FLORIDA POWER AND LIGHT COMPANY
TURKEY POINT UNITS 3 AND 4
ADMINISTRATIVE PROCEDURE 0103.16
JULY 29, 1982

1.0 Title:

DUTIES AND RESPONSIBILITIES OF THE SHIFT TECHNICAL ADVISOR

2.0 Approval and List of Effective Pages:

2.1 Approval:

Change Dated 7/29/82 Reviewed by Plant Nuclear Safety Committee: 82-69

and Approved by Plant Manager - Nuclear: 7/29/82

2.2 List of Effective Pages:

Page	Date	Page	Date	Page	Date	Page	Date
1	7/29/82	5	11/5/81	9	7/29/82	13	7/29/82
2	11/5/81	6	7/29/82	10	7/29/82	14	7/29/82
3	11/5/81	7	7/29/82	11	7/29/82		
4	11/5/81	8	7/29/82	12	7/29/82		

3.0 Scope:

3.1 Purpose:

This procedure provides guidelines for the performance of the duties and responsibilities of the Shift Technical Advisor (STA).

3.2 Discussion:

NUREG 0578 - "TMI Lessons Learned Task Force Status Report and Short Term Recommendations" establishes the requirements for the STA position and for the performance of two functions:

3.2.1 Accident Assessment Function

General Description:

The primary task of the STA is to provide an independent, dedicated concern for the safety of the Turkey Point Plant. The STA is to provide advice to the Nuclear Plant Supervisor (NPS) during off-normal and emergency situations on actions to be taken to terminate or mitigate the consequences of such events.

3.2.2 Operating Experience Assessment Function

General Description:

The STA, along with other Turkey Point Plant staff members and Power Resources staff, will provide the Operating Experience Assessment Function required by the NRC in accordance with Administrative Procedure 0103.15, Operating Experience Feedback. The STAs daily tasks and routine are also discussed in this plant procedure.

3.3 Authority:

3.3.1 NUREG 0578

3.3.2 NUREG 0737

3.4 Definitions:

3.4.1 Shift Technical Advisor - an individual with a bachelor's degree or equivalent in a scientific or engineering discipline who has been trained and qualified in accordance with Administrative Procedure 0307.

3.4.2 Plant Abnormal Occurrence or Event - an unplanned departure from normal plant operating conditions. This may or may not result in a plant trip, but usually will fall into the NRC notification or significant events category.

4.0 Precautions:

None

5.0 Responsibilities:

5.1 The Technical Department Supervisor has overall responsibility for staffing, implementing, coordinating, evaluating, and reviewing the STA program.

5.2 The Technical Staff Training Coordinator is responsible for developing, implementing, and documenting the STA training program in accordance with Administrative Procedure 0307.

5.3 Each STA shall be responsible for performing the accident assessment and operating experience assessment functions described in paragraphs 8.2 and 8.3.

5.4 The STA is responsible to the Technical Department Supervisor. During off-normal reactor plant conditions, the STA shall advise the Nuclear Plant Supervisor (NPS) in the Control Room or Technical Support Center (TSC).

5.5 The on-shift STA group will be composed of personnel assigned to the plant Technical Staff. The Technical Department Supervisor has the option of assigning STA shifts. During the off-STA time period, the STA is responsible for performance of those functions normally accomplished by the Technical Staff as assigned by the Technical Supervisor.

5.6 The STA position is required to be manned during power operation, startups, hot standby, and hot shutdown.

6.0 References:

6.1 NUREG 0578 "TMI Lessons Learned Task Force Status Report and Short-Term Recommendations".

6.2 USNRC letter of September 13, 1979, "Follow-up Actions Resulting from the NRC Staff Reviews Regarding the Three Mile Island Unit 2 Accident"

6.3 USNRC Information Notice 80-06 and Supplement No. 1 "Notification of Significant Events at Operating Power Reactor Facilities"

6.4 10 CFR 20.403 "Notification of Incidents"

6.5 FPL Letter of December 27, 1979, Revised March 7, 1980, from A. D. Schmidt to J. K. Hays, "Duties and Responsibilities of Nuclear Plant Supervisors and Shift Technical Advisors"

6.6 NUREG 0737

7.0 Records and Notifications:

None

8.0 Instructions:

8.1 Qualifications and Training:

8.1.1 The STA shall have a bachelor's degree or equivalent in a scientific or engineering discipline.

8.1.2 The STA shall be trained in:

1. The response and analysis of the plant for transients and accidents.

2. Details of the design, function, arrangement, and operations of plant systems, including the capabilities of instrumentation and controls in the Control Room.

8.1.3 STA initial qualification and re-qualification shall be certified in writing by the Technical Staff Training Coordinator, Technical Department Supervisor and the Plant Manager - Nuclear upon completion of qualifications.

8.2 Accident Assessment Function:

8.2.1 The primary task of the STA is to provide an independent dedicated concern for the safety of the Turkey Point Plant. This is accomplished by providing diagnostic support to Operations personnel during off-normal events and by advising the NPS on actions to terminate or mitigate the consequences of such events.

8.2.2 The role of the STA is to serve in an advisory capacity only, and not to assume any command or control functions.

8.2.3 To accomplish this function, the STA should physically remain within an area which will allow him to be available to the Control Room, preferably immediately, but at the most within ten minutes.

8.2.4 The STA should remain in the Control Room during the course of the accident in order to assess vital core parameters to ensure safety of the reactor. He should not become involved with administrative and phone calling duties.

8.2.5 An off-shift STA will act as liason between the Control Room and TSC when the TSC is activated and manned.

8.2.6 The STA is responsible for assisting the Plant Manager - Nuclear in assuring that notification is made to the NRC Operations Center as soon as possible and in all cases within one hour by telephone of the occurrence of any of the following significant events.

1. Any event requiring initiation of the licensee's emergency plan or any section of the plan.

2. The exceeding of any Technical Specification Safety Limit.

3. Any event that results in the nuclear power plant not being in a controlled or expected condition while operating or shut down.

4. Any act that threatens the safety of the nuclear power plant or site personnel, or the security of special nuclear material, including instances of sabotage or attempted sabotage.

5. Any event requiring initiation of shutdown of the nuclear power plant in accordance with Technical Specification Limiting Conditions for Operation.

6. Personnel error or procedural inadequacy which, during normal operations, anticipated operational occurrences, or accident conditions, prevents or could prevent, by itself, the fulfillment of the safety function of those structures, systems, and components important to safety that are needed to (1) shutdown the reactor safely and maintain it in a safe shutdown condition, or (2) remove residual heat following reactor shutdown, or (3) limit the release of radioactive material to acceptable levels or reduce the potential for such release.

7. Any event resulting in manual or automatice actuation of Engineered Safety Features, including the Reactor Protection System.

8. Any accidental, unplanned, or uncontrolled radioactive release. (Normal or expected releases from maintenance or other operational activities are not included).

9. Any fatality or serious injury occurring on the site and requiring transport to an off-site facility for treatment.

10. Any serious personnel radioactive contamination requiring extensive on-site decontamination or outside assistance.

11. Any event meeting the criteria of 10 CFR Para. 20.403 for notification.

12. Strikes of operating employees or security guards, or honoring of picket lines by these employees.

13. If one or more ENS (red) phone extensions is found to be inoperable, the NRC Operations Center must be notified within one hour by the best available means (an operable ENS extension, commercial telephone, dispatcher phone to Miami, relay to NRC, etc.)

14. The STA may be releived of his accident assessment responsibilities only by a qualified STA.

8.3 Operating Experience Assessment Function:

8.3.1 The plant operating experience assessment function is performed as a joint effort between members of the plant Technical Staff, and the Power Resources Nuclear Staff, and the STA.

8.3.2 The following are guidelines for the STA in the performance of his responsibilities:

NOTE: Except where noted, initialling to indicate review of each of the below documents is not required.

1. Operator Logs - During each shift, the STA should review and be cognizant of equipment out of service, jumpers and disconnected leads, shift turnover sheets, surveillance and special testing progress.

2. Licensee Event Reports (LERs); IE Bulletins, Circulars, and Notices; and pertinent NRC or utility assessments of operating experience. The STA will review the output of the Operational Experience Feedback Program and other information as directed by the Technical Department Supervisor.

3. Jumper and Disconnected Leads - The STA on-shift or the STA Engineer Supervisor shall review each request to jump or disconnect lead or cables and provide recommendation to the Nuclear Plant Supervisor as to how the request affects the equipment and plant parameters.

4. The STA shall monitor the RCS leakage once per shift and ascertain if the source of the RCS leakage is known or assist in the investigation of the RCS leakage source if not known when RCS leakage exceeds 0.5 gpm.

8.4 Operating Experience Assessment Function

8.4.1 Surveillance Testing - The STA should be aware of Operations surveillance testing on his shift and any unsatisfactory results.

8.4.2 Reactor/Turbine Trips and Transients -

1. The STA should immediately assess all reactor trips and transients occurring on his shift with regard to safety. This assessment should include (but not limited to):

   (1) Sequence of Events
   (2) Causes
   (3) Plant Response
   (4) Corrective action taken to ensure plant safety
   (5) Violations of Technical Specifications and/or Safety Limits
   (6) Procedural Inadequacies/Equipment Out of Service

2. The STA should commence preparing a report* on all unusual or abnormal events occurring on his shift as soon as practicable after his accident assessment duties are completed.

   This report should be drafted by the STA or by another member of the Technical Staff within 24 hours of the event.

   The STA should collect copies of all pertinent strip charts, logs, data sheets, etc., from the control room and include them with his report.

   The report should include enough information in the following areas to enable off-site personnel to understand the event and to provide feedback if appropriate. The following criteria should be used when assembling pertinent information about the event:

   a. UNIT STATUS:

      Record the initial conditions. This should include the unit number, date, time when the event began, and the unit power level when the event began. If the unit was not at a steady state power operation, given an explanation of the status.

      Record the effect on the unit. If a power reduction or unit shutdown results, note the method used to achieve this, (manual trip, automatic trip etc.) record the time that power reduction was initiated and the time when hot shutdown or desired reduced power level was reached. During recovery, record the time of startup or load increase. Note the time when full power was reached.

*[illegible]ment 2 provides a form for reporting Plant Abnormal Occurrences.

CAUSE OF EVENT

Name any known component failures and the failure mode, if available.

When available, note possible procedural and personnel errors, design/construction problems and the plant department involved.

METHOD OF DISCOVERY

State how the problem was discovered (routing, testing, non-routine testing, visual observation, etc.) Note which plant department made the discovery (NO's, RCO's, Maintenance, Etc.)

FOR RADIOLOGICAL INCIDENTS

Provide available information concerning any uncontrolled release. Note Health Physics Shift Supervisor on duty, in case additional information is required.

PERSONNEL INVOLVED

Make note of the operations group that was on shift during the event, in case additional information or clarification of events is required.

CORRECTIVE ACTIONS

Describe to the fullest extent possible what corrective actions were taken at the time of the event.

Data from strip charts, logs, data sheets, etc., should be presented in a concise and usable form. This may include replotting data, labeling data, combining data on a single plot, etc. Simplified system drawings should be included if they aid in understanding the event.

3. Radiological Conditions - The STA should be cognizant (on a shift basis) of the radiological conditions existing in the plant, including any gaseous or liquid releases in progress.

4. General Plant Conditions - The STA should be alert to and responsive toward any of the following:

   Plant Efficiency
   Maintenance Items
   Housekeeping
   Procedures
   Quality Assurance

   Problems noticed in any of these areas should be brought to the attention of the appropriate department head and Nuclear Plant Shift Supervisor.

8.5 Shift Turnover and Relief:

8.5.1 The off-going STA should review applicable plant documents and conditions to determine any off-normal system conditions or trends, and shall fill out a turnover checklist (Attachment 1)

8.5.2 The off-going STA shall pass on to the relieving STA the plant status, emphasizing system off-normal conditions and trends, tests in progress, etc.

8.5.3 A tour of the plant should be conducted by the relieving STA as soon as practical after assuming the shift in order to verify correct equipment operating parameters and status.

8.5.4 At the beginning of each work week (Monday morning) all completed shift turnover sheets with the exception of the most recent three (3) days should be forwarded to the STA Engineer Supervisor for review and disposition.

8.5.5 In the event of sickness, personal emergency, or other problem which precludes the STA from assuming or completing his/her scheduled shift, the STA Engineer Supervisor should be informed. The off-going STA should arrange a relief if feasible before leaving. A relief should be arranged to fill the STA position within at least two (2) hours.

ATTACHMENT I
SHIFT TURNOVER SHEET

Date: ____________

Shift: ____________

I. Summary of Shift Events:

II. Components/Systems Under Test

1.
2.
3.

III. Major Equipment Out of Service (If yes, complete Attachment 2 and submit to Plant Licensing Engineer)

	TECH SPEC ITEM	
1.	Yes	No
2.	Yes	No
3.	Yes	No
4.	Yes	No

IV. Potential LER's (Complete Attachment 2 and submit to Plant Licensing Engineer)

1.
2.
3.

V. Planned/Scheduled Evolutions

1.
2.
3.

VI. Additional Information

1.
2.
3.

Off-going STA ____________________

On-coming STA ____________________

Indicates attachment with additional details

ATTACHMENT 2

STA REPORT OF PLANT ABNORMAL OCCURRENCES

Copies To: Operations Supervisor - Nuclear
Operations Superintendent - Nuclear
Maintenance Superintendent - Nuclear
Technical Department Supervisor

Prepared by____________________ Date__________

Reviewed by____________________ Date__________
STA Engineer, Supervisor

Reviewed by____________________ Date__________
Licensing Engineer, Supervisor

Reviewed by____________________ Date__________
Technical Department Supervisor

I. Title:

II. Summary Description of Event(s):

1. Date of Event:__________ Unit:__________

2. Facility Status Prior to the Occurrence: (Circle One)

   c. Routine Startup Operations
   d. Routine Shutdown Operations
   e. Steady State Operations, Power Level:______
   f. Load Changes During Routine Power Operation
   g. Shutdown (Hot or Cold) Except Refueling
   h. Refueling
   x. Other (including special tests, emergency shutdown operations, etc.)
   z. Item Not Applicable

   Time of Event:__________

   Time when power reduction or shutdown began:__________

   During recovery, time when startup or load increase began:__________.

   During recovery, time when full power or desired power was reached:__________

3. Effect on Plant (circle one) — Shutdown Method (circle one)

   a. Unit Trip — a. Normal
   b. Manual Trip
   c. Automatic Trip

ATTACHMENT 2 (cont'd)

b. Forced Power Reduction

c. No Significant Effect

4. System (on which event occurred): ____________________

5. Event Description:

a. Evolutions in progress when the event occurred: ____________________

____

b. Circumstances leading to the event: ____________________

____

c. Significant occurrences as a result of the event: ____________________

____

d. Chain of events: ____________________

____

e. Other unit affected: [ ] Yes [ ] No

f. Were redundant systems available and operable: ____________________

____

____

6. Description of Damage or Loss of Facility: ____________________

____

____

____

7. Preliminary Indication of Principle Cause: (Circle One)

a. Personnel Error, Department: ____________________

b. Procedure Inadequacy, Procedure Number: ____________________

c. Component Failure, Component: ____________________

Component Manufacturer: ____________________

Location of Component: ____________________

____

____

____

d. Other: ____________________

____

____

____

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ATTACHMENT 2 (cont'd)

8. Cause Description and Corrective Action

a. Probably Cause or Failure Mode (if known): ______

b. Nature of Damage (if known): ______

c. Department(s) Performing Corrective Actions: ______

9. Method of Discovery: (Circle One and Describe)

a. Operational Event: ______

b. Routine Test/Inspection: ______

c. Special Test/Inspection: ______

d. External Source: ______

e. Item Not Applicable: ______

Release

Activity Released: (Circle One)

a. Liquid

b. Solid

c. Gas

d. Mixture

e. Item Not Applicable

Location of Release (if known): ______

ty Supervisor: ______

ATTACHMENT 2 (cont'd)

11. Personnel Exposure:

a. Number of Personnel: ____________________

b. Type (if known) (Circle One)

i. Internal Exposure

e. External Exposure

b. Both

z. Item Not Applicable

c. Description: ____________________

HP Duty Supervisor: ____________________

12. Personnel Injuries:

a. Number of personnel involved: ____________________

b. Description: ____________________

13. Immediate Corrective Action Taken (if known): (Circle One)

a. Replace Part(s)

b. Repair Part(s)

c. Replace Total Component

d. Repair Total Component

e. Recalibrate/Adjust

f. Redesign/Modify

g. Change of Procedure

h. Retrain/Reinstruct Personnel

x. Other ____________________

z. No Corrective Action

ATTACHMENT 2 (cont'd)

14. Operations Shift on Duty During Event (number): ______________

The following personnel were directly involved with the event and may be able to supplement the information in this report if clarification is needed:

______________________________   ______________________________

______________________________   ______________________________

______________________________   ______________________________

15. Additional Comments:

____

____

____

____

____

____

____

____

____

____

1.2 POST-TRIP REVIEW - DATA AND INFORMATION CAPABILITY

1. Capability for assessing sequence of events (on-off indications)

Response:

1) Brief description of equipment (e.g. plant computer, dedicated computer, strip chart)

Equipment

The equipment utilized as the primary source of information for assessing the post trip review is the plant computer. This computer is referred to as DDPS or Digital Data Processing System. This equipment consists of a Data General Nova 840 central processor with 32 kilo bytes of internal memory. Mass storage is performed by an ampex megastore with 750 kilo bytes of magnetic core memory. In addition, a Diablo 1 mega byte disc drive provides backup mass storage and program loading capability.

Inputs to the central processor are serviced by four (4) digital input/output controllers, two per nuclear unit and four (4) wide range analog input systems also two per unit. This peripheral equipment was manufactured by Computer Products.

Sequence of Events

Sequence of Events inputs feed the central processor through the digital input/output controllers and output on the control room line printer which is shared between both units. This output is nearly instantaneous, reflecting the real time events as they occur. Two hundred and sixty-five (265) on-off indications can be supported on each nuclear unit by this hardware. Two hundred and thirty-seven (237) are active on Unit 3 and two hundred and thirty-eight (238) are active on Unit 4. PC/Ms and controlled PWOs necessitate frequent additions and reassignments of input channels.

2) Parameters Monitored

The Sequence of Events parameters that are monitored by the plant computer are listed on the attached pages marked, "Digital Channels". All of the digital input channels on the computer are responded to by the control room line printer. In addition to the digital inputs, the computer software also displays events of internal nature. These are listed in the digital channels' pages with an "X" prefix on the channel number.

3) Time Discrimination Between Events

Printed events are time tagged with a time discrimination of .01 seconds. That is, events occurring within the same .01 second will share the same time tag.

4) Format for displaying data and information

In addition to a time tag, the Sequence of Events channel number is displayed. Alarm status, that is, alarm or clear and the channel description (event) are also printed. All of this information is printed on a single line situated in the left half of the printed page to indicate Unit 3 events or situated to the right half of the page for Unit 4.

5) Capability for retention of data and information

The hard copy sequence of events print-out is administratively routed to Document Control for storage following the review of the trip (O.P. 0208.1); there is no software retention of sequence of events.

6) Power source(s) (e.g., Class IE, Non-Class IE, non-interruptable)

The power source for the plant computer supporting both sequence of events and analog variables is a dedicated inverter. This inverter is powered from the DC Bus. It is non-Class IE and its output feeds through a static switch. This switch senses power interruptions and switches input from a second source (Breaker 40534 on the Unit 4 A Bus). This switching may also be performed manually for required maintenance on the inverter or DC breaker. The plant computer's central processor is programmed to enable recovery and resume operation following power interruptions. Static switch operation will only interfere with computer function for a short period of time.

2. Capability for assessing the time history of analog variables needed to determine the cause of unscheduled reactor shutdowns, and the functioning of safety-related equipment.

Response:

1) Brief description of equipment (e.g., plant computer, dedicated computer, strip charts)

Covered in the response to 1.1.

2) Parameters monitored, sampling rate, and basis for selecting parameters and sampling rate

The analog variables which are monitored by the plant computer are inputs to the Wide Range Analog Input Systems (WRAIS). The WRAIS has reference voltage inputs which provide for automatic scale ranging and the reference voltage inputs may be displayed as a quick indication of analog to digital conversion errors. There are two Wide Range Analog Input Systems per unit with each one accepting 128 input channels.

Many other analog parameters are monitored by strip chart recorders. There are to be forty-four (44) recorders providing retention of data and information.

There is an attached list of analog variables which are monitored by the plant computer. This list includes the DDPS channel number, sampling rate, and signal name.

Another list includes the analog variables which are displayed and retained by strip chart recorders.

The basis for selecting parameters monitored on DDPS was the consensus of qualified personnel engaged in the original development of the specifications for the Digital Data Processing System. Along with the monitored parameters determination, the sampling rates were determined. The main considerations being the maximum expected rate of change of the parameter and also data base size limitations.

Assignment of parameters monitored and sampling rates is available for inspection by all qualified plant personnel by the distribution of a reference document "DDPS Commands Summary". Reassignment of parameters monitored and sampling rates has been in support of PC/Ms and controlled PWOs.

3) Duration of time history (minutes before trip and minutes after trip)

A program of the plant computer called "Post Trip Review" maintains a data base for each unit. The Post Trip Review Program along with the data base allows for retention and recall of data and information from five (5) minutes prior to a trip until three (3) minutes following a trip. Information can only be recalled one channel at a time. The format for displaying data is as follows: The output is displayed on the applicable unit's CRT and if a hard copy is desired, a single keyboard key initiates the line printer output. The output is printed on the single line printer in the dual unit control room. The right side of the page is used for Unit 4 and

the left side of the page is used for Unit 3. Individual channel requests are separated by a row of asterisks (*). A header line follows in the form:

UNIT # PTR CHANNEL ####

The time tagged data lines follow the header line. The time tag displays the number of seconds before the trip time, preceeded by a minus (-) sign or no sign with seconds following a trip. The time tag refers to the first sampled record following the time tag. Three more records follow on the same line, displayed with their analog readings in E format. In this manner, every fourth reading has a time tag and the total number of readings depends upon the sampling rate.

4) Format for displaying data including scale (readability) of time histories

Covered in the response to 2.3.

5) Capability for retention of data, information and physical evidence (both hardware and software)

The DDPS Post Trip Review data base is retained in magnetic core memory. This mass storage memory device retains its contents in the event of a total interruption of power. The data base itself is only lost when it is released by a keyboard command by reactor control operators prior to unit startup. The capability exists for copying the post trip review data base on a removable magnetic disc. This, however, has not been found to be necessary and would require removing the computer from service. Storage of pertinent post trip review printouts is covered in Operating Procedure 0208.1. It is routed through the Operations Supervisor and stored in Document Control.

6) Power source(s) (e.g., Class IE, non-Class IE, non-interruptable)

Covered in the response to 1.6

3. Other data and information provided to assess the cause of unscheduled reactor shutdowns.

<u>Response</u>:

Other available data present for the operating staff to evaluate the causes of a reactor trip include trend recorders, bistable status lights and the plant annunciation system. The following recorders provide real time trends for evaluation:

```
Reactor Power:          NR-*-45 Nuclear Power Recorder
                        NR-*-46 Overpower Recorder
                        NR-*-47 Overpower Recorder

Reactor Coolant System:   TR-*-420 RCP Status Recorder
                          FR-*-154B RCP Seal Leak-off
                                    Recorder - Hi Range
                          TR-*-423 Loop A Temp Recorder
                          TR-*-424 Loop B Temp Recorder
                          TR-*-425 Loop C Temp Recorder
                          PR-*-444 Pressurizer Pressure Recorder
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Secondary System:	TR-*-454 A S/G Level Flow and Feed
	TR-*-464 B S/G Level Flow and Feed
	TR-*-474 C S/G Level Flow and Feed
	TR-*-444 Turbine Valve Position
	TR-*-441 Turbine Vibration Recorder

These recorders allow hard copy of the transient. Real time instrument channel indications are located on the console, vertical panels and on the Nuclear Instrument Cabinets. These provide the operator with the status of the total plant.

4. Schedule for any planned changes to existing data and information capability.

Response:

In approximately one year, a new central processor will replace the DDPS Data General Nova. At about the same time, an additional processing system will carry out the Safety Assessment System (SAS) functions. Along with these modifications, the existing sequence of events channels on DDPS will remain essentially unchanged but many more sequence of events parameters will be added with the SAS. The channels remaining in DDPS will become much more accessible with the use of four (4) Post Trip Review software files. In this implementation, the operations "Release Post Trip Review" command will no longer be necessary. The software retention of data will be much improved.

The exact implementation of retrieval of Post Trip Review data from the Safety Assessment System has not been finalized at the time. Data and information will be stored on magnetic tape but format of output, that is, printed on line printer or played back in a simulated real time display has not been decided.

SEQUENCE-OF-EVENTS FUNCTION

The first function of the DDPS is to sense, and report on the line printer, any of a defined set of discrete "events". An event may be the result of a contact change-of-state, an operator command, or detection of an unusual condition by the program. Each event is identified by a unique event number, Associated with each recorded event is a current state, which is either "alarm" or "clear".

The event numbers which may be reported are:

001-256	One of a possible 256 contacts has changed state from open to closed, or vice-versa.
257-264	One of a possible 8 AC voltages has dropped below 90 V r.m.s.
265	The crystal oscillator which provides a time base for computer operations has failed, and a back-up oscillator has taken over.
C001-C265	One of the above inputs has been either inhibited from further event reporting, or re-enabled for reporting after having been previously inhibited.
A001-A256	One of the possible 256 analog inputs has been either inhibited from further alarm checking, or re-enabled for checking after having been previously inhibited.
X001-X006	Some unusual condition has been detected by the program.

An English-language text string is associated with each event number. Each event is reported as a single printed line containing the event number, the associated text label, and the work "alarm" or "clear", indicating the current status of the input.

SOE SCAN

Once every 10 milliseconds all the digital inputs corresponding to events 1-256 above are examined for a change of state. For each input which has changed state since the previous scan, an event is reported. The assignment of an "alarm" or "clear" indication to an "open" or "closed" contact state is made independently for each discrete input. The collection of these associations forms a part of the program's permanent data base.

The DDPS I/O LIST, lists for each event number the corresponding discriptive label and the physical input state interpreted by the program as an "alarm" condition.

If a given input changes state too frequently in a short period of time, it will automatically be inhibited from further reporting. In such a case, a "C"-type event will be recorded, with state = "clear". To re-enable the input for event detection, an ENABLE CCI command must be issued from the CRT terminal.

ENABLE/INHIBIT CCI

Each time a CCI (Contact Closure Input) is inhibited from, or re-enabled for, SOE reporting, an event is generated. The Report consists of:

- The letter "C" followed by the input number (1-256) of the affected digital input.
- The identifying text belonging to the addressed SOE number.
- The word "Alarm" if the input is enabled for reporting, or "Clear" if reporting is inhibited.

ENABLE/INHIBIT A/D

Each time an A/D (Analog-to-Digital Converter) input is inhibited from, or enabled for, limit checking an event is generated, The report consists of:

- The letter "A" followed by the channel number (1-256) of the affected A/D input.
- The identifying text belonging to the addressed A/D input.
- The word "Alarm" if the input is enabled for limit checking, or "clear" if checking is inhibited.

MISCELLANEOUS EVENTS

A few special conditions will cause "events" to be recorded. Each such event is assigned a number beginning with the letter "X" and a self-explanatory text label. Although the word "alarm" or "clear" is printed, this state indication has no significance for these events.

The following is a complete catalog of "X" type events:

X001 EVENT BUFFER FULL. One or more detected events have gone unreported. (Because too many events occurred within a very short time span.)

X002 PTR FILE FROZEN. The Post-Trip Review file (an 8 minute history of A/D data) has been locked, and may be examined with with the Post-Trip Review program.

X003 PTR FILE RELEASED. The Post-Trip Review file has been freed for data gathering. In this state, the file contents are not subject to examination.

X004 A/D FAILURE, CH 1-128. The A/D converter containing the low-numbered channels has failed to deliver data when requested.

X005 A/D FAILURE, CH 129-256. The A/D converter containing the high-numbered channels has malfunctioned.

X006 DATE-TIME CHANGE. This event is generated whenever the program's time base is altered (by operator command).

ANALOG DATA INPUT

The DDPS can handle up to 256 channels (per unit) of analog inputs, coming into two multiplexed analog-to-digital converters. Each input is identified by a channel number (1-256), and has associated with it an English-language label and several items of control information.

For purposes of discussion, it is convenient to recognize the following input groups:

- Plant-monitor channels. (Approximately 130 channesl.) Each input in this group is sampled either once a second, every ten seconds, or once a minute. Every value input is written to the Post-Trip Review file on disc, unless that file happens to be frozen.

- Reactor Core Analysis channels. (Approximately 70 channels.) These inputs are applicable only to flux mapping, and are sampled only upon specific request.

- A/D Calibration channels. (16 channels) There are reference voltage inputs used for self-calibration of the A/D converters. They are sampled once every 20 minutes.

- Unimplemented and Spare channels. (Approximately 40 channels.) These channels are never scanned.

DATA RECORDING

A circulating disc file, the Post Trip Review File, is used to maintain an eight-minute history of all readings taken of A/D channels in the "plant monitor" category. Each reading is written to the disc as it is received, along with the channel number and a time tag. Data in this file may be selectively examined by means of the Post-Trip Review Program described in Section 3.4.11.

The PTR file is locked -- i.e., no more data are written to it-- three minutes after closure of the "Freeze PTR" CCI is detected. At the moment the contact closure is detected, a blinking message "PTR FREEZE PENDING" is flashed on the CRT screen. When the file is actually locked three minutes later, an event is logged on the line printer.

The file is unlocked--i.e., data output resumes -- when either:

1) The command "RLS" is entered on the CRT terminal, or

2) The "Release PTR" CCI changes from open to closed.

Whenever the file is released after having been frozen, it is "wiped clean" I.e., data gathered prior to the freeze are completely lost, even if the file is immediately re-frozen. This is not true, however, if a "RLS" command is issued after the CCI trip has occurred but before it has been frozen. In this case, the impending lock-up is simply cancelled, and the file retains a full eight minutes worth of data.

TURKEY POINT PLANT

DDPS SCAN RATES

TABLE 1

A/D CH#	SIGNAL	SCAN RATE
1	0 MV REF	20 MIN
2	2 MV REF	20 MIN
3	8 MV REF	20 MIN
4	32 MV REF	20 MIN
5	128 MV REF	20 MIN
6	512 MV REF	20 MIN
7	2048 MV REF	20 MIN
8	8192 MV REF	20 MIN
9	CHARGING FLOW F-122	1 MIN
10	CHARGING PRESS P-121	1 MIN
11	VCT LEVEL L-115	1 MIN
12	VCT TEMP T-116	1 MIN
13	VTC PRESS P-117	1 MIN
14	RC FLOW LP A CH1	10 SEC
15	RC FLOW LP A CH2	10 SEC
16	RC FLOW LP A CH3	10 SEC
17	T AVE LP A CONTROL	10 SEC
18	T AVE LP A PROT	10 SEC
19	OVER TEMP SP LP A	10 SEC
20	OVERPOWER SP LP A	10 SEC
21	DELTA T LP A CONTROL	10 SEC
22	DELTA T LP A PROT	10 SEC
23	RC FLOW LP B CH1	10 SEC
24	RC FLOW LP B CH2	10 SEC
25	RC FLOW LP B CH3	10 SEC
26	T AVE LP B CONTROL	10 SEC
27	T AVE LP B PROT	10 SEC
28	OVERTEMP SP LP B	10 SEC
29	OVERPOWER SP LP B	10 SEC
30	DELTA T LP B CONTROL	10 SEC

A/D CH #	SIGNAL	SCAN RATE
31	DELTA T LP B PROT	10 SEC.
32	RC FLOW LP C CH1	10 SEC
33	RC FLOW LP C CH2	10 SEC
34	RC FLOW LP C CH3	10 SEC
35	T AVE LP C CONTROL	10 SEC
36	T AVE LP C PROT	10 SEC
37	OVERTEMP SP LP C	10 SEC
38	OVERPOWER SP LP C	10 SEC
39	DELTA T LP C CONTROL	10 SEC
40	DELTA T LP C PROT	10 SEC
41	PZR LEVEL CHAN 1	10 SEC
42	PZR LEVEL CHAN 2	10 SEC
43	PZR LEVEL CHAN 3	10 SEC
44	PZR LEVEL WIDE RANGE	10 SEC
45	PZR PRESS CH1	1 SEC
46	PZR PRESS CH2	1 SEC
47	PZR PRESS CH3	1 SEC
48	PZR PRESS LOOP 444	10 SEC
49	PZR PRESS LOOP 445	10 SEC
50	TURB FIRST STS PR CH3	1 SEC
51	TURB FIRST STG PR CH4	1 SEC
52	PZR LIQUID TEMP	1 MIN
53	PZR STEAM TEMP	1 MIN
54	PZR SURGE LINE TEMP	1 MIN
55	PZR SPRAY TEMP LP B	1 MIN
56	PZR SPRAY TEMP LP C	1 MIN
57	ROD BANK A POSITION	10 SEC
58	ROD BANK B POSITION	10 SEC
59	RAD BANK C POSITION	10 SEC
60	ROD BANK D POSITION	10 SEC
61	RCS WIDE RANGE PRESS	10 SEC
62	AUCTIONEERED T AVE	10 SEC
63	AUCTIONEERED DELTA T	10 SEC
64	T REF	10 SEC
65	FEED FLOW LOOP A CH3	10 SEC
66	FEED FLOR LOOP A CH4	10 SEC

A/D CH #	SIGNAL	SCAN RATE
67	SREAM FLOW LOOP A CH3	10 SEC
68	STEAM FLOW LOOP A CH4	10 SEC
69	STEAM GEN A LEVEL CH1	10 SEC
70	STEAM GEN A LEVEL CH2	10 SEC
71	STEAM GEN A LEVEL CH3	10 SEC
72	STEAM GEN A LEVEL WR	10 SEC
73	STEAM GEN A PRESS CH2	10 SEC
74	STEAM GEN A PRESS CH3	10 SEC
75	STEAM GEN A PRESS CH4	10 SEC
76	STEAM HDR PRESS CH2	10 SEC
77	STEAM HDR PRESS CH3	10 SEC
78	STEAM HDR PRESS CH4	10 SEC
79	STEAM GEN BLOWDOWN FLOW	1 MIN
80	FEED FLOW LOOP B CH3	10 SEC
81	FEED FLOW LOOP B CH4	10 SEC
82	STEAM FLOW LOOP B CH3	10 SEC
83	STEAM FLOW LOOP B CH4	10 SEC
84	STEAM GEN B LEVEL CH1	10 SEC
85	STEAM FEN B LEVEL CH2	10 SEC
86	STEAM GEN B LEVEL CH3	10 SEC
87	STEAM GEN B LEVEL WR	10 SEC
88	STEAM GEM B PRESS CH2	10 SEC
89	STEAM GEN B PRESS CH 3	10 SEC
90	STEAM GEN B PRESS CH4	10 SEC
91	FEED FLOW LOOP C CH3	10 SEC
92	FEED FLOW LOOP CH CH4	10 SEC
93	STEAM FLOW LOOP C CH3	10 SEC
94	STEAM FLOW LOOP C CH4	10 SEC
95	STEAM FEN C LEVEL CH1	10 SEC
96	STEAM FEN C LEVEL CH2	10 SEC
97	STEAM GEN C LEVEL CH3	10 SEC
98	STEAM FEN CL PRESS CH2	10 SEC

A/D CH#	SIGNAL	SCAN RATE
99	STEAM GEN C PRESS CH3	10 SEC
100	STEAM GEN C PRESS CH4	10 SEC
101	STEAM GEN C LEVEL WR	10 SEC
102	CONTAINMENT SUMP LEVEL	1 MIN
103	(SPARE)	SPECIAL
104	(SPARE)	SPECIAL
105	ABS CONDENSER PRESS	1 MIN
106	CONTAINMENT PRESS WR	1 MIN
107	CONTAINMENT PRESS WR	1 MIN
108	N41 DET A CURRENT	1 SEC
109	N41 DET B CURRENT	1 SEC
110	N42 DET A CURRENT	1 SEC
111	N42 DET B CURRENT	1 SEC
112	N41 % POWER	1 SEC
113	N42 % POWER	1 SEC
114	N43 % POWER	1 SEC
115	N44 % POWER	1 SEC
116	N31 LEVEL	1 MIN
117	N32 LEVEL	1 MIN
118	N35 LEVEL	1 MIN
119	N36 LEVEL	1 MIN
120	FE 476 DIFF PRESSURE	1 MIN
121	FEEDWATER PRESSURE	1 MIN
122	STEAM PRESSURE LOOP A	1 MIN
123	FE 486 DIFF PRESSURE	1 MIN
124	STEAM PRESSURE LOOP B	1 MIN
125	FE 496 DIFF PRESSURE	1 MIN
126	STEAM PRESSURE LOOP C	1 MIN
127	TURBINE CONTROL OIL PRESS	1 SEC
128	TOTAL POWER - NUCLEAR	1 MIN
219	0 MV REF	20 MIN
130	2 MV REF	20 MIN
131	8 MV REF	20 MIN
132	32 MV REF	20 MIN

A/D CH #	SIGNAL	SCAN RATE
133	128 MV REF	20 MIN
134	512 MV REF	20 MIN
135	2048 MV REF	20 MIN
136	8192 MV REF	20 MIN
137	GEN MEGAWATT, REC	1 SEC
138	GEN MEGAWATT IND.	1 SEC
139	T COLD LP A	1 MIN
140	T HOT LP A	1 MIN
141	T COLD LP B	1 MIN
142	T HOT LP B	1 MIN
143	T COLD LP C	1 MIN
144	T HOT LP C	1 MIN
145	FEEDWATER TEMP 1	1 MIN
146	FEEDWATER TEMP 2	1 MIN
147	GEN MEGAVARS IND	1 SEC
148	N43 DET A CURRENT	1 SEC
149	N43 DET B CURRENT	1 SEC
150	N44 DET A CURRENT	1 SEC
151	N44 DET B CURRENT	1 SEC
152	D.D.P.S. TC REF JCT	1 MIN

DIGITAL CHANNELS

DDPS CH #	SIGNAL NAME
001	CONTAINMENT SPRAYS
002	HI CONT. PR 10% CH 1
003	HI CONT. PR 10% CH 2
004	HI CONT. PR 10% CH 3
005	HI CONT. PR 50% CH 1
006	HI CONT. PR 50% CH 2
007	HI CONT. PR 50% CH 3
008	MAIN STM ISOL VLV A CL
009	MAIN STM ISOL VLV B CL
010	MAIN STM ISOL VLV C CL
011	RPI ROD BOTTOM
012	LO T AVE TO SI LOOP A
013	LO T AVE TO SI LOOP B
014	LO T AVE TO SI LOOP C
015	RC LOW FLOW LOOP A CH 1
016	RC LOW FLOW LOOP A CH 2
017	RC LOW FLOW LOOP A CH 3
018	METAL IMPACT MONITOR (UNIT 4 ONLY)
019	OVERPOWER DELTA T LP A
020	OVERTEMP DELTA T LP A
021	RC PUMP A OFF
022	RC LOW FLOW LOOP B CH 1
023	RC LOW FLOW LOOP B CH 2
024	RC LOW FLOW LOOP B CH 3
025	OVERPOWER DELTA T LP B
026	OVERTEMP DELTA T LP B

DIGITAL CHANNELS

DDPS CH #	SIGNAL NAME
027	RC PUMP B OFF
028	RC LOW FLOW LOOP C CH 1
029	RC LOW FLOW LOOP C CH 2
030	RC LOW FLOW LOOP C CH 3
031	OVERPOWER DELTA T LP C
032	OVERTEMP DELTA T LP C
033	RC PUMP C OFF
034	PZR LO PRESS TRIP CH 1
035	PZR LO PRESS TRIP CH 2
036	PZR LO PRESS TRIP CH 3
037	PZR HI PRESS TRIP CH 1
038	PZR HI PRESS TRIP CH 2
039	PZR HI PRESS TRIP CH 3
040	PZR HI LVL TRIP CH 1
041	PZR HI LVL TRIP CH 2
042	PZR HI LVL TRIP CH 3
043	PZR LO PRESS TO SI CH 1
044	PZR LO PRESS TO SI CH 2
045	PZR LO PRESS TO SI CH 3
046	PZR LO LVL TO SI CH 1
047	PZR LO LVL TO SI CH 2
048	PZR LO LVL TO SI CH 3
049	PZR LO LVL AND PRESS SI
050	SI UNBLOCKED

DIGITAL CHANNELS

DDPS CH #	SIGNAL NAME
051	STEAMLINE SI UNBLOCKED
052	AUTO SI
053	MANUAL SI
054	PZR PR > 2000 CH 1
055	PZR PR > 2000 CH 2
056	PZR PR > 2000 CH 3
057	SI BLOCK MAN PB
058	SI UNBLOCK MAN PB
059	SI MANUAL PB
060	RT 1 AND 2 RELAYS
061	RT 3 AND 4 RELAYS
062	RT 5 AND 6 RELAYS
063	RT 7 AND 8 RELAYS
064	RT 9 AND 10 RELAYS
065	N 31 HI LEVEL TRIP
066	N 32 HI LEVEL TRIP
067	SOURCE RANGE NIS BLOCKED
068	N 35 HI LEVEL TRIP
069	N 36 HI LEVEL TRIP
070	INT RANGE NIS BLOCKED
071	N 41 LOW RANGE TRIP
072	N 42 LOW RANGE TRIP
073	N 43 LOW RANGE TRIP
074	N 44 LOW RANGE TRIP
075	NIS POWER RANGE BLOCKED

DIGITAL CHANNELS

DDPS CH #	SIGNAL NAME
076	N 41 HI RANGE TRIP
077	N 42 HI RANGE TRIP
078	N 43 HI RANGE TRIP
079	N 44 HI RANGE TRIP
080	POWER ABOVE P 10
081	START UVTD RELAY TEST
082	CONT ISOL PHASE A
083	CONT ISOL PHASE B
084	GENERATOR LEADS BACK-UP RELAY TRIP
085	GENERATOR BACK-UP DISTANCE RELAY TRIP
086	PC446 A1X ABOVE 10%
087	PC447 E1X ABOVE 10%
088	REVERSE POWER RELAY TRIP
089	GENERATOR OVEREXCITATION RELAY TRIP
090	UVTD-RX TRIP LOW 4 KV
091	UV-4KV BUS A CH 1
092	UV-4KV BUS A CH 2
093	UV-4KV BUS B CH 1
094	UV-4KV BUS B CH 2
095	NIS ROD DROP
096	TURB LOAD LIMIT RUNBACK
097	DELTA T TURBINE RUNBACK
098	MANUAL PB RX TRIP CONS
099	MANUAL PB RX TRIP VP B
100	UNDERFREQ TO RCP A
101	UNDERFREQ TO RCP B AND C

DIGITAL CHANNELS

DDPS CH #	SIGNAL NAME
102	WARN-GEN STATOR DELTA TEMP HI
103	STEAM GEN A LO LEVEL CH 1
104	STEAM GEN A LO LEVEL CH 2
105	STEAM GEN A STM > FW CH 4
106	STEAM GEN A STM > FW CH 4
107	STM GEN A HI LVL TRIP CH 1
108	STM GEN A HI LVL TRIP CH 2
109	STM GEN A HI LVL TRIP CH 3
110	STEAM GEN A FW ISOLATION
111	STM GEN A LOLO LEVEL CH 1
112	STM GEN A LOLO LEVEL CH 2
113	STM GEN A LOLO LEVEL CH 3
114	STEAM LINE A HI DIFF CH 2
115	STEAM LINE A HI DIFF CH 3
116	STEAM LINE A HI DIFF CH 4
117	STEAM LINE A HI DIFF SI
118	STEAM LINE A HI FLOW CH 3
119	STEAM LINE A HI FLOW CH 4
120	STEAM LINE HI FLOW 2/3
121	STEAM LINE A LO PR TO SI
122	LO T AVE OR STM PR TO SI
123	MAIN STM ISOL LOOP A
124	MAIN STM ISOL MAN PB A
125	STM GEN B LO LEVEL CH 1
126	STM GEN B LO LEVEL CH 2

DDPS CH #	SIGNAL NAME
127	STEAM GEN B STM > FW CH 3
128	STEAM GEN B STM > FW CH 4
129	STM GEN B HI LVL TRIP CH 1
130	STM GEN B HI LVL TRIP CH 2
131	STM GEN B HI LVL TRIP CH 3
132	STM GEN B FW ISOLATION
133	STM GEN B LOLO LVL CH 1
134	STM GEN B LOLO LVL CH 2
135	STM GEN B LOLO LVL CH 3
136	STEAM LINE B HI DIFF CH 2
137	STEAM LINE B HI DIFF CH 3
138	STEAM LINE B HI DIFF CH 4
139	STM LINE B HI DIFF SI
140	STM LINE B HI FLOW CH 3
141	STM LINE B HI FLOW CH 4
142	STM LINE B LO PR TO SI
143	MAIN STM ISOL LOOP B
144	MAIN STM ISOL MAN PB B
145	STEAM GEN C LO LEVEL CH 1
146	STEAM GEN C LO LEVEL CH 2
147	STEAM GEN C STM > FW CH 3
148	STEAM GEN C STM > FW CH 4
149	STM GEN C HI LVL TRIP CH 1
150	STM GEN C HI LVL TRIP CH 2
151	STM GEN HI LVL TRIP CH 3

DDPS CH #	SIGNAL NAME
152	STEAM GEN C FW ISOLATION
153	STEAM GEN C LOLO LVL CH 1
154	STEAM GEN C LOLO LVL CH 2
155	STEAM GEN C LOLO LVL CH 3
156	STEAM LINE C HI DIFF CH 2
157	STEAM LINE C HI DIFF CH 3
158	STEAM LINE C HI DIFF CH 4
159	STEAM LINE C HI DIFF SI
160	STEAM LINE C HI FLOW CH 3
161	STEAM LINE C HI FLOW CH 4
162	STEAM LINE C LO PR TO SI
163	MAIN STM ISOL LOOP C
164	MAIN STM ISOL MAN PB C
165	AUTO STOP OIL < 45 PSI
166	UNDERFREQ LO RELAY OPER
167	TURB AUTO STOP OIL CH 1
168	TURB AUTO STOP OIL CH 2
169	TURB AUTO STOP OIL CH 3
170	TURB AUTO STOP OIL 2/3
171	LEFT STOP VLV CLOSED
172	RIGHT STOP VLV CLOSED
173	TURB STOP VLVS CLOSED
174	INT RANGE > P 6
175	POWER RANGE 1 > P 8

DIGITAL CHANNELS

DDPS CH #	SIGNAL NAME
176	POWER RANGE 1 > P 10
177	INT RANGE 2 > P 6
178	POWER RANGE 2 > P 8
179	POWER RANGE 2 > P 10
180	POWER RANGE 3 > P 8
181	POWER RANGE 3 > P 10
182	POWER RANGE 4 > P 8
183	POWER RANGE 4 > P 10
184	POWER ABOVE P 8
185	RX TRIP BYPASS BKR A OPEN (FUTURE)
186	POWER ABOVE P 7
187	TIME CHECK
188	TURB TRIP MAN PB CONS
189	RX TRIP BKR A OPEN
190	RX TRIP BKR B OPEN
191	THRUST BRG TRIP ARMED
192	RX TRIP BYPASS BKR B OPEN (FUTURE)
193	GEN BKR A OPEN
194	GEN BKR B OPEN
195	OVERPOWER DELTA T TRIP
196	OVERTEMP DELTA T TRIP
197	PZR HI-LO PR TRIP 2/3
198	PZR HI LVL TRIP 2/3
199	NIS POWER RANGE LO TRIP
200	TURB OVERSPEED OIL TRIP

DIGITAL CHANNELS

DDPS CH #	SIGNAL NAME
201	THRUST BRG TURB TRIP
202	LOW VACUUM TURB TRIP
203	LOW BRG OIL TURB TRIP
204	AUX XFMR DIFF'L TRIP
205	A SG FEED PUMP OFF
206	EXHST HD HI TEMP TRIP
207	TURB ANTI MOTORING TRIP
208	AUX XFMR BRKR A OPEN
209	AUX XFMR BRKR B OPEN
210	REHEAT OR INTERCEPT CL
211	GEN NEG SEQUENT TRIP
212	GEN LOSS OF FIELD TRIP
213	OVEREXCITATION TRIP
214	GEN DIFFERENTIAL TRIP
215	GEN GROUND TRIP
216	MAIN XFMR DIFFERENTIAL
217	MAIN XFMR GROUND
218	SU XFMR DIFFERENTIAL
219	RC LO FLOW LOOP A TRIP
220	RC LO FLOW LOOP B TRIP
221	RC LO FLOW LOOP C TRIP
222	NIS POWER RANGE HI TRIP
223	STM GEN A LO FLO AND LVL
224	STM GEN A LOLO LVL TRIP

DIGITAL CHANNELS

DDPS CH #	SIGNAL NAME
225	STM GEN B LO FLO AND LVL
226	STM GEN B LOLO LEVEL TRIP
227	STM GEN C LO FLO AND LVL
228	STM GEN C LOLO LVL TRIP
229	AUX TRANSFORMER GROUND (FUTURE)
230	GENERATOR LOCKOUT RELAY
231	TURBINE OVERSPEED PROT
232	B SG FEED PUMP OFF
233	MAIN TRANS FAULT PRESS (FUTURE)
234	AUX TRANS FAULT PRESSURE (FUTURE)
235	S.U. BKR-A OR EMERG. S.U. BKR (FUTURE)
236	STARTUP TRANS BKR B (FUTURE)
237	S.U. TRANS FAULT PRESS (FUTURE)
238	S.U. TRANS GROUND (FUTURE)
239	SPARE
240	SPARE
241	C BUSS TRANS SUDDEN FAULT PRESSURE (FUTUF
242	C BUSS TRANS LOCKOUT (FUTURE)
243	C BUSS TRANS HI-TEMP (FUTURE)
244	4C BUSS LOCKOUT (FUTURE)
245	BKR 4AC01 (FUTURE)
246	BKR 4AC16 (FUTURE)
247	SPARE
248	SPARE
249	SPARE
250	SPARE

DIGITAL CHANNELS

DDPS CH #	SIGNAL NAME
251	SPARE
252	SPARE
253	SPARE
254	SPARE
255	SPARE
256	SPARE
257	INST AC BUS 3P08 DOWN
258	INVERTER P08 SPARE DOWN
259	INST AC BUS 3P07 DOWN
260	INVERTER P07 SPARE DOWN
261	INST AC BUS 3P06 DOWN
262	INVERTER P06 SPARE DOWN
263	INST AC BUS 3P09 DOWN
264	INVERTER P09 SPARE DOWN
265	A CLOCK FAILURE
X001	EVENT BUFFER FULL
X002	PTR FILE FROZEN
X003	PTR FILE RELEASED
X004	A/D FAILURE CH 1-128
X005	A/D FAILURE CH 129-256
X006	DATE TIME CHANGE
X007	MIDS INACTIVE
X008	SPARE
X009	SPARE
X010	SPARE

DIGITAL CHANNELS

DDPS CH #	SIGNAL NAME
X011	SPARE
X011	SPARE
X013	SPARE
X014	SPARE
X015	SPARE
X016	SPARE
X017	SPARE

SIGNAL NAME

T RECORDERS

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Feed Flow Level, Wide Range Levels

.ls, Bearings and Stator Windings)

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ıture Setpoint

and Cooler Inlet and Outlet Temperatures
earing Vibration
Expansion
and Thrust Bearing Positions
ser Temperatures
ıd Cooling Water

ncy Containment Cooler
mperatures

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POST TRIP REVIEW

ANALOG VARIABLES

DDPS CH #	SCAN RATE	SIGNAL NAME
009	1 min	CHARGING FLOW F-122
010	1 min	CHARGING PRESSURE P-121
011	1 min	VCT LEVEL L-115
012	1 min	VCT TEMP T-116
013	1 min	VCT PRESS P-117
014	10 sec	RC FLOW LP A CH 1
015	10 sec	RC FLOW LP A CH 2
016	10 sec	RC FLOW LP A CH 3
017	10 sec	T AVE LP A CONTROL
018	10 sec	T AVE LP A PROT
019	10 sec	OVER TEMP SP LP A
020	10 sec	OVERPOWER SP LP A
021	10 sec	DELTA T LP A CONTROL
022	10 sec	DELTA T LP A PROT
023	10 sec	RC FLOW LP B CH 1
024	10 sec	RC FLOW LP B CH 2
025	10 sec	RC FLOW LP B CH 3
026	10 sec	T AVE LP B CONTROL
027	10 sec	T AVE LP B PROT
028	10 sec	OVERTEMP SP LP B
029	10 sec	OVERPOWER SP LP B

POST TRIP REVIEW

ANALOG VARIABLES

DDPS CH #	SCAN RATE	SIGNAL NAME
030	10 sec	DELTA T LP B CONTROL
031	10 sec	DELTA T LP B PROT
032	10 sec	RC FLOW LP C CH 1
033	10 sec	RC FLOW LP C CH 2
034	10 sec	RC FLOW LP C CH 3
035	10 sec	T AVE LP C CONTROL
036	10 sec	T AVE LP C PROT
037	10 sec	OVERTEMP SP LP C
038	10 sec	OVERPOWER SP LP C
039	10 sec	DELTA T LP C CONTROL
040	10 sec	DELTA T LP C PROT
041	10 sec	PZR LEVEL CHAN 1
042	10 sec	PZR LEVEL CHAN 2
043	10 sec	PZR LEVEL CHAN 3
044	10 sec	PZR LEVEL WIDE RANGE
045	1 sec	PZR PRESS CH 1
046	1 sec	PZR PRESS CH 2
047	1 sec	PZR PRESS CH 3
048	10 sec	PZR PRESS LOOP 444
049	10 sec	PZR PRESS LOOP 445
050	1 sec	TURB FIRST STG PR CH 3
051	1 sec	TURB FIRST STG PR CH 4

POST TRIP REVIEW

ANALOG VARIABLES

DDPS CH #	SCAN RATE	SIGNAL NAME
052	1 min	PZR LIQUID TEMP
053	1 min	PZR STEAM TEMP
054	1 min	PZR SURGE LINE TEMP
055	1 min	PZR SPRAY TEMP LP B
056	1 min	PZR SPRAY TEMP LP C
057	10 sec	ROD BANK A POSITION
058	10 sec	ROD BANK B POSITION
059	10 sec	ROD BANK C POSITION
060	10 sec	ROD BANK D POSITION
061	10 sec	RCS NARROW RANGE PRESS
062	10 sec	AUCTIONEERED T AVE
063	10 sec	AUCTIONEERED DELTA T
064	10 sec	T REF
065	10 sec	FEED FLOW LOOP A CH 3
066	10 sec	FEED FLOW LOOP A CH 4
067	10 sec	STEAM FLOW LOOP A CH 3
068	10 sec	STEAM FLOW LOOP A CH 4
069	10 sec	STEAM GEN A LEVEL CH 1
070	10 sec	STEAM GEN A LEVEL CH 2
071	10 sec	STEAM GEN A LEVEL CH 3
072	10 sec	STEAM GEN A LEVEL WR
073	10 sec	STEAM GEN A PRESS CH 2
074	10 sec	STEAM GEN A PRESS CH 3
075	10 sec	STEAM GEN A PRESS CH 4

POST TRIP REVIEW

ANALOG VARIABLES

DDPS CH #	SCAN RATE	SIGNAL NAME
076	10 sec	STEAM HDR PRESS CH 2
077	10 sec	STEAM HDR PRESS CH 3
078	10 sec	STEAM HDR PRESS CH 4
079	1 min	STEAM GEN BLOWDOWN FLOW
080	10 sec	FEED FLOW LOOP B CH 3
081	10 sec	FEED FLOW LOOP B CH 4
082	10 sec	STEAM FLOW LOOP B CH 3
083	10 sec	STEAM FLOW LOOP B CH 4
084	10 sec	STEAM GEN B LEVEL CH 1
085	10 sec	STEAM GEN B LEVEL CH 2
086	10 sec	STEAM GEN B LEVEL CH 3
087	10 sec	STEAM GEN B LEVEL WR
088	10 sec	STEAM GEN B PRESS CH 2
089	10 sec	STEAM GEN B PRESS CH 3
090	10 sec	STEAM GEN B PRESS CH 4
091	10 sec	FEED FLOW LOOP C CH 3
092	10 sec	FEED FLOW LOOP C CH 4
093	10 sec	STEAM FLOW LOOP C CH 3
094	10 sec	STEAM FLOW LOOP C CH 4

POST TRIP REVIEW

ANALOG VARIABLES

DDPS CH #	SCAN RATE	SIGNAL NAME
095	10 sec	STEAM GEN C LEVEL CH 1
096	10 sec	STEAM GEN C LEVEL CH 2
097	10 sec	STEAM GEN C LEVEL CH 3
098	10 sec	STEAM GEN C PRESS CH 2
099	10 sec	STEAM GEN C PRESS CH 3
100	10 sec	STEAM GEN C PRESS CH 4
101	10 sec	STEAM GEN C LEVEL WIDE RANGE
102	1 min	CONTAINMENT SUMP LEVEL
103	10 sec	SPARE
104	1 min	INTAKE COOLING WATER TEMP
105	1 min	ABS CONDENSER PRESS
106	1 min	O.O.S.
107	1 min	O.O.S.
108	1 sec	N41 DET A CURRENT
109	1 sec	N41 DET B CURRENT
110	1 sec	N42 DET A CURRENT
111	1 sec	N42 DET B CURRENT
112	1 sec	N41 % POWER
113	1 sec	N42 % POWER
114	1 sec	N43 % POWER
115	1 sec	N44 % POWER

POST TRIP REVIEW

ANALOG VARIABLES

DDPS CH #	SCAN RATE	SIGNAL NAME
116	1 min	N31 LEVEL
117	1 min	N32 LEVEL
118	1 min	N35 LEVEL
119	1 min	N36 LEVEL
120	1 min	FE 476 DIFF PRESSURE
121	1 min	FEEDWATER PRESSURE
122	1 min	STEAM PRESSURE LOOP A
123	1 min	FE 486 DIFF PRESSURE
124	1 min	STEAM PRESSURE LOOP B
125	1 min	FE 496 DIFF PRESSURE
126	1 min	STEAM PRESSURE LOOP C
127	1 sec	TURBINE CONTROL OIL PRESSURE
128	1 min	TOTAL POWER-NUCLEAR
137	1 sec	GEN MEGAWATT, REC
138	1 sec	GEN MEGAWATT, IND
139	1 min	T COLD LP A
140	1 min	T HOT LP A
141	1 min	T COLD LP B
142	1 min	T HOT LP B
143	1 min	T COLD LP C
144	1 min	T HOT LP C
145	1 min	FEEDWATER TEMP 1
146	1 min	FEEDWATER TEMP 2

Screening of Orders Entered

All orders entered are screened by the Westinghouse District Sales Offices, Distributors, Engineering Service Offices, or Repair Plants to determine whether or not safety-related nuclear equipment, parts or services are involved. Our nuclear utilities have been informed that they must ensure that procurement documents issued by them or by their sub-contractors or agents to Westinghouse clearly indicate if the equipment, parts or services are for a nuclear plant and if so whether or not they are safety-related if they expect Westinghouse to ensure that applicable NRC regulatory requirements are met.

If an order is received for equipment, parts or services for a nuclear plant without any indication of whether or not it is safety-related, then the customer will be asked the following question:

> Does the customer have any responsibilty under 10 CFR Part 21 for reporting defects, as defined in that regulation, which exist in the equipment, parts or services at the time of delivery or which may subsequently occur?

If the answer to this question is yes, the order is entered through an approved order entry channel.

If the answer to this question is no, normal commercial practices apply.

Approved Order Entry Channels

All orders for equipment parts identified by the purchaser or through screening for safety-related applications are processed through the Water Reactor Divisions (WRD). All safety-related service orders will be placed with divisions who are on an approved supplier list for the particular services requested. If no approved supplier is listed for requested services, the order will be processed through WRD.

Qualification and Training

NRC regulations place the responsibility for ensuring that regulatory requirements are met in the procurement of safety-related equipment, parts or services on the utility. The utility is required to provide any special requirements to those providing safety-related equipment, parts or services. Westinghouse will ask for any special requirements or instructions from the customer when safety-related equipment, parts or services are involved and to ensure that the work done conforms to those special requirements or instructions.

Informing Customers - Instruction Books

Instruction books are provided by divisions supplying equipment or parts. Such instruction books include information necessary for proper and safe installation, operation, maintenance and repair in ordinary commercial applications of such equipment and parts.

Westinghouse divisions providing services are to obtain copies of appropriate instruction books for the equipment and parts which they service from the divisions which originally supplied the equipment or parts. WRD is to include in its equipment specifications and purchase orders for equipment and parts appropriate requirements for copies of instruction books for its customers and for its own use in developing recommendations to its customers for installation, operation, maintenance and repair of the equipment and parts in nuclear applications.

Instruction books for safety-related equipment are subject to the quality assurance requirements of 10 CFR Part 50 Appendix B and are to be included on the quality assurance release which must accompany any delivery of safety-related equipment or parts.

Substantive errors discovered in instruction books for safety-related equipment after delivery to the customer are subject to the reporting requirements of 10 CFR Part 21. Correction of errors may be accomplished by WRD Technical Bulletins in lieu of revising the instruction books or, in cases in which WRD is not involved, by supplementary information provided directly to its customers by the involved division.

Informing Customers - Technical Bulletins

Information supplementing or revising the instruction books or other similar materials necessary for proper and safe installation, operation, maintenance or repair of WRD-supplied equipment or parts in nuclear applications is provided by WRD in the form of Technical Bulletins. Preparation of such Technical Bulletins has been centralized within WRD and is subject to the same design control process which applies to the equipment, parts and services to which they relate. Technical Bulletins which are safety-related are so identified.

All Technical Bulletins will be transmitted by WRD to every Westinghouse NSSS customer, domestic and international, and such other WRD customers as are affected. Responsibility for this distribution has been centralized for customers with operating plants and for customers for plants not yet in operation in order to ensure that all Technical Bulletins are promptly distributed. Customers have been requested to provide the necessary distribution lists for their organizations.

All distributions of safety-related Technical Bulletins are now accompanied by a return receipt. The return receipts are pre-addressed to a central point in WRD for recording all Technical Bulletins transmitted and their status. Technical Bulletins for which receipt is not acknowledged with a reasonable time are retransmitted.

A list of current Technical Bulletins and Data Letters will be prepared and transmitted to all customers periodically but not less frequently than once per year. This transmittal is to be in the form of a Technical Bulletin and is to be transmitted in the same manner as any other Technical Bulletin. Appendix A is Bulletin NSD-TB-83-05 which gives the index of currently valid Technical Bulletins.

FPL Control of Vendor Maintenance Work

FPL Quality Procedures 4.1 (Control of Requisitions and the Issuance of Purchase Orders to Spare Parts, Replacement Items and Services) and 4.4 Review of Procurement Documents for Items and Services Other Than Spare Parts) provide a system to assure that the appropriate technical and quality requirements are placed upon suppliers who provide material, equipment and services for operating nuclear power plants. Safety-related Purchase Orders for services are issued only to suppliers whose Quality Assurance Program and implementing procedures have been evaluated and approved by the FPL Quality Assurance Department. Vendor maintenance work on reactor trip systems components must comply with applicable plant procedures. Vendor implementing procedures prepared in accordance with the vendors QA program, which must be approved by FPL QA, may also be used. Vendor procedures used to perform on-site work must be approved by the Plant Nuclear Safety Committee.

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

Position

Licensees and applicants shall confirm that all components whose functioning is required to trip the reactor are identified as safety-related on documents, procedures, and information handling systems used in the plant to control safety-related activities, including maintenance, work orders, and parts replacement. In addition, for these components, licensees and applicants shall establish, implement and maintain a continuing program to ensure that vendor information is complete, current and controlled throughout the life of the plant, and appropriately referenced or incorporated in plant instructions and procedures. Vendors of these components should be contacted and an interface established. Where vendors cannot be identified, have gone out of business, or will not supply the information, the licensee or applicant shall assure that sufficient attention is paid to equipment maintenance, replacement, and repair, to compensate for the lack of vendor backup, to assure reactor trip system reliability. The vendor interface program shall include periodic communication with vendors to assure that all applicable information has been received. The program should use a system of positive feedback with vendors for mailings containing technical information. This could be accomplished by licensee acknowledgement for receipt of technical mailings. The program shall also define the interface and division of responsibilities among the licensees and the nuclear and nonnuclear divisions of their vendors that provide service on reactor trip system components to assure that requisite control of and applicable instructions for maintenance work are provided.

Response

FPL has determined that at Turkey Point Units 3 and 4 components whose functioning is required to trip the reactor are included in systems which are treated as safety related for plant activities, such as maintenance, work orders and parts replacement. Future changes or modifications of these systems will be reviewed by FPL engineering to ensure that the correct safety classification is made. A description of the information handling systems for component classification is provided in our response to action 2.2.1.

In addition, a continuing program exists to receive and review vendor information. It is the policy of Westinghouse Electric Corporation to be a reliable supplier of equipment, parts, and services needed by FPL for use in nuclear power plants. Orders for such equipment, parts and services for safety-related applications are to be given the special attention which is required by applicable regulatory requirements as well as commercial practices. The following information summarizes pertinent parts of this Westinghouse policy on the W/customer interface:

Definition of Safety-Related Equipment, Parts and Services

Equipment, parts and services are safety-related if the utility customer for whom the equipment, parts or services are intended indicates that he has responsibility under Nuclear Regulatory Commission Regulation 10 CFR Part 21 for reporting any defects, as defined in that regulation, in the equipment, parts or services as delivered to him or which may subsequently occur.

2.2 EQUIPMENT CLASSIFICATION AND VENDOR INTERFACE (PROGRAMS FOR ALL SAFETY-RELATED COMPONENTS)

Position

Licensees and applicants shall submit, for staff review, a description of their programs for safety-related* equipment classification and vendor interface as described below:

1. For equipment classification, licensees and applicants shall describe their program for ensuring that all components of safety-related systems necessary for accomplishing required safety functions are identified as safety-related on documents, procedures, and information handling systems used in the plant to control safety-related activities, including maintenance, work orders and replacement parts. This description shall include:

   1. The criteria for identifying components as safety-related within systems currently classified as safety-related. This shall not be interpreted to require changes in safety classification at the systems level.

   2. A description of the information handling system used to identify safety-related components (e.g., computerized equipment list) and the methods used for its development and validation.

   3. A description of the process by which station personnel use this information handling system to determine that an activity is safety-related and what procedures for maintenance, surveillance, parts replacement and other activities defined in the introduction to 10 CFR 50, Appendix B, apply to safety-related components.

   4. A description of the management controls utilized to verify that the procedures for preparation, validation and routine utilization of the information handling system have been followed.

   5. A demonstration that appropriate design verification and qualification testing is specified for procurement of safety-related components. The specifications shall include qualification testing for expected safety service conditions and provide support for the licensees' receipt of testing documentation to support the limits of life recommended by the supplier.

   6. Licensees and applicants need only to submit for staff review the equipment classification program for safety-related components. Although not required to be submitted for staff review, your equipment classification program should also include the broader class of structures, systems, and components important to safety required by GDC-1 (defined in 10 CFR Part 50, Appendix A, "General Design Criteria, Introduction").

2. For vendor interface, licensees and applicants shall establish, implement and maintain a continuing program to ensure that vendor information for safety-related components is complete, current and controlled throughout the life of their plants, and appropriately referenced or incorporated in plant instructions and procedures. Vendors of safety-related equipment should be contacted and an interface established. Where vendors cannot be identified, have gone out of busines, or will not supply information, the licensee or applicant shall assure that sufficient attention is paid to equipment maintenance, replacement, and repair, to compensate for the lack of vendor backup, to assure reliability commensurate with its safety function (GDC-1). The program shall be closely coupled with action 2.2.1 above (equipment qualification). The program shall include periodic communication with vendors to assure that all applicable information has been received. The program should use a system of positive feedback with vendors for mailings containing technical information. This could be accomplished by licensee acknowledgment for receipt of technical mailings. It shall also define the interface and division of responsibilities among the licensee and the nuclear and nonnuclear divisions of their vendors that provide service on safety-related equipment to assure that requisite control of and applicable instructions for maintenance work on safety-related equipment are provided.

Response

1.1 and 1.2 Quality Procedure 2.7 (Identification of Safety-Related and Nuclear safety-Related QA Required Structures, Systems Components and Services) describes the FPL system for identifying nuclear safety-related structures, systems, components. A copy of this QP is attached. The Turkey Point Q-List is a system level document which has been generated and maintained in accordance with this QP. A program to increase the specificity of the Q-List is now underway. Our initial contractor has prepared a detailed safety-related component list for the CVCS system, and other systems are expected to be completed by a vendor yet to be selected. A draft of the general guidelines to be used for evaluating the quality classification of systems and structures for Turkey Point Units 3 & 4 is attached as an example of effort that is currently underway. The final guidelines are planned to be provided with the contract to the vendor when selected.

The final product of this effort is expected to be a component level safety-related listing and drawings to designate Q-boundaries.

1.3 Administrative Procedure 0103.4 (In Plant Equipment Clearance Orders) requires that the equipment involved in a plant clearance be checked to determine if safety-related. Administrative Procedure 0190.15 (Plant Projects-Approval, Implementation and Regulatory Requirements) provides controls for safety-related Plant Changes/Modifications. Administrative Procedure 0190.19 (Control of Maintenance on Nuclear Safety-Related and Fire Protection Systems) provides guidelines for procedural requirements on maintenance of nuclear safety-related and fire protection systems, components and equipment. The safety-related determination for the above procedures are made by reference to the Turkey Point Q-List. In addition the FPL Quality Procedures as contained in the FPL Quality Assurance Manual are followed to ensure that procedures for maintenance, surveillance, parts replacement and other activities defined in the introduction to 10 CFR 50, Appendix B apply to safety-related components. The most recent revision to the FPL Topical Quality Assurance Report was found to be acceptable by the NRC on September 7, 1983.

1.4 - Like all aspects of the QA Program, procedures for preparation, validation and routine utilization of the safety-related identification documents are the subject of routine Quality Assurance audits.

1.5 Quality Procedure 4.1 provides a system to assure that the appropriate technical and quality requirements are placed upon suppliers who provide material, equipment and services for FPL nuclear plants. A copy of this Quality Procedure is attached.

1.6 No response is required.

2.0 FPL is a member of the Nuclear Utility Task Action Committee (NUTAC) on Generic Letter 83-28, Section 2.2.2 formed on September 1, 1983, for the specific purpose of defining an appropriate vendor interface program. At present, we intend to incorporate the results of the NUTAC. Our schedule for submission of our program description was provided in our letter L-83-480, dated September 7, 1983. This report is scheduled for submittal by February 29, 1984.

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GENERAL GUIDELINES FOR Q-BOUNDARIES:

The following general guidelines shall be used in evaluating the Quality Classification of Systems and Structures for Turkey Point Units 3 & 4.

1. Nuclear Safety Related

   a) Nuclear Safety Related is defined as those structures, systems or components which are necessary to assure:

      i) the integrity of the reactor coolant pressure boundary;

      ii) the capability to shut down the reactor and maintain it in a safe shutdown condition; or

      iii) 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 Part 100.

   b) Reactor coolant pressure boundary means all those pressure-containing components such as pressure vessels, piping, pumps and valves which are:

      i) part of the reactor coolant system or,

      ii) connected to the reactor coolant system up to and including any and all of the following:

         1. The outermost containment isolation valve in system piping which penetrates primary reactor containment,

         2. The second of two valves normally closed during normal reactor operation in system piping which does not penetrate primary reactor containment,

         3. The reactor coolant system safety and relief valves.

   c) A loss of capability to achieve a required safety function cited in Paragraph 1(a) means the loss of a safety function to the extent that there is a major reduction in the degree of protection provided to public health and safety.

      A major reduction in the degree of protection means:

      i) loss of an item, in conjunction with a single failure, results in the inability to perform the nuclear safety-related function utilizing equipment normally designed for the purpose without the availability of offsite power, or,

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ii) a partial or total loss of function of a radioactive confinement system that compromises its ability to contain radioactive materials such that offsite exposures approach 25 rem whole body or 300 rem thyroid, or

iii) a partial or total loss of function of a radioactive confinement system that results in an operator exposure approaching 25 rem whole body or 300 rem thyroid during the performance of an operator action normally required to achieve a safety function cited in Paragraph 1(a) assuming an exposure period of 20 minutes.

d) Safe shutdown means the reactor is in the hot shutdown condition, i.e., it is subcritical by an amount greater than or equal to a margin specified in the Technical Specifications and $T_{avg.}$ is above 540°F.

e) A single failure means an occurrence which results in the loss of capability of a component to perform its intended function. Multiple failures resulting from a single occurrence are considered to be a single failure.

All items classified nuclear safety related shall be considered Q and shall meet the quality assurance requirements of the FPL QA Program.

2. Safety Related Design Feature

Safety Related Design Feature is defined as those structures, systems or components which are not nuclear safety related and are in one or more of the following categories:

a) Equipment, components and structures designed to meet seismic requirements or whose failure could:

i) damage safety-related equipment such that the equipment would be prevented from performing its safety function, or

ii) resulted in releases exceeding the exposure guidelines of Part C) below.

b) Fire protection equipment

i) required to protect nuclear safety related equipment, or

ii) whose failure could result in water damage to nuclear safety related equipment which could prevent the equipment from performing its safety function.

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c) A partial or total loss of function of a radioactive confinement system that results in an accidental, unplanned, or uncontrolled release of radioactivity exceeding an:

i) exposure to an individual in an unrestricted area in any period of one calendar year of 0.5 rem whole body, or

ii) exposure to an individual in a restricted area of 5 rem while performing normal operator actions required to achieve and maintain safe shutdown conditions.

d) Equipment whose failure under normal operating conditions or an anticipated transient, results in

i) exceeding a safety limit specified in the Technical Specifications, or

ii) initiation of an FSAR Design Basis Accident, or

iii) results in the reactor coolant system not being in a controlled or design condition while operating or shutdown.

e) Equipment, components, or structures required to be operable by the Technical Specifications

f) Instrumentation that is essential to monitor emergency reactor shutdown, containment isolation, reactor core cooling, and containment and reactor heat removal, or otherwise are essential in preventing release of radioactive material to the environment which would exceed the guidelines of Part c) above.

Items classified as safety related design feature will be considered "Q" but need not meet the full quality assurance requirements of 10 CFR 50, Appendix B. Applicable quality assurance requirements for items important to safety shall be specified in the FPL QA Program.

3. Functional and Seismic Classification:

For Class 1 systems and structures (seismic Category I) the entire functional boundary and support system shall be considered "Q". The Q-level for the systems and structures shall be based on the guidelines of Items 1 and 2 above.

For determining Q-level for boundary areas and supports, whenever a nuclear safety related system has both Class 1 and Class 3 portions, the "Q" functional boundary shall in most cases be taken to the first normally closed valve or valve capable of automatic closure. The pipe support system shall be considered nuclear safety related to the first anchor downstream of the boundary valve for stress analyzed piping. The piping downstream of the boundary valve to the seismic anchor shall be considered important to safety since any physical modifications to this piping could affect the seismic analysis.

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For 2" and smaller piping that is not stress analyzed, the "Q" functional boundary ends at the isolation valve. The 2" and smaller piping beyond the "Q" boundary must be supported per the small pipe installation manual M-18.

For systems evaluated for high energy line break (Florida Power & Light's response to AEC dated February 26, 1973), where any support system or whip restraint whose failure could damage safety-related equipment, the support or whip restraint shall be considered important to safety.

4. Containment Penetrations:

Mechanical and electrical containment penetrations for any system, whether essential or non-essential, shall be considered safety-related since they are required to keep accident doses less than the guidelines of Item 1.a)iii). FSAR Section 6.6 and Table 6.6-1 shall be the basis for determining containment mechanical penetration boundaries. The pipe support system shall follow the same criteria as listed in Paragraph 3.

5. Instrumentation:

For Instrumentation within the "Q" boundaries of a system, any portion of the instrument which forms a part of the system boundary shall be considered "Q" and shall meet the same "Q" requirements as the portion of the system in which it is located. However, the associated circuitry need not necessarily be "Q". The associated circuitry shall be "Q" only if it performs a nuclear safety related function or is important to safety per Paragraphs 1 and 2. If the instrument output function is only for normal operation and has no safety significance, then the electrical circuitry shall be considered non-Q.

An instrument that is required to perform an active function (initiation of safeguard equipment) is considered nuclear safety related. Instrumentation used to monitor plant conditions and which serve as a basis for operator action will be considered important to safety.

A review will be performed to determine which instruments are required by the operator to achieve or maintain a safe shutdown.

6. Power Actuated Valves:

For power operated valves within "Q" boundaries, the valve, operator, solenoid valve and associated circuitry shall be considered "Q" and shall meet the same "Q" requirements as the portion of the system in which it is located.

Turkey Point's Instrument Air System is non-safety related and not important to safety, therefore, the air supply line is non-Q. Air actuators shall be considered the same level of "Q" as the portion of the system in which they are located since they are required to ensure that the valve fails in the safe position.

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There are a few exceptions to the above criteria and they are as follows:

A. Air Reservoirs - Some air operated valves are required to operate a minimum number of times after an accident, and in these cases, they are provided with air reservoirs. These reservoirs and associated tubing, controls, valves, etc., shall be considered the same "Q" level as the valve.

B. Nitrogen Back-up Supply - Some air operated valves and instrumentation are required to operate on an extended basis after an accident and in these cases a nitrogen bottle back-up supply is provided. The Nitrogen Back-up Supply System shall be considered the same "Q" level as the component it serves.

7. Other Devices (Heat Exchangers, Pumps, Fans, MOVS, etc.):

All components such as heat exchangers, pumps, and other vessels that are within the "Q" functional boundaries and which are required for the functions listed in Items 1 and 2 are considered "Q" and shall meet the same "Q" requirements as the portion of the system in which they are located.

8. Electrical Power Supplies:

Power supplies to any nuclear safety related equipment or equipment important to safety which are required for the functions listed in Items 1 and 2 shall be considered "Q" and shall meet the same "Q" requirements as the portion of the system to which they supply power.

9. Structures:

The containment, auxiliary building, control building, diesel generator building, radwaste facility building, intake structure, portions of the turbine building, and spent fuel pool building support equipment that is safety-related or important to safety. These structures shall meet the same "Q" requirements as the systems they support..

Any structure that could interact with nuclear safety-related equipment shall be considered important to safety.

10. Piping and Valve Classification:

The design codes for piping and valves and how they were applied at Turkey Point requires some clarification. At the time Turkey Point as designed, ASME III Code for piping was not in use. Therefore, ASME I and ANSI B31.1 were used in the design with additional commitments such as seismic design and non-destructive testing (NDT) as indicated in FSAR Appendix 5A and Table 6.2-3. The ASME I and ANSI B31.1 Codes are not, at present, applicable for piping used in safety systems of nuclear power plants. Specification of ANSI B31.1, at present, is not

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sufficient for safety-related systems, since the required documentation for seismic and non-destructive testing is beyond the scope of the Code. For most of the retrofit work which Bechtel has designed under Job Number 5177, the term "upgraded B31.1" has been applied to safety related piping and valves, in order to extend the scope of ANSI B31.1 to include the additional FSAR commitments. In essence, "upgraded B31.1" generally follows the requirements of the applicable class of pipe contained in ASME III and in some cases, the actual equipment purchased in ASME III since vendors do not recognize the meaning of "upgraded B31.1". However, the PC/M makes no commitment to ASME III. The equipment is installed per Upgraded B31.1.

3.1 POST-MAINTENANCE TESTING (REACTOR TRIP SYSTEM COMPONENTS)

1. Licensees and applicants shall submit the results of their review of test and maintenance procedures and Technical Specifications to assure that post-maintenance operability testing of safety-related components in the reactor trip system is required to be conducted and that the testing demonstrates that the equipment is capable of performing its safety functions before being returned to service.

2. Licensees and applicants shall submit the results of their check of vendor and engineering recommendations to ensure that any appropriate test guidance is included in the test and maintenance procedures or the Technical Specifications, where required.

3. Licensees and applicants shall identify, if applicable, any post-maintenance test requirements in existing Technical Specifications which can be demonstrated to degrade rather than enhance safety. Appropriate changes to these test requirements, with supporting justification, shall be submitted for staff approval.

Responses:

Post maintenance testing of the reactor trip system components is governed by MP 0707.10 and AP 190.19. MP 0707.10 covers the maintenance and lubrication of the reactor trip breakers. It has been verified to be in accordance with all manufacturer's recommendations concerning lubrication and testing of the trip breakers and their shunt and undervoltage trip devices. It is performed at 6 month intervals with a check of the manual pushbutton trips conducted each refueling. QC Department surveillance checks have been instituted to ensure that maintenance is performed at the correct intervals. AP 190.19 covers the performance of maintenance and operability testing after maintenance. The reactor trip system including the reactor trip breakers is classified as nuclear safety related under the Turkey Point Q-List. Any maintenance performed on these systems requires that the appropriate Maintenance Superintendent specify retest requirements and that these retest requirements ensure that the component is operable upon completion of maintenance. A review of Technical Specifications did not identify any post maintenance test requirements that would degrade rather than enhance safety.

3.2 POST-MAINTENANCE TESTING (ALL OTHER SAFETY RELATED COMPONENTS)

1. Licensees and applicants shall submit a report documenting the extending of test and maintenance procedures and Technical Specifications review to assure that post-maintenance operability testing of all safety-related equipment is required to be conducted and that the testing demonstrates that the equipment is capable of performing its safety functions before being returned to service.

Response:

Testing of safety related equipment to ascertain its proper post maintenance functioning is already required by Section 3.2 of Administrative Procedure 0190.19, "Control of Maintenance on Nuclear Safety Related and Fire Protection Systems". Additionally, Sections 3.4.3 and 3.4.4 of Administrative Procedure 0103.4, "In-Plant Clearance Orders" requires verification of operability, and surveillance testing of all Technical Specification systems or components prior to return to service after a clearance has been issued.

Representatives of each Turkey Point Maintenance Department were contacted to discuss how the above referenced requirements are being met.

A. The Mechanical Maintenance Department utilizes Administrative Procedure 0190.28, "Mechanical Test Control (Post Maintenance)" to select and document the tests which are conducted on specific safety related equipment following specific maintenance work.

B. The Electrical Department has incorporated the post maintenance test requirements and documentations into specific maintenance procedures such as MP 0729, "Safety Related Motor Operated Valves (MOV) Motor Maintenance". In cases where the maintenance work on safety related electrical equipment is not covered by a specific procedure the post maintenance test requirements are included in the "Work Description" block of the PWO and documented in the "Journeyman's Work Report" on the back of the canary copy of the PWO.

C. The I&C Department performs the necessary calibration checks on safety related equipment after repair or replacement and documents this work on the calibration forms and PWO.

At the completion of maintenance work on safety related equipment, the PWO and supporting documentation is reviewed by a Q.C. inspector IAW AP 0190.19 and these records are retained in accordance with AP 0190.14, "Document Control and Quality Assurance Records".

In order to provide the documentation requested by the NRC and to further strengthen the Administrative control on post maintenance testing of safety related equipment, a change was made to AP 190.19 and was submitted for review.

2. Licensees and applicants shall submit the results of their check of vendor and engineering recommendations to ensure that any appropriate test guidance is included in the test and maintenance procedures or the Technical Specifications were required.

Response

Vendor recommendations for NSSS equipment are reviewed and acted on by means of Westinghouse Technical Bulletins which are entered into the FPL Operating Experience Feedback Program for tracking and implementation.

Information supplementing or revising the instruction books or other similar materials necessary for proper and safe installation, operation, maintenance or repair of WRD-supplied equipment or parts in nuclear applications is provided by WRD in the form of Technical Bulletins. Preparation of such Technical Bulletins has been centralized within WRD and is subject to the same design control process which applies to the equipment, parts and services to which they relate. Technical Bulletins which are safety-related are so identified.

All Technical Bulletins will be transmitted by WRD to every Westinghouse NSSS customer, domestic and international, and such other WRD customers as are affected. Responsibility for this distribution has been centralized for customers with operating plants and for customers for plants not yet in operation in order to ensure that all Technical Bulletins are promptly distributed. Customers have been requested to provide the necessary distribution lists for their organizations.

All distributions of safety-related Technical Bulletins are now accompanied by a return receipt. The return receipts are pre-addressed to a central point in WRD for recording all Technical Bulletins transmitted and their status. Technical Bulletins for which receipt is not acknowledged with a reasonable time are retransmitted.

A list of current Technical Bulletins and Data Letters will be prepared and transmitted to all customers periodically but not less frequently than once per year. This transmittal is to be in the form of a Technical Bulletin and is to be transmitted in the same manner as any other Technical Bulletin.

In addition, as participants in the INPO SEE-IN Program, significant events occurring throughout the nuclear industry and important vendor information items are entered into the FPL Operating Experience Feedback Program. This program provides additional assurance that test and maintenance items which have caused problems at other plants will be reviewed for applicability to Turkey Point Units 3 and 4 and dispositioned.

3. Licensees and applicants shall identify, if applicable, any post-maintenance test requirements in existing Technical Specifications which are perceived to degrade rather than enhance safety. Appropriate changes to these test requirements, with supporting justification, shall be submitted for staff approval.

Response:

There were no post maintenance test requirements identified by the Maintenance Department representatives contacted that were perceived to degrade rather than enhance safety.

There were, however, 3 periodic surveillance requirements that could be perceived as degrading safety. They are:

A. The Containment Spray Pump periodic test per OP 4004.1 requires shutting the manual discharge valve of the pump being tested. This effectively takes the pump out of service for the duration of the test.

B. As part of the test of the auxiliary feedwater pumps during periods of single unit operation, the 2 pumps not under test are manually tripped effectively removing them from service while the third pump is running for test.

C. The monthly battery charge imposes a higher than normal voltage on the DC buss and results in increased failure rates for the normally energized NBFD relays in the reactor protection system. Additionally, the Tech. Specs. requirement for a monthly battery charge conflicts with the manufacturer's recommendation to charge only when needed, as indicated by voltage and gravity readings.

In cases A and B above, an operator is available throughout the test to return the pumps to service if needed. A Tech. Spec. change request was submitted in 1977 to test these pumps in accordance with Section XI "In Service Testing" of ASME Code. When approved, this change will require pump testings every quarter versus the current monthly frequency. This will reduce the frequency at which these systems are put into this degraded mode.

4.1 REACTOR TRIP SYSTEM RELIABILITY (VENDOR-RELATED MODIFICATIONS)

Position

All vendor-recommended reactor trip breaker modifications shall be reviewed to verify that either: (1) each modification has, in fact, been implemented; or (2) a written evaluation of the technical reasons for not implementing a modification exists.

For example, the modifications recommended by Westinghouse in NCD-Elect-18 for the DB-50 breakers and a March 31, 1983, letter for the DS-416 breakers shall be implemented or a justification for not implementing shall be made available. Modifications not previously made shall be incorporated or a written evaluation shall be provided.

Response

Westinghouse records indicate that the modifications recommended in NCD-Elec-18 for DB50 switchgear have been implemented for the Turkey Point Units 3 and 4 reactor trip breakers. The Westinghouse Owners Group has recommended a method to visually inspect for reconfirmation of implementation. This inspection was conducted for the Unit 3 trip and bypass breakers and the Unit 4 bypass breakers and confirmed the presence of post 1972 units (modified units).

The two other UVTA modifications identified by Westinghouse in the March 31, 1983 letter and in NSD-TB-75-2 of February 20, 1955 do not apply.

4.2 REACTOR TRIP SYSTEM RELIABILITY (PREVENTATIVE MAINTENANCE AND SURVEILLANCE PROGRAM FOR REACTOR TRIP BREAKERS)

1. A planned program of periodic maintenance, including lubrication, housekeeping, and other items recommended by the equipment supplier.

2. Trending of parameters affecting operation and measured during testing to forecast degradation of operability.

3. Life testing of the breakers (including the trip attachments) on an acceptable sample size.

4. Periodic replacement of breakers or components consistent with demonstrated life cycles.

Responses:

This item is accomplished under MP 0707.10, OP 1004.2, and ONOP 0208.1. MP 0707.10 covers cleaning, lubrication and testing of the breakers. Every 6 months, each circuit breaker is cleaned, lubricated as required, checked for alignment, checked for proper operation of shunt and undervoltage trip devices, and tested on the bench and in-place. Every refueling overhaul operability of the shunt trip is checked via the manual trip pushbuttons.

No specific trending of reactor trip breaker opening times is performed. However, OP 1004.2 performed monthly and ONOP 0208.1 (Reactor Shutdown Resulting From a Trip) require that breaker opening times of greater than 120 msec be reported to the Assistant Superintendant-Electrical Maintenance in order to schedule preventive action. This opening time is 70% of the maximum permissible and will ensure maintenance is conducted prior to opening time specifications being exceeded.

Life cycle testing of the shunt trip attachment and the undervoltage trip attachment of the reactor trip switchgear is being conducted by Westinghouse for the Westinghouse Owners Group. This program is aimed toward establishing the service life of these devices, and substantiating periodic test requirements with proper maintenance. The results of this program will be factored into maintenance, replacement and qualification programs. The test program is scheduled for completion in the second quarter of 1984.

4.3 REACTOR TRIP SYSTEM RELIABILITY (AUTOMATIC ACTUATION OF SHUNT TRIP ATTACHMENT FOR WESTINGHOUSE AND B&W PLANTS)

Position

Westinghouse and B&W reactors shall be modified by providing automatic reactor trip system actuation of the breaker shunt trip attachments. The shunt trip attachment shall be considered safety related (Class IE).

Response

A detailed generic design package for incorporation of an automatic shunt trip feature into various Westinghouse Reactor Protection Systems has been developed under WOG sponsorship. The complete generic design package of the automatic shunt trip modification was submitted to NRC on June 14, 1983, J. J. Sheppard, Chairman of WOG by letter OG-101.

FPL is in the process of requesting that our Architect Engineer begin preliminary design work for the automatic shunt trip modification for Turkey Point Units 3 & 4. The generic design provided by the WOG and plant specific recommendations will be considered. A proposed design will be submitted to NRC review by May 15, 1983. As part of the submittal, a response to the thirteen NRC concerns in the SER to the WOG design will be provided. A schedule for implementation of the design change will also be provided at that time.

.4 REACTOR TRIP SYSTEM RELIABILITY (IMPROVEMENTS IN MAINTENANCE AND TEST PROCEDURES FOR B&W PLANTS)

This item is not applicable to Turkey Point Units 3 & 4.

4.5 REACTOR TRIP SYSTEM RELIABILITY (SYSTEM FUNCTIONAL TESTING)

1. The diverse trip features to be tested include the breaker undervoltage and shunt trip features on Westinghouse, B&W and GE plants; the circuitry used for power interruption with the silicon controlled rectifiers on B&W plants; and the scram pilot valve and backup scram valves (including all initiating circuitry) on GE plants.

2. Plants not currently designed to permit periodic on-line testing shall justify not making modifications to permit such testing. Alternatives to on-line testing proposed by licensees will be considered where special circumstances exist and where the objective of high reliability can be met in another way.

3. Existing intervals for on-line functional testing required by Technical Specifications shall be reviewed to determine that the intervals are consistent with achieving high reactor trip system availability when accounting for considerations such as:

    1. uncertainties in component failure rates
    2. uncertainty in common mode failure rates
    3. reduced redundancy during testing
    4. operator errors during testing
    5. component "wear-out" caused by the testing

Licensees currently not performing periodic on-line testing shall determine appropriate test intervals as described above. Changes to existing required intervals for on-line testing as well as the intervals to be determined by licensees currently not performing on-line testing shall be justified by information on the sensitivity of reactor trip system availability to parameters such as the test intervals, component failure rates, and common mode failure rates.

Responses

Turkey Point currently has a complete on-line system test procedure which is performed monthly. This test checks the reactor trip breakers (via the undervoltage trip device), the reactor trip relays and their associated logic relays, and a calibration check of the protective system inputs. The current shunt trip device and undervoltage device are functionally bench tested every 6 months. The current shunt trip device is operationally checked at each refueling outage. There is currently no provision for on-line testing of the shunt trip devices on the reactor trip breakers. It is only operated using the manual trip pushbuttons, all automatic trips utilize the undervoltage device. The shunt device is bench tested every 6 months and checked via the manual pushbutton each refueling. Provisions for on-line testing of the shunt device will be considered more formally when the PC/M for inclusion of the shunt device in the automatic trip system is designed.

As noted above in Section 4.3, the WOG generic design package for the automatic shunt trip includes an installation for on-line surveillance testing of the UVTA and automatic shunt trip that provided independent verification of each attachment.

The Westinghouse Owners Group in January 1983, submitted WCAP-10271 to the NRC for review. WCAP-10271, "Evaluation of Surveillance Frequencies and out of Service Times for the Reactor Protection Instrumentation System" documents and evaluation of the impact on RPS unavailability of current and extended surveillance intervals.

The WCAP considers common mode failure, operator error, reduced redundancy during testing and equipment bypass. WCAP-10271 also considers correlative effects on plant peration and safety including the manpower expenditure associated with surveillance, the number of inadvertent trips which occur during testing and the distraction from plant monitoring on the part of the control room operator and shift supervisor associated with testing. Supplement 1 to WCAP-10271 which was submitted to the NRC on October 4, 1983 is an extension of the evaluation and provides a discussion of component wearout caused by testing. The NRC review of WCAP-10271 to date has resulted in a request for additional information the NRC felt necessary to complete the review. Information that was submitted to the NRC in response to that request includes an overall evaluation of the impact on plant safety of RPS surveillance, a discussion of the uncertainty of failure rates and common mode failure and more detail concerning the impact of surveillance intervals on RPS unavailability. WCAP-10271, Supplement 1, and the information provided to the NRC in defense of WCAP-10271 provides in a comprehensive form the information requested by item 4.5.3. The conclusion of WCAP-10271 and Supplement 1 is that although RPS unavailability is increased less frequent testing of RPS components is warranted and will result in an improvement in overall plant safety and equipment reliability.