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MURRAY R. EDELMAN
VICE PRESIDENT
NUCLEAR

April 6, 1984
PY-CEI/NRR-0100 L

Darrell G. Eisenhut, Director
Division of Licensing
Office of Nuclear Reactor Regulations
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Perry Nuclear Power Plant
Dockets Nos. 50-440; 50-441
Perry Nuclear Power Plant
Response to Generic Letter 83-28

Dear Mr. Eisenhut:

As you requested, the attached submittal forwards to you the status, plans and schedules for the Cleveland Electric Illuminating Company's Perry Nuclear Power Plant (PNPP) in response to Generic Letter 83-28, "Required Actions Based On Generic Implications of Salem ATWS Events."

This submittal presents the positions which the BWR Owners Group and INPO NUTAC have developed to address the generic concerns of the Letter, as well as PNPP status and scheduling information.

If you have any additional questions about our program, please feel free to call.

Very truly yours,

A. Kaplan for M. Edelman

Murray R. Edelman
Vice President
Nuclear Group

MRE:kay

cc: Jay Silberg, Esq.
John Stefano
Max Gildner

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CLEVELAND ELECTRIC ILLUMINATING COMPANY
PERRY NUCLEAR POWER PLANT

RESPONSE TO GENERIC LETTER 83-28
"REQUIRED ACTIONS BASED ON
GENERIC IMPLICATIONS OF
SALEM ATWS EVENTS"

APRIL 1984

SECTION 1.1

POST TRIP REVIEW: Program description and procedure

The program for review and analysis of unscheduled reactor shutdowns at Perry Nuclear Plant (PNPP) is under development. However, the following information is provided on the planned program and procedures for assuring that unscheduled reactor shutdowns are analyzed and that a determination is made that the plant can be restarted safely.

Item 1.1.1

The basic restart criteria developed by the BWR Owners Group, in combination with the draft INPO OP-211 recommendations, form the basis for the PNPP procedures. These PNPP specific procedures are under development and will be available for review when completed. The Owners Group guidelines are outlined below.

Based upon technical judgement utilizing approved plant procedures, control room indication and operator knowledge of the plant, the shift supervisor may make the decision to recommend restart of a plant. The following five restart criteria need to be met:

Criterion A

The plant is shown to be in a safe condition

The determination of the safe condition of the plant is assumed before any other criteria need to be examined. It is necessary to determine that safety limits have not been exceeded and that the issue at hand is one of justifying restart from a stable shutdown condition. If this is the case then the operator may begin an evaluation of the advisability of restart.

Criterion B

The cause of the event is either understood or, after a comprehensive investigation, is considered to have been a spurious trip with a reasonably low potential for reoccurence.

The operator has many sources of information available to him which can be used both as a diagnostic tool in evaluating the cause of an unanticipated scram and in the identification of other-than-expected performance of plant systems and equipment. The readouts of both safety related and non-safety related indicators (including such sources as the sequence of events recorder, alarm typer, trend recorder and process computer) provide a basis upon which technically defensible actions can be initiated to determine the cause of the event and assure that the cause of the scram no longer exists. See Caution No. 3 of the BWR Emergency Procedure Guidelines (EPG's). (Attachment 1) See also the response to section 1.1.4 and 1.2.

(Criterion B continued)

It is important to understand the cause of an unscheduled trip so that reoccurences can be minimized. However, it is not realistic to ignore the possibility for spurious trips whose cause can not be identified. In the event that the cause of the unscheduled reactor shutdown cannot be determined, the Plant Manager or designated alternate approves a restart based on the following conditions:

a) All reasonable actions to determine the cause have been considered.
b) No physical damage was done by the event and a determination has been made that the plant had not operated beyond the boundaries established by approved plant safety and transient analyses.
c) Safety systems have actuated properly.

The discussion of the qualifications and responsibilities of the personnel making the restart recommendation is included in sections 1.1.2 and 1.1.3.

Criterion C

The expected on-off automatic operation of plant safety related systems has been verified.

If the operator determines that a particular system should have initiated for a particular event, he need only establish that the system did indeed initiate and in the proper sequence. A detailed analysis of the actual performance of that system following an unscheduled shutdown is not a criterion for restart. Such a detailed analysis is accomplished through the normal surveillance testing procedure done at regular intervals. This step is consistent with the philosophy espoused in Caution No. 1 of the NRC approved BWR EPGs.

Since confidence in the accuracy of Control Room readout is provided both by the routine maintenance and surveillance activities associated with Engineered Safety Features, and normally scheduled and performed calibration activities associated with such devices, adherence to there efforts mitigates the need to enter into a complete recalibration (i.e., pressure, flow, operating times, etc.) or performance reevaluation of the adequacy of system operation.

Criterion D

Any need for corrective action has been determined and appropriately implemented.

Once the cause of the event is determined the operator then needs to determine what, if any corrective action(s) need to be implemented. The INPO Good Practice OP-211 contains Conditions I and II which are relevant to when the cause is understood. "Demonstrating compliance to these Conditions justifies the initiation of restart activities. When the cause of the scram is determined a decision can be made on the need for corrective action. Such a decision can fall into three categories: (a) no corrective action is necessary; (b) corrective action is necessary but does not need to be performed before restart (i.e.; Action is not required in order to meet Technical Specification conditions prior to restart); and (c) corrective action is required before restart.

If no corrective action has been determined to be necessary, normal restart procedures apply. If corrective action is necessary but is not required to meet Technical Specification requirements, then restart procedures apply and the needed corrective actions are taken following restart. If corrective action is required then it would be necessary to complete the effort before initiation of restart activities. These actions range in effort from a simple recalibration of the device causing the scram to replacement and/or recalibration of major portions of a system. This determination also needs to be based on the Technical Specification associated with startup activities (i.e., Technical Specifications allow restart with some devices out-of-service). Before startup activities are commenced, compliance to the Technical Specification must be assured. Also, assurance must be provided that, as a result of the investigation into the event, matters such as valve alignments are brought back into the proper sequence and/or arrangements.

Criterion E

The approval of the Plant Manager or designated alternate has been obtained.

The review of the reactor trip is performed by the Shift Supervisor and the Shift Technical Advisor (STA). The recommendation to restart is then made by the Shift Supervisor to the Plant Manager. The recommendation must be approved by the Plant Manager or designated alternate in order to authorize restart.

Item 1.1.2

The review and analysis of the unscheduled reactor trip will be performed by the Shift Supervisor and the STA. Input to the review process comes from operators or maintenance, I & C and other personnel involved in the reactor trip or corrective actions.

The responsibilities and authorities of the Shift Supervisor are detailed in FSAR Section 13.1.2.2. This says that "The Shift Supervisor on duty is responsible for operating the plant in compliance with licensing requirements, administrative controls and operating instructions. This includes, when warranted, approving on-shift operations that deviate from established procedures and instructions, evaluating operating experience and providing on-shift technical advice to the Unit Supervisors.

Administrative procedures will be written to clearly define the Shift Supervisor's command and control responsibilities and authorities and to emphasize his responsibility for safe operation of the plant. Those functions which clearly detract from responsibility for assuring safe operation of the plant will be assigned to other personnel."

The responsibilities of the STA are described in FSAR Section 13.1.2.3. A Shift Technical Advisor will be available to provide technical support to the Shift Supervisor, including advising him on the safety status of the plant, diagnosing plant accidents and recommending actions to mitigate the consequences of accidents. Further details on the duties of the STA are given in procedure TAP-0101, "Duties of the Shift Technical Advisor."

The responsibilities and authorities of the Plant Manager who approves the restart are included in FSAR Section 13.1.2.2.

Item 1.1.3

As stated in FSAR Section 13.1.3.1, "Perry Nuclear Power Plant follows the guidelines set forth in Regulatory Guide 1.8 for selection and training of management personnel. Table 13.1-1 lists members of the plant staff and designates equivalent ANSI/ANS 3.1-1978 titles as a comparison." The Shift Supervisor resumes are included in the FSAR Table 13.1-3. The resume of the Plant Manager is also included in Table 13.1-3. FSAR Table 13.2-1 shows the training schedule for PNPP positions including the Shift Supervisor, STA and Plant Manager.

Shift Technical Advisors have not yet been designated. They will be qualified and trained as described in FSAR Appendix 1A "Response to Requirements of NUREG-0737," Item I.A.1.1. PNPP has committed to provide a Shift Technical Advisor who offers shift technical support to the shift supervisor and who advises the shift supervisor on the safety status of the plant, diagnoses plant accidents, and recommends actions to mitigate the consequences of accidents. An STA at PNPP must have a bachelor degree in Engineering or related sciences or a High School diploma and sixty semester hours of college-level education in mathematics, reactor physics, chemistry, materials,

reactor thermodynamics, fluid mechanics, heat transfer, electrical and reactor control theory. In addition, an STA must have one year of professional level nuclear power plant experience. The STA's will complete additional instruction at PNPP including pertinent portions of on-site training dealing with FSAR accident analyses, technical specifications, normal and off-normal operating procedures and Perry system operating modes and construction.

Item 1.1.4

As stated in Item 1.1.1 above, the PNPP procedures which will address the sources of information used to conduct the review and analysis of an unscheduled reactor trip are under development and will be available for review when completed.

The plant information sources available at PNPP are described in Section 1.2 of this report. These include the Annunciator/Sequence of Events Recorder for assessing sequence of events during the scram, as well as analog recorders for assessing the time history of analog variables and the functioning of safety-related equipment.

When the plant computer is available there is additional sequence of events information on the sequence of events log, and time history and equipment functioning information on the post-trip logs.

In addition to all of the above, supplemental plant information is available through the Emergency Response Information System (ERIS).

The information gleaned from the above instrumentation is combined with operator observations during the transient, operator knowledge of the plant, post-trip observations of equipment status and available information from previous surveillance tests and transients in order to reconstruct the event accurately.

Item 1.1.5

As stated in Item 1.1.1 above, the PNPP procedures for Post-Trip Review which will address the methods and criteria for comparing the event information with expected plant behavior are under development and will be available for review when completed.

Item 1.1.6

As stated in Item 1.1.1 above, the PNPP procedures for Post-Trip Review which will address the need for independent assessment of an event are under development and will be available for review when completed. Guidelines on the preservation of physical evidence to support independent analysis of the event will also be included in those procedures.

Item 1.1.7

PNPP is establishing a systematic method to assess unscheduled reactor shutdowns. The procedures which address the above items will be available for review as stated above.

SECTION 1.2

POST TRIP REVIEW: Data and Information Capability

Item 1.2.1 Sequence of Events Assessment

1.2.1.1 General

Sequence of event discrimination is provided as a means of diagnosing causes of unscheduled turbine and reactor shutdowns. This information is available on the plant computer, the annunciator/sequence of events recorder (SER), and the instrument recorders. The Emergency Response Information System (ERIS) computer is also available as a supplementary tool to the above data systems, depending on the process variable. These event recording systems are non-safety-related unless otherwise stated herein.

Reactor Protection System trip inputs by channels and Emergency Core Cooling actuation signals are monitored by the sequential events log program of the plant computer. Turbine supervisory instrumentation, electrical transformers, breakers, busses, diesel generators, and feedwater equipment are monitored for sequential events by the SER system.

Additionally, ERIS is an integrated system that gathers required plant data, stores and processes that data, generates visual displays for the operator and other personnel who need plant status information, and provides printed records of transient events.

1.2.1.2 Plant Computer

1.2.1.2.1 Description

The sequence of events log is one of the special programs handled by the plant computer, to store inputs with two millisecond resolution. There are 128 NSSS digital inputs are assigned to this function.

Upon detection of a status change of any of the preselected sequential events contacts, the sequence-of-events log is initiated and signals the beginning of an "event". When 64 contact changes have been sensed or 30 seconds have elapsed since the first detected change, the log is automatically printed.

1.2.1.2.2 Parameters Monitored

Attachment A is a listing of the computer input list included in this program.

1.2.1.2.3 Time Discrimination Between Events

Changes of state of digital inputs received 2 milliseconds or more apart (for different points) and 20 milliseconds or more apart (for same point) are sequentially differentiated on the printed log.

1.2.1.2.4 Format for Displaying Data and Information

The sequence of events log is printed on the line printer located in the control room. The sequence of events log includes the event number, English description, and time of occurrence, which is printed in hours, minutes, seconds, and milliseconds. Once initiated, the sequence annunciator log will continue as long as events which have been recorded and stored remain to be printed. Sequence annunciator events occurring during output logging activities continue to be recorded. Status changes which occur when the edit table is full (128 events) may not have the correct time-of-occurrence recorded. If this happens, a "flag" is printed for these additional events to advise operations' personnel about misleading data.

1.2.1.2.5 Capability for Retention of Data and Information

All printed logs and strip charts are stored for future recall. The sequence of events log is retained as part of the on-site Records Management System. A computerized index system is provided for easy retrieval of data. Data will be maintained for the life of the plant.

1.2.1.2.6 Power Source

The plant computer and peripherals are powered by 120 VAC non-class 1E vital bus which is backed up through a static transfer switch and an inverter from a plant battery.

1.2.1.3 Annunciator/Sequence of Events Recorder

1.2.1.3.1 Description

An annunciator system and a sequence of events recorder (SER) are provided for each unit. The SER and annunciator system share inputs through diode isolation devices. The SER can record up to 300 events with 48 character English alpha-numeric printout on a 120 character per second typer. The SER has a solid state sequential memory capable of storing 128 events with one millisecond resolution. Events in excess of the first 128 will also be printed but in the order scanned.

1.2.1.3.2 Parameters Monitored

The annunciator system provides the operator with visual/audible indication of abnormal plant and equipment paramenters, as well as return to normal indication. Alarms are displayed on lighted engraved front window light boxes on the most appropriate panel. The more significant events or parameters, which could result in major equipment trips are monitored by the SER. These inputs primarily include electrical bus abnormalities, transformer trouble, turbine supervisory trips, generator faults, large motor trips, and diesel generator actuation. A listing of the SER inputs is found on Attachment B.

The majority of the SER inputs are also wired as annunciator inputs. Transformer faults are routed to the SER only.

1.2.1.3.3 Time Discrimination Between Events

The SER memory can discriminate and print up to 128 inputs with one millisecond resolution. When the memory is full, additional inputs will be printed in the order scanned.

All annunciator inputs are continuously monitored, but time discrimination is available only for the first 128 SER events.

1.2.1.3.4 Format for Display

The SER interface with the operator is the typer located in the control room. The output format includes day of year, time of day in hours, minutes, seconds and milliseconds, event status code, event number and 48 character maximum English identification.

A backup printer in the SER cabinet automatically operates on failure of the typer and prints time, event number and event status.

1.2.1.3.5 Data Retention

SER data from the typer or printer is retained in the plant files. Event information is stored in the SER memory in event of typer failure.

1.2.1.3.6 Power Sources

The annunciator/SER system is powered from 125 VDC.

1.2.2 Time History Assessment

1.2.2.1 General

Capability for assessing the time history of analog variables needed to determine the cause of unscheduled reactor and turbine shutdowns, and the functioning of safety-related equipment is available to the operator primarily by means of analog recorders and the post trip logs of the plant computer. Periodic logs and special logs may be used to assess plant conditions also.

The variables on the dedicated logs and recorders are those associated with the Reactor Protection System (RPS), the Emergency Core Cooling System (ECCS) actuation, the Nuclear Steam Supply Shutoff System (NSSSS), the Reactor Core Isolation Cooling System (RCIC), and the Redundant Reactivity Control System (RRCS). They are among those identified as type B, C, D, and E variables in Regulatory Guide 1.97.

1.2.2.2 Analog Recorders

Safety related display instrumentation for which a trend display is deemed to be an important operating tool are assigned to single and multiple pen/point recorders in the control room. The instrumentation and ranges were selected on the basis of giving the reactor operator the necessary information to perform normal plant startup and loading operations and to be able to track all important process variables during operational perturbations.

These analog records also serve as backup historical records to plant computer logs, and some provide verification to the operator that certain events have occurred. For example, power range and startup range neutron monitoring recorders (C51-R603A, B, C, D and C51-R602A, B) indicating downscale verify reactor shutdown has occurred as otherwise indicated by CRD status mimic, neutron monitoring system indicating lights and annunciator on the main reactor control panel H13-P680.

Two trend recorders are also available to trend any computer variable or calculated point, as selected by the operator. These recorders may be of value to watch a trace of a particular variable during any phase of plant operation.

1.2.2.2.1 Parameters Recorded

The parameters monitored as well as recorder characteristics of the dedicated analog recorders are given in Attachment C. The majority of these dedicated analog parameters are continuously monitored on single or multi-pen recorders. Certain relatively slow response temperature measurements, turbine supervisory instruments and containment combustible gas concentrations are handled by multipoint strip chart recorders.

The parameters selected are those that reflect the condition of the reactor, turbine and containment, and include all recorders listed as safety related display instrumentation in the PNPP FSAR, Section 7.5.

1.2.2.2.2 Data Retention

Most strip charts for the analog records accumulate data for one month on a chart roll at specified chart speeds, others record data for shorter periods of time. The completed rolls are stored in the plant files.

1.2.2.2.3 Power Sources

Chart motors for the analog recorders are powered from non-interruptable vital busses as noted in Attachment C.

1.2.2.3 Plant Computer

1.2.2.3.1 Description

The post trip log is a printout of historical data collected at a predefined interval for a specified number of minutes before and after a plant trip. There is a separate log for NSSS and BOP inputs, in chronological order.

The trip review data file, consisting of up to 64 significant plant variables, is periodically updated in memory. The file contains 31 BOP and 24 NSSS variables permanently assigned, and 9 BOP variables selectable by the operator. When a trip is detected, the pre-trip data is "frozen" in memory and collection for the post-trip data is initiated.

A trip mechanism is activated by a trip of the reactor or turbine/generator unit. Upon activation of the trip mechanism, collection of data continues for an additional 30 minutes for the BOP log and 5 minutes for the NSSS log.

After all data is collected, the log automatically prints. Once output is initiated, the log runs to completion.

1.2.2.3.2 Parameters Monitored

See Attachments D & E for listings of preassigned variables and their computer identification numbers and scan rates. The BOP data is updated at 15 second intervals for 30 minutes. The NSSS data is updated at 5 second intervals for 5 minutes.

1.2.2.3.3 Format for Display

The first line of the printout identifies the data point and serves as a column heading for subsequent printouts. The values of the data points commence on the second line under appropriate headings. Points which are bad, deleted from scan or processing, or supplied with substitute values, are identified in the log.

The computer distinguishes data before and after the trip, i.e., rows of asterisks, blank lines, etc. At this point in the log, the disturbance(s) causing the trip log to be activated are identified.

1.2.2.3.4 Retention of Data

After all trip data has been collected, the data is printed. The printed logs are retained in the on-site Records Management System. The data will be maintained for the life of the plant.

1.2.2.3.5 Power Source

The plant computer and peripherals are powered by a 120 VAC and non-class 1E vital bus which is backed up thru a static transfer switch and an inverter from a plant battery.

1.2.2.3.6 Other Logs

The plant periodic log is composed of hourly, daily, and monthly values. The hourly values consist of sensor readings, averages, accumulations and performance calculation results. The daily and monthly values consist of daily and monthly averages and accumulations.

There is provision for 10 special log groups. Each group contains up to 48 variables. Upon operator request, a special log is initiated and printed out until cancelled. Special logs include the time, log number, log title, point identification, and the value or status.

Print intervals from one minute to 24 hours are selectable with default to 10 minute print if not selected.

A turbine and generator log of 33 assigned variables is listed in Attachment F. This log is printed once daily, and contains four sets of readings at one minute intervals.

1.2.3 Other Data and Information Capabilities

1.2.3.1 Emergency Response Information System

1.2.3.1.1 Description

The Emergency Response Information System (ERIS) was designed to implement NUREG 0696 and other associated regulations intending to upgrade the understandability of plant information. As implemented at PNPP, ERIS monitors approximately 2200 permanent channels. To accomplish this ERIS consists of two (2) Digital VAX 11/780 computer systems. One system is for the Real Time Analysis and Display (RTAD) subsystem, and the other for Transient Recorder and Analysis (TRA) subsystem. Display systems are located in critical areas of the plant to allow operator access to the acquired information. These devices include and are located as follows: CRT displays and CRT copiers and plotters in the Control Room (3), Technical Support Center (TSC 4), Emergency Offsite Facility (EOF 2), CRT displays in the Remote Shutdown Room and Health Physics Office, and line-printers in the TSC for the printing of the various logs produced.

Major functions for ERIS are as follows:

a. Critical parameter validation.

b. Display of selected emergency response (SPDS) CRT displays in the Control Room, TSC, EOF, and Remote Shutdown Room.

c. Real-time and historical trend plots.

d. Two dimensional plots.

e. Sequence of events resolution.

f. Transient data recording and generation of associated hard copy.

1.2.3.1.2 Parameters

The parameters monitored will be listed in the ERIS I/O list C95-4030. A preliminary list of permanent monitored variables is included in Attachments G, H, and I. Attachments G includes ERIS Digital inputs, Attachment H includes ERIS Analog inputs, and Attachment I includes ERIS Control Rod Position inputs.

1.2.3.1.3 Time Discrimination

ERIS has the ability to resolve events that are five (5) or more milliseconds apart on both analog and digital events.

1.2.3.1.4 Display/Report Format

A. Sequence of Events Log. The sequence of events log shall be accessible from either CRT monitors or printed hard copy and will consist of single line entries for each status change and shall designate its time of occurence, point identification and description, where:

1. Time is expressed to the nearest hour, minute, second and millisecond.

2. A comprehensive functional description (name) of the input variables shall be provided.

3. Status designates the nature of the input event (e.g., alarm, high, low, set, etc).

B. Transient Data Log. The transient data log function shall monitor selected process input channels, composed input points, and transformed or calculated input points. It shall measure these selected variables at preselected time intervals. These intervals shall be the same as the scan intervals selected on the ERIS I/O list. Up to 700 inputs are capable of being treated in this manner. The operator is capable of initiating the printing or plotting of a transient log by entering the point identification and time interval of interest. This log will include the time the log was printed or plotted, point identification, variable identification, status or value of the point and real time when either the reading was taken or the calculation was made.

C. Variable Trend and Plot (time display plot). This is an XY plot of variables with engineering units versus elapsed time, either real time or historical.

D. Data Trend (tabular trend). This is a table of time in minutes, seconds, and milliseconds and corresponding engineering unit values of up to 6 operator selected variables, either real time or historical.

E. Trigger Mode Data Capture. In the trigger mode, a real-time check shall be performed on up to 100 prespecified process variables for change to a prespecified state. A change to this state shall cause a trigger generation which shall result in a data capture of 1/10 of pretrigger test interval and 9/10's post trigger interval data for those points identified as "startup and extended transient recording option" variables in the ERIS I/O List. The test interval may be selected by the operator. Recording will automatically terminate after the aforementioned data has been captured. A trigger may also be generated by operator input. Trigger recording may be terminated by the operator at any time.

1.2.3.1.5 Historical (Archival Data) Recording

The operator has the capability to store and retrieve all captured data from magnetic tape.

1.2.3.1.6 Power Sources

The ERIS computer system is powered from the ERIS Uninterruptible Power Supply (UPS) which contains a dedicated one (1) hour battery for the entire UPS load. This power source feeds two main transformer - distribution panels (one for each computer system), one miscellaneous ERIS distribution panel and several other non-ERIS panels. This arrangement was chosen to allow one transformer to be shutdown for maintainence without making ERIS unavailable.

Item 1.2.4 Data and Information Capabilities

No changes are planned to existing data and information capability.

Sequence of Events Assessment

Sequence of events discrimination and display of the combined systems described in section 1.2.1 are adequate for operator diagnosis of plant trip and emergency systems initiation. NSSS plant events are primarily handled by the plant computer while BOP events are handled by the SER. Additional sequence of events information is contained in the ERIS system.

The information display to the operator is handled automatically in a concise format, and records are retained for historical purposes.

Time History Assessment

Time history records of major plant parameters are available to the operator to assess events immediately preceding a reactor or turbine trip and for a post trip period of time. The bulk of this data is available on a digital trend printout via the plant computer post trip logs. Rate of change of those parameters assigned to analog recorders can very simply be observed by the control room operators.

The BOP post trip log continues for 30 minutes; however, the NSSS post trip log terminates after 5 minutes.

Other Data and Information Capabilities Assessment

Some of the SER and Plant Computer inputs are also monitored by ERIS thus providing a degree of redundancy for these systems. Additionally this enables events recorded on all three systems to be time correlated manually. This capability would not be present otherwise because the system clocks for the Plant Computer, SER, and ERIS are not synchronized.

All ERIS inputs are time-tagged and once recorded are capable of being recalled by the operator for review and evaluation. This capability is in addition to the established logs and graphs that ERIS automatically produces.

While the SER and Plant Computer are capable of assisting the operator in determining the cause of an unanticipated plant event without ERIS, the existence of ERIS provides an additional tool for the operating staff to diagnose the cause of the event.

SECTION 2.1

EQUIPMENT CLASSIFICATION AND VENDOR INTERFACE: Reactor Trip Function Components

This section consists of two activities. The first is to confirm that all safety-related components required to trip the reactor are identified consistently on documents, procedures and information handling systems used in the plant. The second activity is to maintain a program to ensure that vendor equipment technical information is current, controlled and appropriately referenced in plant documents.

The following paragraphs relate to the first activity. PNPP is presently involved in verifying and approving our Quality Items List (Q-List). This list assigns safety classifications at the component, Master Parts List (MPL) level. It is planned to combine this list with the list developed by the Equipment Qualifications element to create a comprehensive tool for classification usage. Once this master list is approved, all the safety related components within the systems at PNPP will be readily identifiable. All classification determinations for maintenance, work orders and parts procurement are made by checking the master list. This single source of component classification ensures that consistency is maintained in all safety-related activities.

The Reactor Trip System as described in NUREG-1000 includes those power sources, sensors, initiation circuits, logic matrices, bypasses, interlocks, racks, panels and control boards, and actuation and actuated devices, that are required to automatically initiate the control rods in order to assure that specified acceptable fuel design limits are not exceeded.

As described in section 3.1.2.5 of NUREG 1000, the GE Boiling Water Reactor trip system design differs from the PWR designs. The GE reactor trip system consists of redundant plant process instrumentation that feed one-out-of-two-taken-twice logic that initiates a reactor trip by deenergizing solenoid operated scram pilot valves which vent air from the scram valve diaphragms and insert the control rods. These components are contained within several systems at PNPP rather than one system called a reactor trip system. The plant systems involved are as follows:

a) Sensors - Inputs to the Reactor Protection System are provided from the Neutron Monitoring, Control Rod Drive, Nuclear Boiler, and Process Radiation Monitoring Systems.
b) Power Sources - Supplied by the Reactor Protection system.
c) Initiation Circuits, Logic Matrices, Bypasses, Interlocks, Racks, and Panels - Contained in the Reactor Protection System and Neutron Monitoring System.
d) Actuated Devices - Contained in the Control Rod Drive System.

Since creation of a new "Reactor Trip System" would cause confusion with the existing plant systems, we will respond to Letter 83-28 Item 2.1 on a system level basis by covering the systems that perform the reactor trip function as part of the Item 2.1 response, and provide the program for the remaining safety related systems as part of the Item 2.2 response. The Item 2.1 review consequently includes all "Reactor Trip System" components as well as all other safety related components involved in the reactor trip function. The specific components that form a "Reactor Trip System" were not separately identified.

In response to the concerns expressed in the second part of Section 2.1, CEI joined with 55 other utilities and formed an INPO Nuclear Utility Task Action Committee (NUTAC). This committee has developed and approved an industry-wide Vendor Equipment Technical Information Program (VETIP), which is described in detail in Attachment (2). This program promotes interaction among the major organizations involved with commercial nuclear power. As illustrated in Figure 1 to Attachment (2), individual utilities exchange and disseminate safety-related systems and components information with vendors, the NRC, INPO and other utilities. This exchange of information takes place via written notification (i.e., Licensee Event Reports, NRC I & E Bulletins and Information Notices, industry newsletters, etc.) as well as industry meetings and day to day verbal communications. The purpose of these information exchanges is to share equipment technical information to improve the safety and reliability of nuclear power generating stations. The primary purpose of the VETIP program is to ensure that current information and data will be made available to those personnel responsible for developing and maintaining plant instructions and procedures. These information systems and programs currently exist and are capable of identifying to the industry precursors that could lead to a Salem-type event. It should be noted that the VETIP is industry-controlled and a mainly hard-ware oriented program that does not rely on vendor action, other than the NSSS supplier, to provide information directly to utilities. Instead, the VETIP provides information developed by industry experience through Significant Event Reports (SER's) and Significant Operating Experience Reports (SOER's) to the equipment vendor for comment before it is circulated to the utilities concerned.

In addition to the VETIP, PNPP has an existing vendor equipment information program with General Electric (GE) Company, our NSSS vendor. This program consists of two major categories: (a) information regarding safety-related systems and components; and, (b) technical information intended to enhance safety and non-safety related equipment reliability and improve plant performance. These programs include, but are not limited to:

(1) <u>10CFR21 Reporting</u> - The General Electric Company has established a reporting system to handle safety concerns that complies with the requirements of 10CFR21.

(2) Urgent Communications - In addition to the 10CFR21 reports, a procedure for handling urgent communications to BWR owner/operators has been established for use in providing fast notification of safety concerns. These communications are usually in the form of a short letter which provides a brief explanation and advice or precautionary measures to be observed to avoid potential operational hazards. Due to their urgent nature, these communications are processed to operating plants by the most effective method (i.e. telex, telecopy, cable, special mail handling, etc.) and, if transmitted in written form, they will be followed up or preceded by telephone calls.

In addition the following information is also made available to us.

(1) Service Information Letters (SILs) - These provide recommendations for equipment modification, plant design improvements or changes to procedures to improve plant performance. They are distributed through the GE Domestic Apparatus and Engineering Service Operations (DAESO) or GE Nuclear Services Operation Regional Offices and are normally followed up by discussion during periodic service plan conferences. A PNPP procedure is in place to control handling of SIL's.

(2) Turbine Information Letters (TILs) - These documents are issued by GE's Large Steam Turbine Generator Department to provide descriptions of product problems/improvements and to recommend modifications that will mitigate problems or improve product performance. These documents are distributed through the GE-DAESO District Offices are followed up by the Turbine Department to encourage implementation.

(3) Service Advice Letters (SALs) - These documents are issued by GE Product Departments other than the San Jose based Nuclear Energy Product Departments and are used to provide notification of product problems and/or service information on a broad range of GE consumer and industrial products. Those Service Advice Letters that are recognized by the issuing product department as applying to devices used in nuclear plants are specially identified for distribution to all nuclear plants.

(4) Operation and Maintenance Manuals - These documents are issued by all GE product departments to provide instructions for installation, operation and maint nance of GE designed repairable equipment and systems. Final revisions to the manuals provided for the NSSS scope of supply are delivered as contractually required, and usually are shipped at about the time of plant commercial operation.

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

(6) Field Disposition Instructions (FDIs) - These documents are used to communicate engineering instruction to the field that implement approved design modifications of GE supplied, NSS equipment or procedures, authorize field work, and confirm that the tasks have been completed on requisition plants.

(7) Field Deviation Disposition Requests (FDDR's) - These documents are used to communicate requests for nonconformance dispositions on GE supplied NSSS equipment or service on requisition plants.

SECTION 2.2

EQUIPMENT CLASSIFICATION AND VENDOR INTERFACE: Programs for all Safety-Related Components

Item 2.2.1

2.2.1.1

PNPP's safety-related component classification is based on the NRC guidelines that define safety-related structures, systems, and components as those that are relied upon to remain functional during and following design basis events to ensure: (1) the integrity of the reactor coolant boundary, (2) the capability to shut down the reactor and maintain it in a safe shutdown condition, and (3) the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the guidelines of 10 CFR Part 100.

The Q-list review program identifies components as safety-related if it's function is determined to be required to meet the above guidelines, in accordance with the definitions below. (A complete listing of the PNPP safety classifications is contained in Attachment 3.)

A pressure boundary component (PBC) has no parts that have to move to mitigate the consequences of a design basis event but must stay physically intact to form a pressure boundary. Active components (AC) involve movement or activity in order to mitigate the consequences of an event. An "AC" classification signifies the component also has a safety function that includes maintenance of a pressure boundary, unless appropriately noted. Isolation devices needed to separate the Class 1E and Non-class 1E electrical systems as well as surveillance and auxiliary devices required by IEEE 308-1974 are safety-related. Electrical components and circuits which must function to supply electrical power during an event are also safety-related.

2.2.1.2

The information handling system used for identifying component classifications is a computerized system called the "Perry Material Management System" (PMMS). The details on the quality-related fields controlled in this program are in contained in Attachment 3.

The original preparation of the Q-list was performed by a consultant and was controlled as described in Section 2.2.1.4. This data is subsequently reviewed by PNPP personnel. This data is then entered into the computer database through a controlled and verified mode. The program is designed in such a manner to provide an auditable record of all transactions, and clearly indicate which data has been approved.

After a component has been classified as safety-related, a procurement requirements evaluation is performed to determine the Specific Technical and Quality Assurance requirements to be used. This data is also entered into the database in a controlled manner. PNPP procedures include a section on Quality Items List Preparation which includes information on the development and verification processes.

2.2.1.3

The Work Order process includes work requests, work orders, corrective maintenance and repetitive maintenance (e.g., Technical Specification surveillances, inservice inspections, inservice testing and mechanical and electrical preventative maintenance).

PNPP personnel work through the Perry Plant Maintenance Information System (PPMIS) to determine which work is safety-related. The MPL number from the Work Order is entered into PPMIS which automatically will go to the Q-list to determine the necessary information, which is then printed out. These Work Orders are verified. Non-equipment portions of PPMIS provide information for Work Orders which do not deal with plant equipment.

Parts Procurement procedures establish the requirements for procuring items such as spare parts, material and replacement components.

The requisitioner determines the stock code then contacts the warehouse personnel. The Q-list is checked to determine the current procurement requirements, and a purchase requisition is then initiated. When a purchase requisition is generated by the computer, whether based on a preestablished reorder point or some present need, these specific, approved data fields are reproduced on the procurement document. Where procurement requirements do not exist, a Q-list evaluation is initiated. If a Q-list evaluation has been performed but the needed procurement requirements are not defined, a Procurement Requirements Evaluation (PRE) is initiated.

2.2.1.4

During the initial preparation of the Q-Lists by the consultant, audits were performed to ensure compliance with the Q-List preparation procedures.

Reviews of the work performed by the consultant are conducted by Perry Plant Department (PPD) and Nuclear Engineering Department (NED).

Utilization of the Q-List will be checked during audits and surveillances and by the NQAD review of documents such as work orders and procurement documents.

Access to the computer data base is controlled through logon/password assignment. Assignment of the logons/passwords to specific personnel is controlled by the General Supervisor, Perry Plant Department Maintenance Section. Logic is designed into the computer programs to prevent inadvertant changing of controlled data fields.

2.2.1.5

The PNPP Equipment Qualification Program is discussed in the FSAR Section 3.10 and 3.11.

As stated in Section 2.1 the Q-list will become the central data base for determining safety classifications, seismic categories, procurement requirements and storage requirements.

Safety-related components are specified to be qualified to, and the qualification documentation is reviewed to assure compliance with, IEEE 323-1974 (as modified by R.G. 1.89, Rev. 0, and NUREG 0588 Category I) and IEEE 344-1975 (as modified by R.G. 1.100, Rev. 0) so as to ensure the equipment can perform its design safety function when exposed to normal, abnormal, accident and post-accident environments. The qualification documentation is also reviewed to determine the qualified life of the component or part.

Item 2.2.2

As stated in the Executive Summary of the report by the NUTAC on Generic Letter 83-28 Section 2.2.2 (Attachment 2), "Generic Letter 83-28 was developed following investigations by the NRC on the Salem events. As a result of these investigations, the NRC determined that better control and utilization of information regarding safety related components might have helped to prevent these events. The NUTAC identified a program to better ensure that plant personnel have timely access to such information.

The NUTAC efforts were guided by the recognition that individual utilities have the greatest experience with and are most cognizant of the application of safety-related equipment. Vendor involvement with such equipment is generally greatest during construction and initial operation of the plant. Vendors are not familiar with the surveillance or maintenance histories, nor with the application of the equipment or its environment. This type of information is most readily available at the plant level within individual utilities.

Based on this recognition, the NUTAC investigated the mechanisms currently available to facilitate information exchange among utilities. The NUTAC identified four activities that currently address information about safety-related components. These are routine utility/vendor and utility/regulator interchange, and the SEE-IN and NPRDS programs managed by INPO.

It was the assessment of the NUTAC that these existing activities, if properly integrated and implemented, would provide a framework for an overall program to ensure effective communication of safety related information among all utilities. Accordingly, the program developed to accomplish this goal (VETIP) utilizes the existing efforts as elements of a more comprehensive program.

The VETIP combines these existing programs, incorporating enhancements, with a coordinated program within each utility. A key element of the VETIP is the development by each utility of an active internal program to contribute information to the NPRDS and SEE-IN programs and to utilize the results of these programs.

The VETIP has been developed to ensure that nuclear utilities have prompt access to and effective handling of safety-related equipment technical information. In addition, it is responsive to the intent of Generic Letter 83-28 Section 2.2.2." Further details are provided in Attachment 2.

Cleveland Electric Illuminating Company endorses the Vendor Equipment Technical Information Program developed by the INPO NUTAC.

PNPP-specific handling of vendor equipment technical information is controlled. All Project personnel are responsible for transmitting, upon receipt, vendor manuals, revisions, and necessary changes to the Document Control Center for controlled processing and distribution. This ensures that appropriate reviews are conducted, approvals are obtained and that all users work to the latest revision.

The Document Control Center controls manuals by transmitting manuals/revisions/inserts/updates to approved holders, maintaining logs and files, and conducting follow-up on transmittals not returned.

The responsible engineer reviews, coordinates other reviews, establishes distribution, approves, sends vendor manuals to the Document Control Center, and controls issuance to the contractors.

Implementing procedures are being revised using the guidelines of INPO Good Practice MA-0304. These are scheduled to be revised by June 1984.

SECTION 3.1

POST MAINTENANCE TESTING: Reactor Trip Function Components

Item 3.1.1

For each surveillance requirement in the Technical Specifications there will be a corresponding instruction.

The PNPP procedure which controls the preparation and formatting of maintenance instructions specifies that Section 6.0 of each instruction shall describe the post-maintenance requirements for the work. It states that "the post maintenance requirements shall be a means to verify that the preventive or corrective maintenance was accomplished correctly, and that the equipment, upon such verification, is indeed operable. The verification method used should be of a type to check upon the specific type of maintenance, preventive or corrective, that occurred. In many cases, it is not feasible to individually check upon the multitude of actions that occurred. Instead, a functional or operability check of the entire device would not only suffice, it would be the preferable method. In this way the integrated operation of all the items would be verified. Examples of such tests would be after the completion of work on a motor to check it for amperage, overheating, vibrations, and proper operation of what it drives. The method selected shall test all pertinent functions of the equipment that may have been affected by the maintenance activity. Technical Specifications, Inservice Inspection requirements, and licensing commitments shall be checked to see if they require any specific post maintenance requirements. General statements, such as returning tools, replacing covers, remo[illegible]gs, informing supervisors, etc. should not be included. For [illegible] if the reinstallation of covers is so critical that the no[illegible]t of maintenance can not be considered sufficient, then it [illegible] be included in Section 5.0, Instructions. The vendor's manual shall be used to the maximum extent possible for verification and acceptance criteria. The latest set of baseline data should be checked and utilized in the development of the acceptance criteria."

The majority of the PNPP-specific instructions are still in draft stages, but will be written to follow this procedure.

Item 3.1.2

The PNPP procedure which controls the preparation of maintenance instructions specifies that "maintenance instructions shall be written by qualified individuals and be based upon the vendor's technical manual, equipment qualification packages, operating experience, INPO findings, industry news letters, startup or test data, vendor interface, information supplied by Perry Plant Department (PPD) and/or Nuclear Engineering Department (NED) engineering groups, and any other pertinent technical information. Prior to using any vendor supplied information, the writer must assure himself that it is site approved. The instruction as written shall be self standing. It should be remembered that in most cases when vendors write their manuals, they write them generically and generally do not know how their product will be installed or used. Only the writer knows this. As such, the instruction shall be written as it applies to Perry."

The majority of the maintenance procedures are still in draft stages but will be written to follow this procedure.

Item 3.1.3

See response to Item 4.5.3. If any recommended changes to Technical Specifications result from the Item 4.5.3 review effort they will be submitted for staff approval.

SECTION 3.2

POST-MAINTENANCE TESTING: All Other Safety-Related Components.

Item 3.2.1

See answer to Item 3.1.1.

Item 3.2.2

See answer to Item 3.1.2.

Item 3.2.3

See answer to Item 3.1.3.

SECTION 4.5

REACTOR TRIP SYSTEM RELIABILITY: System Functional Testing

Item 4.5.1

The diverse reactor trip systems at Perry include the Reactor Protection System (RPS) and the Alternate Rod Insertion trip feature of the Redundant Reactivity Control System (RRCS).

On-line functional testing of the RPS will be performed consistent with the Technical Specifications. Channel functional testing is performed on the multiple and diverse reactor transient trip sensors, the Average Power Range Monitor and Intermediate Range Monitor reactor trip signal channels, and the multiple and diverse Scram Discharge Volume High Water Level trips. During the required trip sensor channel tests identified above, each scram contactor which actuates the Scram Pilot Solenoid Valves is tested. The simple operation of the scram contactors minimizes concerns of wear, and frequent testing assures that any failures are detected early. The Scram Pilot Solenoid Valves which are actuated by the scram contactors are all tested regularly. Redundant Electrical Protection Assemblies (EPAs) which protect the Scram Pilot Solenoid Valves from low voltage chattering (and the associated potential consequence of accelerated wear) are also functionally tested. These surveillance testing requirements related to the Scram Pilot Solenoid Valves assure that the probability of undetected failures of these independently acting solenoid valves is small.

Channel functional tests are performed on-line for the following sensor trips:

- Reactor Vessel Dome Pressure-High
- Reactor Vessel Water Level-Low
- Reactor Vessel Water Level-High
- Main Steam Line Isolation Valve-Closure
- Main Steam Line Radiation-High
- Drywell Pressure-High
- Turbine Control Valve Fast Closure, Control Oil Pressure-Low
- Turbine Stop Valve-Closure

Channel functional tests are also performed for Average Power Range Monitors and Intermediate Range Monitors.

In References 1 and 2, it is shown that each of the above plant variables used to initiate a protective function is backed up by a completely different plant variable. In fact, for the most frequent transients, scram is initiated by three diverse sensors in all but one case (regulator failure-primary pressure increase which is initiated by two diverse sensors). This indicates that adequate redundancy exists in the design to provide protection against multiple independent sensor failures. Also, diversity among sensor types reduces the potential for common cause failures, failures due to human error, and increases in failure rate due to wearout.

Each sensor channel functional test includes full actuation of the associated logic, the two output scram contactors in each channel, and the individual CRD scram air pilot valve solenoids for the associated logic division (solenoids from both logic Division A and B are required for scram initiation).

The most credible failures within the RPS logic will de-energize a set of scram solenoids which causes a half scram, i.e., one of the two scram solenoids required for scram initiation is de-energized at some or all hydraulic control units. These failures would be "SAFE" failures that would increase the probability of plant shutdown.

The less credible logic failures which prevent a channel from de-energizing will be detected during channel functional tests in compliance with Technical Specification requirements. The tests described above ensure that an increase in failure rate due to a wearout condition or a common cause failure potential could be detected early and corrective action taken before the failure condition becomes systemic.

Other channel functional tests include testing of the Scram Discharge Volume (SDV) Water Level-High trip and manual scram trip and test of the reactor mode switch in the shutdown position every refueling. The first two trips involve on-line testing and the latter mode switch test can only be conducted during reactor shutdown. The manual scram trip can be tested on-line without creating a scram.

The testing of the SDV Water Level-High trip is considered adequate based on the current designed redundancy and diversity incorporated into the system. There are two diverse and redundant sets of level sensors which scram the reactor in the unlikely event of high water level in either SDV. These trips are designed to allow sufficient scram water discharge volume given the scram trip point is reached.

Reference 2 concluded that reactor shutdown can be achieved if at least 50% of the control rods in the checkerboard pattern and 69% in a random pattern are inserted in the core. The probability of independent failure of enough rods to prevent shutdown is negligible. The most unlikely type of failure would be some common cause mechanism that if undetected over a long period of time could cause unsafe shutdown. The Technical Specification surveillance requirements and PNPP instructions adequately ensure that a failure mechanism affecting several individual drives (considered to be very remote) would not go undetected. One of the major features that ensures that several drives do not fail at one time due to wearout or a common cause is the staggered maintenance and overhaul of selected CRDs or Hydraulic Control Units (HCUs) at refueling outages. This ensures a mix of drives by age, component lot, maintenance time, servicing personnel, and testing.

The scram insertion time tests include, in addition to drive timing and insertion capability, a test of operability of the HCU scram insert and discharge valves including associated scram air pilot valves. As stated in the previous paragraph, the required testing given in the surveillance instructions ensures that a systemic failure mechanism in the HCUs would be detected early enough and corrective action taken before the condition becomes a critical failure preventing scram.

As a diverse trip feature PNPP has a safety-related Alternate Rod Insertion (ARI) feature of the Redundant Reactivity Control System (RRCS), which is designed to increase the reliability of the Control Rod Drive system scram function. ARI provides for insertion of reactor control rods by isolating and depressurizing the scram air header through valves which are redundant and diverse from the reactor protection system scram pilot solenoid valves. Diversity is provided in that the ARI valves energize to function and are powered by a DC source, where as the scram pilot solenoid valves de-energize to function and are AC powered.

The RRCS signal to insert control rods results in energizing eight ARI valves. Four valves provide for venting of the A and B HCU scram valve pilot air headers to atmosphere to depressurize the headers and scram all rods. Two valves in series assure venting of air from the air header in the event one or more of the ARI valves fails. Two additional valves vent the valve operators of the scram discharge volume drain and vent, valves, closing those valves and isolating the SDV.

The RRCS sensors monitor reactor dome pressure and reactor water level. The sensors, transducers, and trip units are Class 1E, independent from the RPS, and environmentally qualified to perform their protective function. The logic will cause the immediate energization of the Alternate Rod Insertion valves when either the reactor vessel high dome pressure trip setpoint or low water level 2 setpoint is reached. Energization of the RRCS ARI valves depressurizes the scram air header independent of the logic and vent valves of the RPS system.

The RRCS is continually checked by a solid state microprocessor based self-test system. This self-test system checks the RRCS sensors, logic, protective devices and itself.

Although on-line functional testing of other plant diverse trip features is required by Generic Letter 83-28, PNPP is not designed to permit periodic on-line testing of the Alternate Rod Insertion (ARI) valves. Functional testing of these valves during plant operation would require a plant scram, resulting in an unnecessary challenge to plant safety systems and therefore a potential degradation in plant safety. A functional test of the ARI valves during shutdown will be performed in accordance with Technical Specification requirements.

In summary, the current Reactor Protection System on-line surveillance testing requirement, in conjunction with multiple and diverse sensors, assures that the probability of failure of enough control rods to prevent reactor shutdown is negligible. In addition PNPP's Redundant Reactivity Control System sensors, logic and ARI feature with periodic testing, further increases the reliability of the scram function.

REFERENCES

1. NEDO-1-189, "An Analysis of Functional Common-Mode Failures in GE BWR Protection and Control Instrumentation," L. G. Frederick, et. al., July 1970.

2. "BWR Scram System Reliability Analysis," W. P. Sullivan, et. al., September 30, 1976 (Transmitted in letter from E. A. Hughes (GE) to D. F. Ross (NRC), "General Electric Company ATWS Reliability Report," September 30, 1976).

3. Required Actions Based on Generic Implications of Salem ATWS Events, D. G. Eisenhut to Operating Reactor Licensees, July 8, 1983, NRC Generic Letter 83-28.

Item 4.5.2

Included in Item 4.5.1.

Item 4.5.3

CEI is participating in the BWR Owners Group Technical Specification Improvements Committee program. This program will review existing intervals for on-line functional testing required by Technical Specifications to determine that the intervals are consistent with achieving high reactor trip system availability when accounting for considerations such as:

a) Component failure rates.
b) Common mode failure rates.
c) Reduced redundancy during testing.
d) Human error rates during testing.
e) Component "wearout" rates caused by testing.

CEI will then utilize the results for specific application to PNPP.

The schedule for the above generic approach is currently being prepared by the Technical Specification Improvements Committee of the BWR Owners Group.

Attachment 1

OPERATOR PRECAUTIONS

GENERAL

This section lists "Cautions" which are generally applicable at all times.

CAUTION #1

Monitor the general state of the plant. If an entry condition for a [procedure developed from the Emergency Procedure Guidelines] occurs, enter that procedure. When it is determined that an emergency no longer exists, enter [normal operating procedure].

CAUTION #2

Monitor RPV water level and pressure and primary containment temperatures and pressure from multiple indications.

CAUTION #3

If a safety function initiates automatically, assume a true initiating event has occurred unless otherwise confirmed by at least two independent indications.

CAUTION #4

Whenever RHR is in the LPCI mode, inject through the heat exchangers as soon as possible.