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New York Power Authority

J. Phillip Bayne
Executive Vice President
Nuclear Generation

November 9, 1983
JPN-83-92

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

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

Subject: James A. FitzPatrick Nuclear Power Plant
Docket No. 50-333
Response to Generic Implications of Salem
ATWS Events (Generic Letter 83-28)

References:
1. NRC Generic Letter 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).
3. NRC letter, D. B. Vassallo to J. P. Bayne, dated October 19, 1983.

Dear Sir:

Reference 1 requested information, plans and schedules relating to the generic inplications of the March, 1983 Salem ATWS events. Reference 2 provided a preliminary schedule for submitting the requested information. In addition, Reference 2 requested an extension for those items for which a detailed response cannot be provided by November 7, 1983.

Reference 3 denied the extension and requested that information be submitted by November 5, 1983. The attachment to this letter provides the Authority's response to the extent practical at this time. Specifically, the attachment includes:

1. A description of the current (as of November, 1983) programs and status for each item.
2. A description of plans for changes to the current programs and/or procedures for each item.

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In a number of cases, the Authority cannot provide definative schedules for the completion of efforts to meet the requirements of Reference 1. The Authority is participating in both the BWROG's (Boiling Water Reactor Owners Group) and NUTAC's (Nuclear Utilities Task Action Committee) generic efforts to address portions of Reference 1. We expect to be able to provide our plans and implementation schedules shortly after these groups finalize their efforts.

The information, plans and schedules in this submittal are provided in accordance with the information available at this time and the reviews completed to date. The Authority reserves the right to amend this submittal if necessary. If, as a result of this review, a revision is required, it will be submitted promptly.

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

Very truly yours,

for

J. P. Bayne
Executive Vice President
Nuclear Generation

State of New York
County of Westchester

Subscribed and Sworn to before me this 9 day of November 1983

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........

NEW YORK POWER AUTHORITY

JAMES A. FITZPATRICK NUCLEAR POWER PLANT

RESPONSE TO GENERIC LETTER 83-28

Attachment 1 to JPN-83-92
Dated November 9, 1983

This document provides a description of the current procedures or programs for each item identified in NRC Generic Letter 83-28. For those items which are not currently addressed in procedures or programs, a description of the Authority's position or plans is provided. The current best estimated date for completion is also provided, wherever sufficient information exists to project a date.

ITEM 1.1 POST-TRIP REVIEW (PROGRAM DESCRIPTION AND PROCEDURE)

A. CURRENT POST-TRIP REVIEW PROGRAM

1. There is no formal program in place which addresses all the elements outlined in the generic letter for post-trip review and restart.

2. The following actions have always been taken when an unexplained shutdown has taken place.

   a. Knowledgeable plant personnel conduct a thorough review of the transient.

   b. In cases where the cause is not clear, an intensive analysis is initiated to determine the cause. Documentation exists to support the analysis.

   c. There are only a few transients where the exact cause was not identified with a good level of confidence.

   d. Restart authorization from the Plant Manager or his designated alternate is required. This requirement is a sign-off step in F-OP-65, Startup and Shutdown Procedure.

B. PLANNED CHANGES TO THE POST-TRIP REVIEW PROGRAM

1. A new procedure has been drafted for post-trip review. This new procedure will be implemented by January 6, 1984 and will address Items 1.1.1 through 1.1.6 of Generic Letter 83-28.

2. The BWROG is addressing Item 1.1.1 (restart criteria), Item 1.1.3 (qualification and training) and Item 1.1.5 (methods/criteria for event comparison). Funding for these activities was approved by the BWROG at the October 26/27, 1983 meeting and General Electric is expected to complete the work by February 29, 1984. Allowing for possible revision of the recently drafted procedure, the submittal of

a detailed report containing the final program description and procedure will be made by March 31, 1984.

ITEM 1.2 POST-TRIP REVIEW - DATA AND INFORMATION CAPABILITY

A. CURRENT DATA AND INFORMATION CAPABILITY

1. Capability for Assessing Sequence of Events

The sequence of events function is provided primarily by the plant process computer Sequence of Events log. The Sequence of Events log is initiated (printed) by a change in state of any of the digital (on-off) inputs selected for the sequence of events function. Currently 161 points are utilized of which 45 are designated as spares. The 116 points used at this time are listed below:

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	RESERV STATION TFR T2 UV
D032	RESERV 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
D041	SONIC DETECTOR RV-2-71F
D042	SONIC DETECTOR RV-2-71G
D043	SONIC DETECTOR RV-2-71H
D044	SONIC DETECTOR RV-2-71J
D045	SONIC DETECTOR RV-2-71K
D046	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 A2 HI PRESS
D515	REACTOR CHNL B2 HI PRESS
D516	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 RADIATION
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	REACTOR 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 CHNL 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
D554	IRM CHNL A UPSCALE LVL
D555	IRM CHNL B UPSCALE LVL
D556	IRM CHNL C UPSCALE LVL
D557	IRM CHNL D UPSCALE LVL
D558	IRM CHNL E UPSCALE LVL
D559	IRM CHNL F UPSCALE LVL
D560	IRM CHNL G UPSCALE LVL
D561	IRM CHNL H UPSCALE LVL
D562	WEST SDIV NOT DRAINED
D563	EAST SDIV NOT DRAINED
D564	WEST SDIV ROD BLOCK
D565	EAST SDIV ROD BLOCK

Time discrimination between events is one millisecond; i.e., for two events occurring within less than one millisecond of each other, the correct sequence cannot be guaranteed.

The Sequence of Events is printed on a typewriter in the control room. A facsimile of the 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

The process computer does not retain (store) Sequence of Events function data except as part of its providing data printout. Once

the data is provided to the printer it is no longer available within the computer. The Sequence of Events printout hard copy is retained as part of the records retention program. Power for the process computer and printer for the Sequence of Events is from the Un-interruptable Power Supply (UPS) system which is described in FSAR Section 8.9.

2. Capability of Assessing the Time History of Analog Variables

   Equipment used to assess the time history of analog variables consists of the plant process computer Post Trip Log function and a number of strip chart recorders.

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

   b. Strip chart recorders (which are part of the numerous indicators and recorders provided for routine startup, operation, and plant shutdown) also provide data and information which may be useful for analysis of unscheduled shutdown and/or the functioning of safety-related equipment.

   Parameters monitored for assessing the time history of analog variables are listed below for both the Post Trip Log and those strip chart recorders that may also be utilized.

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	REACTOR WATER LEVEL	INCH
B054	TOTAL STEAM FLOW	M#/HR
B057	REACTOR PRESSURE	PSIG
B062	RX FW INLET A1 TEMP	DEGF
F204	MAIN STEAM HEADER	PRESS
M017	DRYWELL PRESS	(ABSOLUTE)
M019	TOR WTR LVL (-72/+72)	INCH
M020	TOR WTR-AT	(NRM LMT=95)
T038	TB BYPASS VLV	POSITION %
T040	TURBINE SPEED	RPM

STRIP CHART RECORDERS

RECORDER ID	DESCRIPTION	
06-LR/PR-97	REACTOR PRESSURE REACTOR WATER LEVEL	0 to 1200 psig 164.5 to 224.5 inches above Top of Active Fuel (TAF)
06-FR-96	REACTOR STEAM FLOW FEEDWATER FLOW	0 to 12 ($x10^6$) lbs/hr. 0 to 12 ($x10^6$) lbs/hr.
06-PR/FR-98	REACTOR PRESSURE TURBINE STEAM FLOW	800 to 1100 psig 0 to 100% of rated
02-3-FR/PR-95	CORE DP CORE FLOW	0 to 25 psid 0 to 90 ($x10^6$) lbs/hr.
07-PR-46A,B,C,&D	APRM A,B,C,D,E,&F IRM A,B,C,D,E,F,G&H	0 to 125% Power 0 to 125
10-FR-143	LPCI LOOP A FLOW LPCI LOOP B FLOW	0 to 25 ($x10^3$) gpm 0 to 25 ($x10^3$) gpm
02-3-LR-98	REACTOR WATER LEVEL (FUEL ZONE)	-100 to +200 inches below (-) or abcve (+) TAF

02-TR-165	RECIRCULATION LOOP A TEMP RECIRCULATION LOOP B TEMP	0 to 600°F 0 to 600°F
02-FR-163	RECIRCULATION LOOP A FLOW RECIRCULATION LOOP B FLOW	0 to 70 ($x10^3$) gpm 0 to 70 ($x10^3$) gpm
94-TS-VP	TUBINE BYPASS VALVE POSITION	0 to 100%
07-R-45	SRM B or D SRM A or C	10^{-1} to 10^6 CPS 10^{-1} to 10^6 CPS

Post Trip Log data i. 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 the 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 causes the data which was stored for 2 minutes prior to the trip (and new data at two second intervals for the same 20 parameters for a 2 minute period after the trip) to printed out as the Post Trip Log. The Post Trip Log therefore 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.

Selection of the parameters currently used was based on the recommendation of the NSSS vendor during initial startup of the plant, and no substanital change to the 20 parameters monitored has taken place since that time. Sample rate for the Post Trip Log parameters is limited by the capability of the plant process computer and its program. Each of the 20 parameters monitored is limited to 120 data points. Selection of the 2 second sample rate is considered optimum, given the limitations of the currently installed equipment.

Strip chart recorder data which may also be used is recorded continuously. Chart paper speed is normally one (1) inch per hour and each chart is generally date/time stamped on a daily basis 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 chart paper speed of one (1) inch per minute. However this feature is normally used by the operator only during plant startup and scheduled shutdown.

Post Trip Log Format is shown below:

Time	PT ID_1	PT ID_2	Pt ID_{20}
XXXXXX	XXXX	XXXX	XXXX
	$VALUE_1$	$VALUE_2$	$VALUE_{20}$
	$\pm$XXXXX	$\pm$XXXXX	$\pm$XXXXX
$Time_{120}$	$VALUE_1$	$VALUE_2$	$VALUE_{20}$
XXXXXX	$\pm$XXXX	$\pm$XXXX	$\pm$XXXX

Strip chart recorder format is typical of most strip chart recorders used in the industry today, i.e., a strip chart 6 inches wide with lines and numerals indicating the scale, with red and black pen traces on the chart.

Power sources for the Post Trip Log and the Strip Chart recorders listed above are as follows:

PROCESS COMPUTER (POST TRIP LOG AND ASSOCIATED PRINTER)	UN-INTERUPTABLE POWER SUPPLY (UPS) SYSTEM (FSAR SECTION 8.9)

Strip Chart Recorders

Recorder ID	
06-LR/PR-97	UPS
06-FR-96	UPS
06-PR/FR-98	UPS
02-3-FR/PR-95	UPS
07-PR-46A,B,C,&D	UPS
10-FR-143	Loop A - Safeguard Control & Instrument BUS A1 (71-ESS-A1) (FSAR) Figure 8.9-1) Loop B - 71-ESS-B1 (FSAR Figure 8.9.1)
02-3-LR-98	71-ESS-B1 (FSAR Figure 8.9-1)
02-TR-165	Common Control & Instrument BUS 9 (71-AC-9) (FSAR Figure 8.9-1)
02-FR-163	71-AC-9 (FSAR Figure 8.9-1)
94-TS-VP	71-AC-9 (FSAR Figure 8.9-1)
07-R-45	UPS

B. PLANNED CHANGES TO DATA AND INFORMATION CAPABILITY

The Authority is in the process of reviewing bids on a new computer system including an SPDS, which will ultimately result in improvement of the data and information capabilities at the plant. Detailed specifications and proposed delivery/installation dates for this new equipment will not be finalized until negotiations with the successful bidder are complete. 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 Event system and strip chart recorders to be adequate for post-trip review in the interim period.

Item 2.1 EQUIPMENT CLASSIFICATION AND VENDOR INTERFACE (REACTOR TRIP SYSTEM COMPONENTS).

A. CURRENT EQUIPMENT CLASSIFICATION AND VENDOR INTERFACE (REACTOR TRIP SYSTEM COMPONENTS).

The component Quality Assurance Category List in existence at the James A. FitzPatrick Nuclear Power Plant has been reviewed. The review included verification that all components in System 5- Reactor Protection System (RPS) 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 Instrumentation (System 02-3), Neutron Monitoring (System 07) and Process Radiation Monitoring (System 17) which comprise part of the "Reactor Trip Function", has been preliminarily 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. As discussed below (under Planned Changes) additional reviews will be completed to assure that all Reactor

Trip Function components are classified QA Catetory I.

The Authority has performed an initial, preliminary review of the documents, procedures and information handling systems used in the plant to control safety-related activities, 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. The Authority will provide a schedule for a complete indepth review of these items by March 31, 1984.

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 using a standardized SIL Status Response form. Periodically a SIL Index is issued, assuring that all applicable information has been received.

B. PLANNED CHANGES TO EQUIPMENT CLASSIFICATION AND VENDOR INTERFACE (REACTOR TRIP SYSTEM COMPONENTS)

The BWROG is addressing Item 2.1-Equipment Classification and Vendor Interface for Reactor Trip System Components. A BWROG committee has been formed to specifically address Generic Letter 83-28. The committee has sub-divided Item 2.1 into several subtasks for ease of handling and in order to allow generic response to the maximum extent practical. The subtasks and the status of each are provided below:

Subtask 1 - Reactor Trip System Equipment List

The BWROG has discussed several options (or alternatives) within this subtask, which have varying degrees of generic application to the individual BWROG members. This is due to the difference in plant design such as product line (BWR 2 through BWR 6) and containment type (Mark I, II and III). At this time a review to achieve the most effective use of available resources continues. The Authority expects a decision on the BWROG course of action to be finalized by February 29, 1984. Therefore the Authority will update the status of this item by March 31, 1984. The Authority considers the review of the Component Quality Assurance Categoty List discussed above, to be adequate in the interim.

Subtask 2 - Updating Operations And Maintenance Manuals For All Safety Related Equipment (RTS And Other)

The BWROG has discussed this subtask with General Electric. Some generic benefit can be realized by the BWROG members by grouping the BWR plants into several classifications. Dis[illegible]sion as to the best course of action continues. The Authority expects the BWROG course of action to be finalized by February 29, 1984. Therefore, the Authority will update the status of this item by March 31, 1984.

Subtask 3 - Description Of General Electric Information Programs

The BWROG approved funding during the October 26-27, 1983 meeting for this subtask. General Electric is expected to [illegible]mplete the description of how information concerning operation and maintenance is obtained from their suppliers, with emphasis in those areas intended to provide support to the plant throughout its lifetime. General Electric will also provide a description of the SIL system and other communication channels which provide plants with information not encompassed by the SIL system. This work is expected to be complete prior to February 29, 1984. Therefore, the Authority will provide additional information on this subtask by March 31, 1984.

Subtask 4 - Procedure For Safety Related Equipment Identification

The BWROG approved funding for this subtask during the October 26-27, 1983 meeting. General Electric will provide BWROG members with a description of the procedures and the process involved in classification of equipment and components, as safety related and non-safety related, at various levels of equipment complexity. Work is expected to be complete prior to February 29, 1984. As a result of this effort, changes may be made to the Component Quality Assurance Category List described above. The Authority will provide additional information on this subtask by March 31, 1984.

Subtask 5 - Evaluation Of Scope Of Vendor Interface Program (VIP)

The BWROG has deferred action on this substask until the Nuclear Utility Task Action Committee (NUTAC), which is addressing Item 2.2.2, has substantially completed its recommendations. Due to uncertainty with respect to the ultimate BWROG action on this subtask and the current status of the NUTAC work on Item 2.2.2, the Authority cannot estimate a schedule for Reactor Trip System components vendor interface program until March 31, 1984. However, the Authority notes that the General Electric information programs discussed under Subtask 3 provide vendor interface for most of the equipment provided within the NSSS vendor scope of supply.

Item 2.2 EQUIPMENT CLASSIFICATION AND VENDOR INTERFACE (PROGRAM FOR ALL SAFETY RELATED COMPONENTS)

A. Current Equipment Classification and Vendor Interface Program for All Safety Related Components.

Equipment Classification

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

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.

The above definition should be interpreted to mean those structures, systems, and components that:

a. Are necessary to assure the integrity of the reactor coolant pressure boundary.

b. Are necessary to assure the capability to shutdown the reactor and maintain it in a safe shutdown condition.

c. 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.

d. 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.

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

2. The JAFNPP presently has a Component Assurance Category List which identifies the safety-related components within the systems and is issued and controlled by the Quality Assurance Department at JAFNPP. The list was developed as part of a contract with a consultant. The consultant had a resident staff at JAFNPP and followed the criteria given in 1. above. In addition to the stated criteria, the consultant used the following data:

- JAFNPP FSAR
- Plant Drawings provided by the 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 on-site staff performed walk-throughs of the plant and verified installation, name plate data,

ratings, and other information, as part of the development 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 complied 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.

The list was originally developed in 1978. Only minor revisions/additions have been made to the list since that time. As a result, components installed as a result of plant modifications which were completed since 1978, are not included. To overcome these omissions, Quality Assurance personnel perform a review of Plant Modification records to ascertain the correct category when performing the reviews outlined in 3.1 and 3.2 below.

3. A controlled copy of the Component Quality Assurance Category List is issued to the Superintendent of each major plant department which requires the information. The Quality Assurance Category of plant components is thus readily available to personnel requiring the information. In addition, Plant Administrative Procedures require that the Plant Operations Review Committee review Administrative Procedures and documents affecting nuclear plant safety, or impacting 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

Radiation & Environmental Services

Maintenance

Technical Services

Quality Assurance

Training

In conjunction with a recent commitment to the NRC, the work procedures in use at JAFNPP for safety related activities are in an on-going process of revision to include Quality Control and Radiation Protection Hold Points where required.

In addition, the following controls are used at JAFNPP:

a. A Work Request Event Deficienty Form (WRED) must be filled out to initiate corrective maintenance. The WRED is routed through Quality Control which reviews and verifies the Quality Assurance Category of the involved component. The WRED is also marked to denote 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 is required. This form is used to properly pre-plan. track, control and document corrective maintenance and provides for sign-offs by the department performing the activity, Quality Control personnel and the Operations Department.

b. The procurement of all materials is initiated by a Purchase Requisition. The Purchase Requisition 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 for Category I material is typed, the P.O., with a copy of the requisition, is routed to Quality Assurance 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
- Pre-Operational Tests and Test Results

4. The management controls utilized to verify that the procedures for preparation, validation and routine utilization of the information handling system have been followed, are as follows:

a. The Plant Operation Review Committee reviews:

1) Plant procedures and changes thereto, which affect nuclear safety.

2) Proposed tests and experiments that affect nuclear safety.

3) 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 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 and R Department. In addition, areas outside of the scope may be assigned as directed by the SRC Chairman or the Executive Vice President-Nuclear Generation. The total audit program covers the 18 criteria of Appendix B to 10CFR50, within at least a 24 month period.

Vendor Interface

The review of General Electric Service Information Letters (SILs) discussed above in the response of Item 2.1 applies equally to systems and components outside of the Reactor Trip System, where those systems or components were within the NSSS Vendor scope of supply. In addition, Authority review of industry and INPO generated operating experience

documents, such as Licensee Event Reports and Significant Operating Experience Reports (SOER), is currently being upgraded and will provide a greater measure of vendor interface for safety related systems and components outside of the NSSS vendor scope of supply.

B. PLANNED CHANGES TO EQUIPMENT CLASSIFICATION AND VENDOR INTERFACE FOR ALL SAFETY RELATED EQUIPMENT

Equipment Classification

The BWROG activities discussed in the responses to Item 2.1 above are integrated to some extent with the requirements for Item 2.2. Specifically Subtask 4 (procedure for safety related component identification) is equally applicable to Item 2.1 and 2.2. As noted above in the response to Item 2.1, the BWROG expects General Electric to complete the work related to safety-related component identification prior to February 29, 1984. In addition, the Authority is in the process of converting a number of component identification, parts purchasing, warehousing and other functions to a computer based system. It is expected that the current Component Quality Assurance Category List will be computerized by mid-1985. Accordingly, a schedule for changes and additions associated with modifications and/or as a result of the BWROG activities discussed above, will be provided by March 31, 1984.

Vendor Interface

The Authority is a member of the Nuclear Utility Task Action Committee (NUTAC) formed to address Item 2.2.2. The Authority will review the results or recommendations of the NUTAC following its scheduled completion of work in February, 1984. Accordingly, the Authority will provide

additional information with respect to vendor interface by March 31, 1984.

Item 3.1 <u>POST MAINTENANCE TESTING (REACTOR TRIP SYSTEM COMPONENTS)</u>

Item 3.2 <u>POST MAINTENANCE TESTING (ALL-OTHER SAFETY-RELATED COMPONENTS)</u>

A. <u>CURRENT POST MAINTENANCE TESTING (REACTOR TRIP SYSTEM AND ALL OTHER SAFETY-RELATED COMPONENTS)</u>

1. Post Maintenance Testing is addressed for safety-related maintenance at the FitzPatrick plant. For safety-related work, Work Activity Control Procedure (WACP) 10.1.1 (Procedure for Control of Maintenance) requires testing of equipment following maintenance and, requires that Quality Control be notified of the impending Post Maintenance Test. Operations Department Standing Order 18 and WACP 10.1.1 provide instructions and guidance to the Shift Supervisor and Quality Control Personnel with respect to test performance, inspection and acceptance, assuring that all safety related equipment is operable when it is returned to service. The requirements of Items 3.1.1 and 3.2.1 are fully met.

2. No formal program exists to verify that vendor and engineering recommendations relating to test guidance is included in test and maintenance procedures at this time.

3. A review of Technical Specification was conducted to determine if any post-maintenance test requirements degraded safety. The review did not identify any cases of obvious safety degradation.

B. PLANNED CHANGES TO POST MAINTENANCE TESTING (REACTOR TRIP SYSTEM AND ALL OTHER SAFETY-RELATED COMPONENTS)

1. As noted above, the Authority considers the requirements of Items 3.1.1 and 3.2.1 to be fully met. Therefore, no changes are planned.

2. The Authority plans to implement a program which will incorporate both current and future vendor and/or engineering recommendations, as appropriate, into Test Performance, Maintenance, Post-Maintenance Tests or Technical Specifications.

   A major revision of maintenance procedures directed to improving the quality and scope of the procedures, as well as addition of Quality Control and ALARA hold points where appropriate, has commenced. This program, in response to an NRC inspection, is currently scheduled for completion in mid-1985. Incorporation of current and future vendor and/or other engineering recommendations, as well as other potential changes resulting from vendor interface programs, is being considered. In view of the magnitude of the programs, and the potential impact on

the mid 1985 completion commitment, additional evaluation is required before a firm commitment with respect to Items 3.1.2 and 3.2.2 can be made. The Authority expects to complete the evaluation of the impact within the next few months and will provide additional information by March 31, 1984.

3. As indicated above the Authority is not aware of any post-maintenance requirement in the Technical Specifications which degrades safety. The Authority notes, however, that BWROG activities associated with improvements in Technical Specifications may at some future date identify recommendated changes to tests required by Technical Specifications. These potential changes may involve surveillance frequency, allowed out-of-service interval or recommended post-maintenance tests, based on probabilistic risk assessment techniques which consider all or some of the same considerations noted in Item 4.5.3.

ITEMS 4.1, 4.2, 4.3, & 4.4 REACTOR TRIP SYSTEM RELIABILITY

The James A. FitzPatrick 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.

Item 4.5 REACTOR TRIP SYSTEM RELIABILITY (SYSTEM FUNCTIONAL TESTING)

A. CURRENT REACTOR TRIP SYSTEM RELIABILITY (SYSTEM FUNCTIONAL TESTING)

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 Authority does not consider any test necessary.

2. The Reactor Trip System installed at the James A. FitzPatrick Nuclear Power Plant is designed to permit on-line testing and such tests are routinely performed to meet the requirements of Technical Specifications as indicated in the response to Item 4.5.1 above, except for the backup scram valves. The design of the backup scram does not permit a qualitative on-line test. The BWROG is addressing Item 4.5.2 and funding was approved at the October 26-27, 1983 meeting for General Electric to develop detailed justification for the position that on-line testing of backup scram is not required.

3. The Authority has performed a preliminary review of Technical Specifications to determine if the test intervals specified are consistent with achieving high reliability. This preliminary review did not address the considerations listed in Item 4.5.3.1 through 4.5.3.5 due to the short time available to evaluate those considerations and due to the activities of the BWROG Technical Specifications Improvements Committee (TSIC).

B. PLANNED CHANGES TO REACTOR TRIP SYSTEM RELIABILITY (SYSTEM FUNCTIONAL TESTING)

1. No changes to testing of the Reactor Trip System are planned. As noted above, the backup scram design at James A. FitzPatrick Nuclear Power Plant does not permit performance of a qualitative periodic on-line test of the backup scram feature, and the Authority does not consider such a test necessary. The BWROG position on this matter is expected to provide detailed justification to that end and will be provided by March 31, 1984, following completion of the General Electric work.

2. On going work by the BWROG Technical Specifications Improvement Committee (TSIC) is addressing the general subject of surveillance frequency determination. This work is intended to consider a number of uncertainties,

human error probabilities, component "wear-out" and considerations similar those listed in Item 4.5.3. The BWROG TSIC recently became aware that the Electric Power Research Institute (EPRI) and the NRC are also conducting or sponsoring work in the same general area. As a result, the BWROG TSIC recently recommended to the BWROG that funding for development and approval of the techniques be defferred until additional evaluation of the related EPRI and/or NRC work can be conducted.

While the Authority is not aware of the date that the NRC sponsored work is expected to be complete it is known that the EPRI work is expected to be complete in the fall of 1984. Accordingly the Authority cannot provide any meaningful schedule of possible future work in this area until at least the fall of 1984.