SEQ D532	REACTOR SCRAM	CHANNEL A RSET

1.2.1.5 The process computer does not retain (store) Sequence of Events data except as part of its providing data printout. Once the data has been printed, it is no longer available within the computer. The Sequence of Events printout "hard-copy" is retained as part of the post trip review records.

1.2.1.6 Power for the process computer and Sequence of Events printer is supplied from the Uninterruptable Power Supply (UPS) system. The UPS is described in FSAR Section 8.9.

1.2.2 <u>Capability of Assessing the Time History of Analog Variables</u>

Equipment used to assess the time history of analog variables consists of the plant process computer's Post Trip Log function and a number of strip chart recorders. (See Table 2 and Appendix B).

1.2.2.1 <u>Brief Description of Equipment</u>

<u>Post Trip Log</u>

The Post Trip Log contains 20 selected plant inputs and is automatically initiated upon occurrence of predefined plant trips. It can also be initiated on operator demand.

<u>Strip Chart Recorders</u>

A number of strip chart recorders are available to the operator to assist in assessing the time history of analog variables.

1.2.2.2 <u>Parameters Monitored, Sample Rate & Selection Basis</u>

Table 2

POST TRIP LOG POINTS

B032	APRM A FLUX LEVEL	%PWR
B033	APRM B FLUX LEVEL	%PWR
B034	APRM C FLUX LEVEL	%PWR
B035	APRM D FLUX LEVEL	%PWR
B036	APRM E FLUX LEVEL	%PWR
B037	APRM F FLUX LEVEL	%PWR
B044	TOTAL CORE FLOW	M#/HR
B045	CORE DIFFERENTIAL	PRESS
B047	FDWTR LOOP A FLOW	M#/HR
B048	FDWTR LOOP B FLOW	M#/HR
B053	REACTER WATER LEVEL	INCH
B054	TOTAL STEAM FLOW	M#/HR
B057	REACTOR PRESSURE	PSIGN
B062	RX FW INLET A1 TEMP	DEGF
F204	MAIN STEM HEADER	PRESS
M017	DRYWELL PRESS	(ABSOLUTE)
M019	TOR WTR LVL (-72/+72)	INCH
M020	TOR WTR-AT (NRM-LMT=95)	
T0038	TB BYPASS VLV	POSITION %
T0040	TURBINE SPEED	RPM

Post Trip Log data is sampled at 2 second intervals by the process computer. Selection of parameters and sample rate were based on the recommendations of the NSSS vendor and limited by the installed equipment. Each of the 20 parameters monitored is limited to 120 data points. The 2 second sample rate is considered optimum for the equipment currently installed.

Strip Chart Recorders

Strip chart recorder data is recorded continously. Chart paper speed is normally one (1) inch per hour and each chart is generally date/time stamped daily for reference. Neutron monitoring recorders (07-PR-46 A,B,C,&D, 07-R-45) and the reactor water level recorder (06-LR/PR-97) may also be operated with a paper speed of one (1) inch per minute. This feature is only used during plant startup and scheduled shutdown. Appendix B lists the strip chart recorders (and the associated parameters) used to determine the cause of unscheduled reactor shutdowns. The parameters listed in Appendix B are used to complete the data recording for the post trip review procedure (Appendix A).

1.2.2.3 Duration of Time History

Post Trip Log

Post Trip Log data is continuously stored and updated at 2 second intervals in a portion of the process computer memory and remains in the memory for 2 minutes. Thus, memory contains the most recent 60 data bits for each Post Trip Log parameter prior to a plant trip. Upon occurrence of a pre-selected plant trip condition the plant process computer program prints the Post Trip Log. The Post Trip Log contains data for 2 minutes prior to the trip and data for 2 minutes after the trip at 2 second intervals totaling 120 data entries for each of the 20 Post Trip Log data points.

Strip Chart Recorders

As previously described, strip chart recorders provide a continuous record of the measured parameters value before and after a trip.

1.2.2.4 Data Format

Post Trip Log

Post Trip Log format is shown below:

	PT ID_1	PT ID_2	Pt ID_{20}
$Time_1$	$VALUE_1$	$VALUE_2$	$VALUE_{20}$
	±XXXXX	±XXXXX	±XXXXX
.	.	.	.
.	.	.	.
.	.	.	.
.	.	.	.
$Time_{120}$	$VALUE_1$	$VALUE_2$	$VALUE_{20}$
	±XXXXX	±XXXXX	±XXXXX

Strip Chart Recorders

Strip chart recorder format is typical of most strip chart recorders used in the industry today, (i.e., a strip chart approximately 5 inches wide with lines and numerals indicating the scale, with two color pen traces on the chart.)

1.2.2.5 Data Retention Capability

Post Trip Log

As noted in the response to Item 1.2.1.5, the computer does not retain (store) data used for functions (such as Post Trip Log) except as part of its data printout. Once this data is printed, it is no longer available within the computer. The Post Trip Log printout is retained as part of the post trip review records.

Strip Chart Recordings

Strip chart recordings are also retained as part of the plant Records Retention Program discussed above.

1.2.2.6 Power Source

Post Trip Log

Power for the process computer and printer for the Post Trip Log is from the Uninterruptable Power Supply (UPS) system which is described in FSAR Section 8.9

Strip Chart Recorders

Appendix B lists the power source for each strip chart recorder that may be used in assessing the time history of analog variables used to determine the cause of an unscheduled reactor shutdown.

1.2.3 In addition to the Sequence of Events printout, Post Trip Log printout and strip chart recordings listed in response to Items 1.2.1 and 1.2.2, the following information is generally available to assist in determining the cause of an unscheduled reactor shutdown.

1. Position indication for containment isolation valves and numerous other valves associated with Emergency Core Cooling Systems (ECCS), Reactor Water Recirculation (RWR), Main Steam, Reactor Water Cleanup (RWC), Reactor Core Isolation Cooling (RCIC) and balance of plant systems such as Feedwater, Condensate, Service Water and Emergency Diesel Generators.

2. Indication of system flow, system pressure, motor amperes, voltage, generator voltage, amperes and frequency, turbine speed (RPM) and similar indication of system initiation, operation or trip as appropriate for the system(s) of concern.

3. Annunciators in the Control Room and/or at local panels for plant system(s).

4. Protective relaying targets (flags) indicating relay operation and/or transmission system and, off-site power oscillograph.

5. Protective system and ECCS logic indicators.

6. Process computer alarm printout, periodic logs, special logs and video displays.

7. Statements relating to individual personnel actions, involvement or observations. Such statements are recorded as part of the critique which is conducted as part of post-event evaluation if the event involved any of the following:

- A complex evolution
- Obvious personnel error
- ECCS actuation
- Scram or main steam isolation
- Event cause or effects are not immediately evident
- The event report alone does not completely describe the event

These indicators (1 through 7 above) provide supplemental sources of information. The primary sources of information used to assess the cause of an unscheduled shutdown are the Sequence of Events Log, Post Trip Log and strip chart recordings listed in response to Items 1.2.1 and 1.2.2.

1.2.4 Schedule for Planned Changes

The Authority has contracted for the design, purchase and installation of a new computer system which will include a Safety Parameter Display System (SPDS) including certain displays associated with Emergency Operating Procedures and the eventual replacement of the existing process computer. This computer may also include some displays to satisfy the requirements of Regulatory Guide 1.97. Detailed specifications and a description of the system as it relates to Regulatory Guide 1.97, SPDS and Emergency Operating Procedures have not been finalized.

The completion date for the new system will be provided in accordance with our NUREG 0737 Supp. 1 commitments. The Authority considers the currently installed Sequence of Events Log, Post Trip Log and strip chart recorders to be adequate for post-trip review in the interim.

<u>Response to Item 2.1 - Equipment Classification and Vendor Interface (Reactor Trip System Components)</u>

The Component Quality Assurance (QA) Category List in existence at the James A. FitzPatrick Nuclear Power Plant has been reviewed. The review included verification that all components in the Reactor Protection System (RPS) (System 5) are presently classified as QA Category I except for the RPS Motor Generator Sets which are classified as QA Category II. Components in the RPS System are protected from Motor Generator Set malfunctions, such as over-voltage, under-voltage, and under-frequency conditions, by electrical protection assemblies which are classified Category I.

The QA Category Classification of other systems, such as Reactor Vessel Instrumentaion (System 02-3), Neutron Monitoring (System 07) and Process Radiation Monitoring (System 17) which comprise part of the "Reactor Trip Function", has been reviewed. All or part of these systems are classified as QA Category I, indicating that those portions of the systems which are associated with the "Reactor Trip Function" are properly classified. Reactor trip function components are classified QA Category I.

The Authority has performed a review of the documents, procedures and information handling systems used in the plant to control safety-related activites, including maintenance work requests (work orders), parts replacement and plant modifications. The documents, procedures and information handling systems concerned are controlled under the QA Program, or are identified as safety related and require review by the Plant Operations Review Committee. This provides assurance that maintenance, parts replacement and modification work is properly classified as QA Category I when required.

<u>Vendor Interface</u> - A formal Operating Experience Review Program is in effect and includes review of, and response to, General Electric BWR Service Information Letters (SILs). BWR SILs are used to document recommended changes in equipment and procedures, as well as convey information concerning unique operating conditions and experiences at BWR plants. The review and implementation of SILs is recorded and fed back to the General Electric Company (GE) using a standardized SIL Status Response Form. Periodically, a SIL Index is issued, assuring that all applicable information has been received. The General Electric Company SIL program, in conjunction with the Operating Experience Review Program, assures that complete, current and controlled NSSS vendor information is appropriately referenced or incorporated into plant specific procedures. Accordingly, the program ensures that reactor trip system vendor information is controlled throughout the life of the plant. No changes to the program are planned.

In addition, General Electric has established a reporting system to handle safety concerns that complies with the requirements of 10 CFR 21. General Electric also has established several other information systems which are described below.

<u>Urgent Communications</u> - A procedure for handling urgent communications to BWR owner/operators has been established by GE for use in providing fast notification of safety concerns. These communications are usually in the form of a short letter which provides a brief explanation and advice or precautionary measures to be observed to avoid potential operational hazards. Due to their urgent nature, these communications are sent to operating plants by the most effective method (i.e., telex, telecopy, cable, special mail handling, etc.).

<u>Service Advice Letters</u> - These documents are issued by GE product departments other than the San Jose based Nuclear Energy Product Departments and are used to provide notification of product problems and/or service information on a broad range of GE consumer and industrial products. Those Service Advice Letters that are recognized by the issuing product department as applying to devices used in

nuclear plants are specially identified and are flagged for distribution to all nuclear plants.

Turbine Information Letters (TILs) - TILs are issued by GE's Large Steam Turbine Generator Department to provide descriptions of product problems or improvements and to recommend modifications that will mitigate problems or improve product performance.

Operation and Maintenance Manuals - These documents are issued by all GE product departments to provide instructions for installation, operation and maintenance of GE designed repairable equipment and systems.

Application Information Documents - These documents describe potential operating problems and provide design change or operating recommendations to mitigate or avoid them. These documents are primarily aimed at requisition plants, but are also forwarded to operating plants when they have any applicability to those plants.

Urgent Communications, Service Advice Letters, TILS and/or Application Information Documents, are processed upon receipt using the existing Operating Experience Review Program.

The Operating Experience Review Program also provides for the systematic review of industry operating experience documents such as NRC I&E Bulletins, NRC I&E Information Notices, Licensee Event Reports, industry newsletters, INPO Significant Event Reports (SERs), INPO Significant Operating Experience Reports (SOERs) and the information received on the INPO Nuclear Network (formerly NOTEPAD).

Further the Authority actively participated in the INPO sponsored Nuclear Utility Task Action Committee (NUTAC) which was formed to address Section 2.2.2 of Generic Letter 83-28. While the Vendor Equipment Technical Information Program (VETIP) discussed in the NUTAC report specifically addressed improvement of non-NSSS vendor information exchange, it is anticipated that much of the VETIP information will be directly applicable to NSSS supplied equipment (including Reactor Trip System Components). It is anticipated that information received as a result of the VETIP will also be processed utilizing the existing Operating Experience Review Program.

Plant Standing Order (PSO) No. 28 entitled "Operating Experience Feedback", identifies responsibilities for review, feedback and incorporation of operating experience information into training programs. PSO 28 further requires audits by the Quality Assurance department. The NRC staff has received PSO 28 and found it to meet the requirements of Item I.C.5 of NUREG 0737 (NRC May 21, 1982 letter, D.B. Vassallo to L.W. Sinclair.)

Response to Item 2.2 - Equipment Classification and Vendor Interface (Program for all Safety-Related Components)

2.2.1.1 During the original classification of components at JAFNPP, the criteria for identifying components as safety-related within systems classified as safety-related was as follows:

> QA Category I is defined as those plant systems, or portions of systems, structures, and equipment whose failure or malfunction would cause a release of radioactivity that would endanger public safety. This category also includes equipment which is vital to a safe shutdown of the plant and the removal of decay and sensible heat, or equipment which is necessary to mitigate consequences to the public of a postulated accident.

This definition was interpreted to mean those structures, systems, and components that:

- Are necessary to assure the integrity of the reactor coolant pressure boundary.
- Are necessary to assure the capability to shutdown the reactor and maintain it in a safe shutdown condition.
- Are necessary to assure the capability to prevent or mitigate the consequences of accidents which could result in potential off-site exposures comparable to the guideline exposures of 10 CFR 100.
- Contain or may contain radioactive material and whose failure would result in conservatively calculated potential off-site doses which are more than 0.5 rem to the whole body or its equivalent to any part of the body.

This same criteria is presently used for identifying components as safety related and are currently included in the Engineering Design Procedures in use at JAFNPP.

2.2.1.2 The JAFNPP presently has a Component Quality Assurance Category List which identifies the safety-related components within safety-related systems. This list is issued and controlled by the JAFNPP Quality Assurance Department. The list was developed by a consultant. The consultant had a staff resident at JAFNPP and used the criteria given in response to 2.2.1.1 above. In addition to the stated criteria, the consultant used the following data:

- JAFNPP Final Safety Analysis Report (FSAR)
- Plant drawings provided by the Architect/Engineer (A/E)
- System descriptions provided by the A/E
- Instrument lists provided by the A/E
- Technical manuals provided by the NSSS vendor
- Vendor manuals and instructions which were provided by the equipment vendors

The consultant's staff performed walk-throughs of the plant and verified installation, name plate data, ratings, and other information, as part of the develoment of the list. Typical entries on the list are as follows:

- Type
- Category (technical)
- Component Description
- Component Number

- Data Reference
- Remarks
- Quality Assurance Category

The lists were compiled and cross checked by the consultant. They were then transmitted to the Site Quality Assurance Department for review and concurrence. Members of the Quality Assurance staff reviewed the lists for completeness and accuracy. Comments were returned to the consultants, as necessary, and when all comments were resolved, the Quality Assurance Department concurred with the lists. Revisions and/or changes to the list are controlled in accordance with approved plant procedures.

Since this update was completed, many plant modifications have been installed involving new equipment. Some of these modifications affect the Component Quality Assurance Category List. Since only minor revisions and additions have been made to the list since that update was completed, new components installed since 1978 may not appear on the list. As described in Sections 2.2.1.3.A and 2.2.1.3.B, Quality Assurance personnel review plant modification records to determine the category.

A program to further improve and update the Component Quality Assurance Category List is described in Section 2.2.1.6.

2.2.1.5. A controlled copy of the Component Quality Assurance Category List is issued to the Superintendent of each major plant department requiring the information. The Quality Assurance Category of plant components is thus readily available to personnel requiring the information.

Plant Administrative Procedures require that the Plant Operations Review Committee (PORC) review Administrative Procedures and other documents which affect nuclear plant safety, or impact on the environment. Procedures requiring PORC review are identified by an asterisk (*) after the title, thereby alerting personnel as to which procedures involve safety related considerations. The following procedures are in effect at JAFNPP which describe the controls and requirements which apply to safety-related activities.

- Administrative Procedures
- Work Activity Control Procedures
- Rules of Practice
- Quality Assurance Program
- Quality Assurance Procedures

In addition, the following departments maintain controlled departmental procedures which govern the conduct of safety-related work.

- Operations
- Instrument & Control
- Radiological & Environmental Services

- Maintenance
- Technical Services
- Quality Assurance
- Training

Further, the following controls are used at JAFNPP.

A. A Work Request Event Deficiency (WRED) form must be completed to initiate corrective maintenance. The WRED is routed through the Quality Control Department which reviews and verifies the Quality Assurance Category of the involved component. The WRED is also marked to indicate if Quality Control Inspection is required. In addition, for corrective maintenance performed on safety-related (Category I) components, or other work requiring QC inspection, the use of a Work Tracking Form (WTF) is required. A WTF is used to properly preplan, track, control and document corrective maintenance and provides sign-offs for the department peforming the activity, Quality Control personnel and the Operations Department.

B. The procurement of materials is initiated by a Purchase Requisition. Purchase Requisitions must be routed to Quality Control which verifies the Quality Assurance Category of the material, denotes if QC Receiving Inspection is required, and specifies the required documentation, test reports, etc., which must be included on the requisition. The requisition is then routed to Quality Assurance which checks and verifies the QC information, includes any further requirements to be imposed on the vendor and denotes the method used to qualify the vendor. When the Purchase Order (P.O.) for Category I material is typed, the P.O. (with a copy of the requisition) is routed to the Quality Assurance Department to verify that the requirements of the requisition have been correctly entered on the Purchase Order.

C. The following documents are required to have a Quality Assurance review and sign-off prior to implementation:

- Administrative Procedures
- Engineering Design Procedures
- Modification Control Forms
- Modification Documentation Tracking Forms
- Modification QA/QC and Design Requirements
- Modification Installation Procedures
- Preoperational Tests and Test Results
- Procurement Specifications

2.2.1.4 The mangement controls to verify that procedures for preparation, validation and routine use of the information handling system have been followed, are as follows:

A. The Plant Operation Review Committee reviews plant procedures and changes thereto, proposed tests and experiments, and proposed changes or modification to plant systems, that affect nuclear safety.

B. The plant departments utilize an internal departmental review.

C. The Quality Assurance Department implements an audit program to provide a comprehensive, independent evaluation of quality related procedures and activities to assure that they are in compliance with the Authority's established program requirements.

D. The QA & R Department, under the direction of the Safety Review Committee (SRC) Chairman and the Executive Vice President-Nuclear Generation, coordinates efforts to schedule an INPO or a Joint Utility Management Audit Group audit. (The Authority is a participant in a group of utilities for the purposes of performing independent assessments of QA activities.) The scope of the INPO or Joint Utility Management Audit Group includes, as a minimum, the activities performed by the Authority's QA & R Department. In addition, areas outside of this scope may be assigned by the SRC Chairman or the Executive Vice President-Nuclear Generation. The total audit program covers the 18 criteria of Appendix B to 10 CFR 50, within a 24 month period.

2.2.1.6 During I & E Inspection 84-11 and a visit by NRC Headquarters personnel from June 18, 1984 to June 22, 1984, NRC personnel reviewed the Equipment Classification Program at JAFNPP. As a result of this review, NRC personnel noted the following:

- The exiting Component Quality Assurance Category list contains inconsistencies.
- The existing Component Quality Assurance Category list is difficult to use.
- The Equipment Classification Program does not "trigger" generic reviews or updates of similar components when a component is identified as requiring review or reclassification.

The Authority agrees with these findings and commits to review the Component Quality Assurance Category List for completeness and accuracy. Considering the resource requirements, the desirability to change the format to enhance ease of use, and the advisability of integrating the program with other activities (such as preventive maintenance, corrective maintenance, parts procurement, and plant modifications), the Authority expects to complete this work by December 31, 1985.

Vendor Interface

2.2.2 As noted in response to Section 2.1, a formal Operating Experience Review Program is currently in effect. This program is based on the recommendations of the Institute of Nuclear Power Operatins (INPO) and provides for formal documented review of operating experience documents from both

Authority internal and external sources including NRC, INPO and the NSSS vendor. The JAFNPP is also an active participant in the Nuclear Plant Reliability Data System (NPRDS). NPRDS is an industrywide system managed by INPO for monitoring the performance of selected sytems and components at nuclear power plants. INPO's Significant Event Evaluation and Information Network (SEE-IN) attempt to ensure that the cumulative learning process from operating and maintenance experience is effective and that lessons learned are reported and corrective action taken in a timely manner to improve plant safety, reliability and availability.

The INPO sponsored NUTAC which was formed to plan a program to address Section 2.2.2. has completed its work. Copies of the NUTAC report were received by the participating utilities in early April, 1984. As part of this program, the NUTAC made several recommendations that would enhance and improve both the NPRDS and SEE-IN.

Enhancements recommended by NUTAC for NPRDS include:

- Expand definition of components to better describe mechanical components.
- Improved failure reporting guidance in areas of: analyzing role of piece parts in failures; inadequate vendor information as a failure cause; and improved failure analysis reports.
- Develop internal methods to assure clear and complete NPRDS reports.
- Provide follow-up NPRDS reports.
- Addition of new systems and components to program scope.

Similarly, the SEE-IN program expansion recommendations include:

- Reports prepared for potential failures due to faulty or missing vendor information or other equipment technical information.
- Broaden program to improve ability to trend NPRDS data.

Since both programs are administered by INPO, the Authority alone cannot assure that each of these enhancements will be fully implemented. Furthermore, the INPO staff responsible for executing these programs may suggest alternate means for enhancing NPRDS and SEE-IN.

Based upon our preliminary review of the report, the Authority endorse NUTAC's alternate approach for implementing the guidance of Section 2.2.2. of Generic Letter 83-28. The Authority plans to request INPO to incorporate NUTAC's recommendations.

Some of NUTAC's recommendations will require that new or revised procedures and programs be prepared and implemented by the Authority. Preliminary changes, based upon the available guidance, will be implemented by October 31, 1984. We expect that further changes may be necessary as the enhanced programs evolve.

Response to 3.1 - Post Maintenance Testing (Reactor Trip System Components)

3.1.1 The Authority has reviewed JAFNPP test and maintenance procedures and Technical Specifications to assure that post-maintenance operabilitity testing of safety-related components in the reactor trip system is required. Work Activity Control Procedure (WACP) 10.1.1 and Operations Department Standing Order 18 require post maintenance testing and provide guidance in the conduct of such testing to assure that the equipment is capable of performing its safety functions before being returned to service.

In addition, we have been reviewing our methods for determining the degree and extent of post maintenance testing. As a result of this review and a recent NRC Inspection (No.84-11 conducted June 18-22, 1984), written guidance for post maintenance testing will be prepared and implemented by September 30, 1984. This new guidance will improve the consistency and thoroughness with which post maintenance testing is conducted.

3.1.2 The Authority will formalize the Vendor Technical Manual Controls system and library at JAFNPP. This system will collect vendor equipment technical information (technical manuals, instructions, service advice etc.) in an organized, easily retrievable and controlled fashion. Access to this library will be controlled by means of a document "check-out" procedure. This library will be established by December 1, 1984.

After this program has been implemented and the library formalized, we will initiate a review of vendor technical manuals (or instructions) to verify that approprate vendor and engineering recommendations are incorporated in test and maintenance procedures, and technical specifications. Completion of this review, revision of test and maintenance procedures is expected to be completed by December 1, 1985.

3.1.3 A review of Technical Specifications has been conducted to determine if any post maintenance test requirements degrade safety. As a result of this review, the Authority is not aware of any post maintenance requirements in the Technical Specifications which degrade safety. The Authority notes, however, that BWROG activities to improve Technical Specifications may at some future date identify recommendated changes to tests required by Technical Specifications. These changes may involve surveillance frequency, allowable out-of-service intervals or recommended post maintenance tests, based on probabilistic risk assessment techniques which consider all or some of the considerations noted in Section 4.5.3.

Response to 3.2 - Post Maintenance Testing (All Other Safety-Related Components

3.2.1 The review of test and maintenance procedures and Technical Specifications discussed in the response to 3.1.1 is applicable to all safety-related equipment.

3.2.2 The formal review of vendor technical manuals and the revision of procedures and/or Technical Specifications discussed in response to 3.1.2 is applicable to all safety-related equipment.

3.2.3 The review of Technical Specifications discussed in response to 3.1.3 is applicable to all safety-related equipment. The BWROG activities discussed are also applicable to all safety-related equipment.

4.1 Reactor Trip System Reliability (Vendor-Related Modifications)

4.2 Reactor Trip System Reliability (Preventative Maintenance and Surveillance Program for Reactor Trip Breakers)

4.3 Reactor Trip System Reliability (Automatic Actuation of Shunt Trip Attachment for Westinghouse and B&W Plants)

4.4 Reactor Trip System Reliability (Improvements In Maintenance and Test Procedures for B&W Plants)

Response to 4.1, 4.2, 4.3 and 4.4

The James A. FitzPatrick Nuclear Power Plant is a boiling water reactor designed by General Electric. Therefore, Items 4.1, 4.2, 4.3 and 4.4 are not applicable.

Response to 4.5 - Reactor Trip System Reliability (System Functional Testing)

4.5.1 Reactor Trip System reliability is demonstrated by completion of the tests and calibrations required by Technical Specification Table 4.1-1 in conjunction with Technical Specification required control rod scram time testing which periodically demonstrates function and reliability of the entire Reactor Trip System. Backup scram valves are not required to be tested by Technical Specifications and the system design does not permit on-line functional test. The Authority does not believe that any significant improvement in system reliability would be achieved if the system were modified to permit on-line functional test and such testing was periodically performed.

4.5.2 As an alternative to on-line test of backup scram valves, the Authority proposes to implement functional testing of the backup scram valves once each refueling cycle while the plant is shutdown as part of the JAFNPP test program. The Authority notes the NRC found acceptable similar testing of backup scram valves in NUREG 0979, April 1983, entitled "Safety Evaluation Report related to the Final Design Approval of the GESSAR II, BWR-6 Nuclear Island Design."

4.5.3 The Authority has performed a review of Technical Specifications to determine if the test intervals specified are consistent with achieving high reliability.

The review did not consider:

- Personnel errors during testing
- Reduced redundancy during testing, or
- Uncertainty in common mode failure rates

In general, our review did not reveal excessive testing (as indicated by excessive component "wear-out") or infrequent testing (as indicated by high failure rates.)

The Authority in a January 18, 1984 letter (JPN-84-01, J.P. Bayne to D.B. Vassallo) commented on the effects of diesel generator cold fast starts. While no forced outages have resulted from diesel generator testing, the long-term impact on reliability and availability are known to be negative. We have not quantified the degree of these negative effects.

The Authority is participating in the development of a BWR Owners' Group (BWROG) Technical Specification Improvements program. This program will include the development of a method for the review of intervals for on-line functional testing required by Technical Specifications. A generic methodology will be developed to show the sensitivity of system unavilability to changes in:

- Comp nent failure rates
- Com on mode failure rates
- Reduced redundancy during testing
- Human error rates during testing, and
- Component "wear-out" rates caused by testing.

The Authority plans to apply the results of this program to JAFNPP.

The schedule for this BWROG activity is currently being prepared by the Technical Specification Improvements Committee. Accordingly, the Authority cannot provide a schedule for this work at this time.

NEW YORK POWER AUTHORITY

James A. FitzPatrick Nuclear Power Plant

Appendix A to JPN-84- 42

Operations Department Standing Order No.23

Post Trip Review

Instrument No.	Parameter Recorder	Range	Power Source
07-PR-46A, B, C & D	(Red) APRM B, D, & F or IRM B, D, F, & G	0 to 125	Note 6
	(Black) APRM A, C, & E or IRM A, C, E, & H	0 to 125	Note 6
07-R-45	(Red) SR, B or D	0.1 to 10^6 CPS	Note 6
	(Black) SRM A or C	0.1 to 10^6 CPS	Note 6

Notes for Appendix B:

1. 120V AC Safeguard Control and Instrument Bus B1 (71-ESS-B1). Power is derived from the "B" Emergency Power 4 KV Bus (Bus 10600). See FSAR Figure 8.9-1.

2. 120V AC Safeguard Control and Instrument Bus A1 (71-ESS-A1). Power is derived from the "A" Emergency Power Bus (Bus 10500). See FSAR Figure 8.9-1.

3. 120V AC Common Control and Instrument Bus 9 (71-AC-9). Power is derived from eith the "A" or "B" Normal Power Bus (Bus 10300). See FSAR Figure 8.9-1.

4. 120V AC Emergency Control and Instrument Bus A2 (71-AC-A2). Power is derived from the "A" Emergency Power 4 KV Bus (Bus 10500). See FSAR Figure 8.9-1.

5. 120V AC Emergency Control and Instrument Bus B2 (71-AC-B2). Power is derived from the "B" Emergency Power 4 KV Bus (Bus 10600). See FSAR Figure 8.9-1.

6. 120V AC Uninterruptable Power Supply (UPS). The UPS motor generator is driven by either AC or DC power. AC power is derived from the "B" Emergency Power 4 KV Bus (Bus 10600) and the DC power is from the "A" 125V DC Station Battery. Maintenance power (used when the motor generator is out of service) is derived from the "A" Emergency Power 4 KV Bus (Bus 10500). See FSAR Figure 8.9-1.

NEW YORK POWER AUTHORITY
James A. FitzPatrick Nuclear Power Plant

Appendix B to JPN-84- 42

Strip Chart Recorders

Instrument No.	Parameter Recorder	Range	Power Source
06-LR/PR-97	(Red) Reactor Pressure	0 to 1200 psig	Note 6
	(Black) Reactor Water Level	164.5 to 224.5 inches above TAF	Note 6
06-FR-96	(Red) Reactor Steam Flow	0 to $12x10^6$ lbs/hr.	Note 6
	(Black) Feedwater Flow	0 to $12x10^6$ lbs/hr.	Note 6
06-PR/FR-98	(Red) Reactor Pressure	800 to 1100 psig	Note 6
	(Black) Turbine Steam Flow	0 to $10x10^6$ lbs/hr.	Note 6
02-3-LR-98	Reactor Water Level (Fuel Zone)	-100 to +200 Inches below (-) or above (+) TAF	Note 1
10-FR-143	(Red) LPCI Loop A Flow	0 to $25x10^3$ gpm	Note 2
	(Black) LCPI Loop B Flow	0 to $25x10^3$ gpm	Note 1
02-FR-163A&B	(Red) Recirc. Loop A Flow	0 to $70x10^3$ gpm	Note 3
	(Black) Recirc. Loop B Flow	0 to $70x10^3$ gpm	Note 3
27-PR-115A1&A2	(Red) Primary Contain. Pressure	0 to 250 psig	Note 4
	(Green) Primary Contain. Pressure	-5 tp +5 psig	Note 4
27-PR-115B1&B2	(Red) Primary Contain. Pressure	0 to 250 psig	Note 5
	(Green) Primary Contain. Pressure	-5 to +5 psig	Note 5
02-3-FR/PR-95	(Red) Core Differential Pressure	0 to 25 psid	Note 6
	(Black) Core Flow	0 to $90x10^6$ lbs/hr.	Note 6
6-PR-61A&B	(Red) Reactor Pressure	0 to 1500 psig	Note 4
	(Green) Reactor Pressure	0 to 1500 psig	Note 5
23-LR-202A & 203A	(Red) Suppression Pool Level	1.7 to 27.5 feet	Note 4
	(Green) Drywell Level	22 to 106 feet	Note 4
23-LR-202B & 203B	(Red) Suppression Pool Level	1.7 to 27.5 feet	Note 5
	(Green) Drywell Level	22 to 106 feet	Note 5