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SUBJECT: Forwards revised see rpt response to Generic Ltr 83-28, "Required Actions Based on Generic Implicaitons of Salem ATWS Events," reflecting updates to Items 1.2.2,2.1,2.2.2 & 4.5.

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MAR 0 1 1984

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

SUSQUEHANNA STEAM ELECTRIC STATION
REVISED RESPONSE TO GENERIC LETTER 83-28
ERS 100450/100508 FILE 841-2
PLA-2095

Docket Nos. 50-387
50-388

Dear Mr. Eisenhut:

Our original response to Generic Letter 83-28, "Required Actions Based on Generic Implications of Salem ATWS Events," was submitted by PLA-1827 on November 4, 1983. Attached is a revision to that response reflecting updates to items 1.2.2, 2.1, 2.2.2 and 4.5 including the output of the BWR Owners' Group and the INPO-NUTAC. Changes to the earlier version are indicated by vertical lines in the margin.

This represents our final response to the Generic Letter except for item 4.5.3, review of reactor trip system on-line functional test intervals required by Technical Specifications. As indicated in the attached, PP&L is participating in a generic effort by the BWR Owners' Group Technical Specifications Committee to develop the methodology for a quantitative review of Technical Specifications surveillance requirements. The Owners' Group has recently awarded a contract to accomplish this. Upon completion of the Owners' Group effort, the methodology will be applied to review SSES Technical Specification required intervals. The results of that evaluation will be forwarded when available.

Very truly yours,

NW Curtis

N. W. Curtis
Vice President-Engineering & Construction-Nuclear

Attachment

cc: R. L. Perch - NRC
R. H. Jacobs - NRC Resident Inspector

A055
1/1

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA)
: SS
COUNTY OF LEHIGH)

I, NORMAN W. CURTIS, being duly sworn according to law, state that I am Vice President, Engineering & Construction-Nuclear of Pennsylvania Power & Light Company and that the facts set forth on the attached response by Applicants to Generic Letter 83-28 are true and correct to the best of my knowledge, information and belief.

N W Curtis

Norman W. Curtis
Vice President,
Engineering & Construction-Nuclear

Sworn to and subscribed before me this 29th day of February, 1984

Martha C Barto

Notary Public
MARTHA C. BARTO, Notary Public
Allentown, Lehigh County, Pa.
My Commission Expires Jan. 13, 1986

Pennsylvania Power & Light Company

Susquehanna Steam Electric Station
Units 1 & 2

Response to Generic Letter 83-28
November 6, 1984

Revised February 29, 1984

Section 1: Post-Trip Review

1.1 Program Description and Procedure

PP&L has developed a systematic program for ensuring that unscheduled reactor scrams are thoroughly evaluated. This program is implemented by plant procedures.

PP&L's Nuclear Safety Assessment Group (NSAG) recently completed an independent review of the scrams to date at SSES. NSAG attempted to determine if each trip was analyzed in sufficient detail to justify the conclusion that the reactor could be safely restarted. NSAG concluded that in each case the decision to restart the reactor was justified . That is, sufficient facts were in hand and sufficient analysis had been done to ensure safety.

The following respond to the respective requirements of Generic Letter 83-28 section 1. Susquehanna SES is in full compliance with the NRC position and no changes are planned.

1.1.1. The general criterion for determining the acceptability for restart is:

- The plant shall be able to be restarted and operated safely.

To implement this general criterion, we impose the following specific criteria:

a) The trip parameter that scrammed the reactor shall be determined.

b) The system or component that is the source of the abnormality causing the trip shall be determined and corrective actions accomplished, as required:

   1) If the suspect system or component is safety-related, the cause of the malfunction shall be identified and corrective actions completed prior to restart.

   2) If the suspect system or component is not safety-related, the plant may be restarted if it is confirmed that the plant is within technical specifications.

c) The correct functioning of safety-related systems and components during the event shall be verified by:

   1) Verifying the proper sequence of safety-related system/component actuation from detection/initiation to shutdown.

2) Comparing the event information with known or expected behavior.

1.1.2. The responsibilities and authorities of personnel who will perform the review and analysis of these events are as follows:

a) The Superintendent of Plant or Duty Manager is responsible for authorizing the Shift Supervisor to re-start the plant following a scram.

b) Section Heads are responsible for resolving items identified in the post scram critiques. They shall ensure resolution of items which are identified as "required for unit startup" are dispositioned prior to Duty Manager authorization of a plant startup.

c) The Shift Technical Advisor is responsible to provide post scram data and preliminary evaluations of plant performance including whether equipment functioned as designed.

d) The Supervisor, Scram Investigations is responsible for supervising and coordinating the investigation and immediate corrective actions following reactor scrams.

e) The Senior Compliance Engineer is responsible to verify appropriate notifications were made within the designated times based on appropriate incident classification by on shift personnel.

1.1.3. The necessary qualifications and training for the responsible personnel are:

a) Superintendent of Plant - required to meet the criteria of section 4 of ANSI/ANS-3.1-1978 for Plant Manager.

b) Duty Manager - required to meet the criteria of section 4.2 of ANSI/ANS-3.1-1978 for managers.

c) Section Heads- Senior Plant Staff personnel who as a minimum, meet the requirements of ANS-3.1-1978 for their respective positions.

d) STA - Minimum qualifications:

- B.S. or equivalent in Engineering or Related Sciences.
- Four years power plant experience of which three or more are nuclear power plant experience.

- Meet requirements for personnel whose duties require the use of self-contained respirators

- Ability to meet requirements for access to vital areas of Susquehanna SES.

- Ability to successfully complete the required training and certification which includes:

  - STA Systems Training - consisting of lectures, quizzes and examinations covering system design bases, flow paths, components, and instrumentation and controls.

  - STA Science - consisting of reactor theory, thermodynamics, heat transfer, fluid flow, BWR chemistry, reactor plant materials, radiation protection, electrical science, and instrumentation.

  - STA Simulator - consisting of technical specifications, operating procedures, emergency procedures, and transient analysis.

  - Mitigating core damage training.

  - Emergency response team training

  - Management Training - consisting of problem analysis, decision making, and communication.

e) Supervisor, Scram Investigations - These are senior plant staff personnel who as a minimum, meet the requirements of ANS 3.1-1978 for their respective positions and is appointed by the plant superintendent.

f) Senior Compliance Engineer - Required to meet the criteria similar to Section 4.4 of ANS-3.1-1978.

1.1.4. The normal sources of plant information necessary to conduct the review and analysis are the General Electric Transient Analysis and Recording System (GETARS), the plant process computer, and chart recorders located in the control room and throughout the plant. Section 1.2 "Post-Trip Review-Data and Information Capability" contains the detailed capabilities of these sources.

1.1.5. When applicable, the plant performance is evaluated against the transients in the FSAR. In addition, the Post Transient/Reactor Scram Evaluation procedure has a safety systems actuations and

performance table which is used to compare actual to expected plant behavior.

1.1.6. Plant Operating Review Committee meetings are conducted to review all scrams from critical and recommend startup to the Plant Superintendent. A Technical Section Instruction, SCRAM Report Data Collection, provides guidelines for the proper retrieval of data for analysis and evaluation. This data is stored in the Document Control Center as part of the permanent plant records.

1.2 Data and Information Capability

Safety-related systems provide actions necessary to assure safe shutdown, to protect the integrity of radioactive material barriers, and/or to prevent the release of radioactive material in excess of allowable dose limits (SSES FSAR 7.1.1a.3.1). Recording, recalling and displaying data to permit diagnosing the causes of reactor shutdowns and for ascertaining the functioning of safety-related equipment are clearly not safety-related functions. Acquiring and storing this data is necessary for the proper and efficient operation of a nuclear power plant. As such, SSES is equipped with systems which allow for the acquisition, processing, recording and display of data for timely diagnosis and correction of conditions rendering the plant inoperable. These systems are more than adequate to perform the functions for which they were intended.

The following respond to the respective requirements of generic letter 83-28 section 1.2.

1.2.1. Capability for assessing sequence of events

The primary mechanism for assessing the sequence of events prior to and during an unscheduled reactor shutdown is the plant computer. An overview of the SSES plant computer is in FSAR Section 7.7.1.7. The computer is equipped to produce two post-trip review logs and a sequence of events log (SOE). Taken together, these logs allow for the detailed assessment of the sequence of events just prior to and during an unplanned reactor shutdown.

The plant computer is programmed to produce both a nuclear steam supply system (NSSS) post-trip review log and a balance of plant (BOP) post-trip review log. The NSSS post-trip review log is automatically activated five minutes after a scram. It provides a log of 24 NSSS data point values (listed in Attachment 1) collected at 5 second intervals for the 5 minute period before and after a scram event. The log is printed on the logging line printer using the format shown in Attachment 2.

The BOP post-trip review log is similar to that for the NSSS except that 48 BOP data points are monitored and logged (see Attachment 3). The log provides the data collected at 15-second intervals for the 30 minutes prior to and following a trip. The log is either automatically initiated 30 minutes after the trip or can be demanded by the operator and is typed on the logging line printer using the same format as in Attachment 2.

The SOE log provides a chronological listing of major events occurring during plant operations. Upon detection of a status change of any of the preselected sequential events contacts, the

SOE log initiates. As soon as 64 contact changes have been sensed or 30 seconds have elapsed since the first detected change, the log begins printing on the log line printer in the Control Room. The log continues until all contact changes have been printed including any changes sensed subsequent to log initiation. The system has the capability of establishing an order of occurrence between state changes with a resolution of 4 milliseconds. Attachment 4 is the list of parameters which are inputs to the SOE log and Attachment 5 is an example of the output format.

All three logs produce hard copy records which are retained in the SSES Records Management System.

The plant computer is powered from an uninterruptable non-1E power supply backed up by an engineered safeguard supply (FSAR 8.3.1.8).

An alternative mechanism for assessing the sequence of events in case of a loss of the computer is to take data from the strip chart recorders in the control room and in the plant (see next section) and supplement this data with operation observations.

1.2.2. 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

The primary mechanisms for assessing the time history of analog variables at SSES are the historical recording capability of the plant computer system, control room strip chart recorders and the General Electric Transient Analysis and Recording System (GETARS).

The plant computer has a historical recording function which records certain preselected changes in input signals from both NSSS and BOP systems. Changes to the software which controls what data is placed in historical records is under plant administrative control. Data is temporarily stored on a drum and transferred to magnetic tape for permanent retention each time the assigned drum area is filled. Data may later be retrieved from the magnetic tape and analyzed. Attachment 6 is an example output.

Strip chart recorders in the control room can be used to assess the functioning of safety-related equipment. With few exceptions, these recorders run at 1 inch per hour. Strip charts provide a continuous indication of analog parameters.

Section 1: Post-Trip Review

GETARS is composed of a mini-computer, magnetic tape unit, magnetic disk drive unit and a printing/plotting station that are integrated by software. The system is capable of handling inputs from more than five hundred channels. These channels may then be monitored and/or recorded.

The channels that are selected for recording are placed in what is called a "work file" created by utilizing system software. The number of points in this file and the frequency of scanning is a balance between the scanning rate of the equipment, the storage capability of the equipment, the desired discrimination time between scans and the desired time following an event that is desired. At the present time we scan approximately 300 channels once every 6.67 milliseconds (see Attachment 8 for a listing of these points).

The recording is triggered by selected points exceeding a predetermined set point. The triggers presently in use are shown in Attachment 9. On triggering, the items selected to be in the work file are recorded for a predetermined time prior to the trigger until the recording capability is filled. This recording time period is dependent on the number of channels in the work file and the selected scan rate.

The information that is stored is in both digital and analog form depending on the specific parameter. These may be plotted at a later time to aid in accident/transient analysis.

GETARS is powered from an uninterruptable non-1E power source that is capable of being powered from the plant diesel generators. This is considered a highly reliable source of power.

Attachment 7 is a list of parameters developed for this response which are required to determine if safety-related systems functioned properly to meet the criteria of 1.1.1 above. Further evaluation of Attachment 7 since our original submittal permits us to conclude that the list is adequate to satisfy the criteria of 1.1.1. Many more parameters, identified in the attachments, are monitored and recorded at SSES than are required to determine the acceptability for restart.

One additional point should be made with regard to ascertaining the proper functioning of safety-related equipment by post-trip reviews. SSES, as do all licensed reactors subject to technical specifications, has a comprehensive surveillance and in-service inspection program. The purpose of this program is to assure the continued high reliability and operability of safety-related equipment. Ascertaining the proper functioning of

safety-related equipment during post-trip reviews is an informative supplement to the current program. However, PP&L continues to rely on the existing surveillance and in-service inspection program to assure the proper functioning of safety-related equipment.

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

Other data and information available to assess the cause of unscheduled reactor shutdowns include: plant instrumentation associated with nonsafety-related systems; data obtained from analysis of samples; data from further engineering analysis; and, in extreme cases, information from disassembly and examination of equipment. Together, all of the above information allows us to determine the ultimate cause for rendering the plant inoperable.

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

The SSES capability to record, recall and display data and information to permit diagnosing the causes of unscheduled reactor shutdowns prior to restart and for ascertaining the proper functioning of safety-related equipment is in full compliance with the NRC position. No changes are planned.

Section 2 - Equipment Classification and Vendor Interface

2.1 Reactor Trip System Components

Equipment Classification

All reactor trip system components 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. No changes are planned.

Vendor Interface

The Susquehanna SES reactor protection (trip) system was supplied by the General Electric Company. General Electric is still under contract to PP&L for the completion of Unit 2. This contract provides for current vendor information through contract termination.

Following contract termination, the existing vendor equipment information program with General Electric ensures that vendor information for the reactor protection (trip) system is current and complete. This program consists of two major categories: (1) information regarding safety-related systems and components; and, (2) technical information intended to enhance equipment reliability and improve plant performance. These programs include, but are not limited to:

(1) 10CFR21 Reporting - 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 are normally preceded or followed by telephone calls to assure receipt of the information.

(3) Service Information Letters (SILs) - These documents are usually brief, providing recommendations for equipment modification, plant design improvements or changes to procedures to improve plant performance.

(4) Service Advice Letters (SAL) - These documents are issued by GE Product Departments other than the San Jose based Nuclear Energy Product Departments and are used to provide notification of product problems and/or service information on a broad range of GE consumer and industrial products. Those Service Advice Letters that are

recognized by the issuing product department as applying to devices used in nuclear plants are specially identified and are flagged for distribution to all nuclear plants.

(5) 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.

The programs for SILs and TILs include a mechanism to ensure our receipt of the GE information.

PP&L has an existing Industry Events Review Program (IERP) to review relevant industry information and experience, including that provided by GE, for impact on SSES. This program is described in section 2.2.2. Technical information not reviewed under the IERP program receives engineering review on a case by case basis.

2.2 Programs for All Safety-Related Components

2.2.1. Equipment Classification

PP&L has a comprehensive program for ensuring that safety-related systems, components and structures are classified and maintained as such. The PP&L program is implemented through a thorough set of procedures containing the necessary ingredients to assure that safety-related quality is maintained throughout the life of SSES. The procedures define the method of determining the classification of equipment, and requirements for design, modification, procurement, maintenance, handling, storing, and testing safety-related equipment.

SAFETY RELATED is a generic term applied to:

1. Those systems, structures, and components that meet one or more of the following requirements:

   (a) Maintain the integrity of the Reactor Coolant System pressure boundary.

   (b) Assure the capability to prevent or mitigate the consequences of accidents that could cause the release of radioactivity in excess of 10CFR100 limits.

   (c) Preclude failures which could cause or increase the severity of postulated accidents or could cause undue risk to the health and safety of the public due to the release of radioactive material.

(d) Provide for safe reactor shutdown and immediate or long term post accident control.

2. Those activities that affect the systems, structures and components discussed in Item 1 above such as their design, procurement, construction, operation, refueling, maintenance, modification and testing.

PP&L maintains a controlled "Q-list" identifying systems, structures and components which have been designated as safety-related. The content of this list is based on Table 3.2.2 of the FSAR. The SSES Q-list does not include every safety-related component of every safety-related system. However, as discussed below, the PP&L procedures for determining unlisted component and part quality are very conservative. As such, this system complies fully with the intent of NRC's position.

The Q-list is one of PP&L's Quality Consideration Lists (QCL). Quality systems, components and structures are defined as those which appear on any of the QCL. The QCL are controlled documents which take precedence over other lists or documents as the source of quality classification. Development and updating of the QCL are the responsibility of the Nuclear Plant Engineering group. The initial QCL and requests for changes thereto are required to be reviewed by quality assurance.

PP&L procedures for modifications, engineering, maintenance, procurement and material control require a determination if the activity affects a quality system, component or structure. For procurement of spare parts, parts used in safety-related (Q-listed) assemblies or systems are classified as either Q1, Q2, Q1E, Q2E or N as defined below:

"Q1" Items - A sub-classification of "Q" - These items are considered essential to the safe operation of the plant and are designed and fabricated in accordance with a QA program approved by PP&L that is consistent with the pertinent provisions of 10CFR50, Appendix B, National Standards or Codes, as appropriate, and other requirements set forth in the purchase order. The supplier shall provide, as a minimum, documentation for all specified requirements and a certificate of compliance certifying that all Q1 items comply with the purchase order requirements in order for the parts to be considered acceptable.

"Q2" Items - A sub-classification of "Q" - These items are considered essential to the safe operation of the plant and may be obtained commercially as catalog/off-the-shelf items (i.e., they do not require unique or special engineering

specifications; or they are manufactured to national standards or by processes generally automated or highly repetitive; or there is little chance for error during manufacture to affect their safety-related characteristics; or they may be obtained commercially in accordance with Article IWA-7000 of the ASME Code). The manufacturer of Q2 items need not have a QA program meeting the requirements of 10CFR50, Appendix B, but should have the usual processes or quality controls normally associated with the production of like items.

"Q1E" - A sub-classification of "Q" - These Class 1E items are classified as Q1 items and are essential to emergency reactor shutdown, containment isolation, reactor core cooling, and containment and reactor heat removal, or otherwise essential in preventing significant release of radioactive material to the environment and are environmentally qualified in accordance with the requirements of NUREG 0588, Category I, and IEEE-323-74 as identified by SSES "Environmental Qualification Report for Class 1E Equipment." The supplier shall provide, as a minimum, the documentation and certificate of compliance consistent with all Q1 items. In addition, for items classified as 1E, the supplier will be required to submit:

1. An Environmental Qualification Test Plan including test set-up, test procedures, and acceptance criteria for at least one of each type of part.

2. An Environmental Test Report, with data, that demonstrates parts qualification in accordance with the Environmental Qualification Test Plan.

"Q2E" Items - A sub-classification of "Q" - These Class 1E items are classified Q2 items and are essential to emergency reactor shutdown, containment isolation, reactor core cooling, and containment and reactor heat removal or otherwise essential in preventing significant release of radioactive material to the environment and are environmentally qualified in accordance with the requirements of NUREG 0588, Category I, and IEEE-323-74 as identified by SSES "Environmental Qualification Report for Class 1E Equipment." As a minimum, the documentation required of the supplier will consist of a certificate of compliance and:

1. An Environmental Qualification Test Plan, including test set-up, test procedures and acceptance criteria for at least one of each type of part.

2. An Environmental Test Report, with data, that demonstrates parts qualification in accordance with the Environmental Qualification Test Plan.

Section 2 - Equipment Classification and Vendor Interface

"N" Items - Those items not within the scope of the PP&L QA program.

Attachment 10 shows the logic followed to classify these parts.

Upon receipt of quality parts, PP&L material control procedures require that acceptance be documented and that the part be clearly identified by means of a color coded tag. The tag remains with the part through handling, storage and installation.

For other than procurement of spare parts, parts or components of safety-related systems are all considered safety-related.

All corrective maintenance and implementation of modifications on all plant structures, systems and components is controlled by a work authorization system. Before physical work is initiated, the work authorization system requires that the work group foreman classify the work authorization as quality or not and ASME code related or not. The work activity classification is conservatively based on the system, structure and major equipment level, rather than on the classification of individual components. This is an important and deliberate feature of the system for classifying work activities intended to assure that work activities which may affect safety-related systems, structures, or equipment are appropriately conducted and controlled even though the objective of the work activity may be maintenance, repair, replacement, etc. of a non-safety component.

The PP&L Operational Quality Assurance Program (FSAR 17.2) applies to all safety-related SSES structures, systems, components and activities. This program provides the necessary management controls, including periodic reviews and audits, to verify that covered procedures are followed. The procedures for preparation, validation and routine utilization of the information handling systems described, including the QCL, are subject to the controls of the PP&L Operational Quality Assurance Program.

Demonstration that appropriate design verification and qualification testing is specified for procurement of safety-related components requires consideration of the status of the component being procured. For procurement of the types of safety-related components acquired during construction, demonstration is provided by the Equipment Qualification Document Files which have been audited by the NRC and found acceptable (NUREG-0776, Supplement 3, p3-7).

Section 2 - Equipment Classification and Vendor Interface

Design verification and qualification testing is not required for procurement of replacement parts and components identical to or equivalent to items already qualified.

For procurement of replacement parts/components for which an alternative to an identical or equivalent is selected, appropriate design verification and qualification testing is required. Procurement specifications for such equipment include design control requirements which comply with ANSI N.45.2.11-1974, including generation and approval of inputs and verification by independent review.

Attachment 11 is taken from a specification for new containment structure electrical cable penetrations prepared by our architect engineer. This specification includes requirements for qualification to harsh environmental conditions and appropriate qualification testing. To date, PP&L has not had occasion to directly procure equipment requiring environmental qualification for a harsh environment.

As an example of a direct PP&L procurement, attachment 12 is taken from a procurement specification for an additional diesel generator. This specification includes designation of mild environment conditions appropriate for this component and requires a certification of compliance. Also, dynamic load qualification requirements are included in this example.

With respect to the equipment classification program in use at SSES for structures, systems and components important to safety, we are participating in the Utility Safety Classification Group and are seeking a generic resolution to the staff's concern in this regard through the efforts of this Group. We do not agree that plant structures and components important to safety constitute a broader class than the safety related set. We nevertheless believe that non safety related plant structures, systems and components have been designed, and are maintained, in a manner commensurate with their importance to the safety and operation of the plant.

The procedures and specifications described above form a system which controls activities affecting safety-related components and which complies with NRC's position. No changes are planned in this area.

2.2.2. Vendor Interface

Vendor information for safety-related components is maintained current and complete at SSES through an Industry Events Review Program, procedures for Installation, Operating and Maintenance Manual control, and the industry wide Vendor Equipment Technical Information.

PP&L has an existing Industry Events Review Program to review relevant industry information and experience for impact on SSES. This program provides for the review of:

- INPO Significant Operating Experience Reports (SOER).
- General Electric Service Information Letters (SIL), and Technical Information Letters (TIL).
- INPO Significant Event Reports (SER).
- SSES Incident and Event Reports.
- Other industry information such as Atomic Energy Clearing House, EPRI Reports, NSAC Reports, IE Info Notices, etc.

The implementation of the action items identified by this program incorporates the lessons learned from industry experience in PP&L practices.

The original set of SSES Installation, Operating and Maintenance Manuals (IOM's) were obtained and reviewed in accordance with administrative and engineering procedures implementing the A/E's Quality Assurance Program which was reviewed and approved by PP&L and subject to audits and management reviews. Revisions to the IOM's were controlled by these same procedures up to the time of turnover of control to PP&L.

New IOM's, or revisions to IOM's are directed to the Document Control Center (DCC) at SSES. DCC does a preliminary administrative review.

A technical review is also performed by individuals which meet the following requirements:

o Have a minimum of five (5) years experience in the appropriate discipline, e.g., Instrumentation and Controls, Chemistry and Radiochemistry, Radiation Protection, Electrical Equipment, and Mechanical Equipment.

- o One of the five years shall be nuclear power plant experience in the appropriate discipline.
- o Two of the five years should be related technical training.
- o A maximum of four of the five years may be fulfilled by related technical or academic training.

After a review is conducted to determine that the IOM is appropriate for the applicable equipment and approved by the technical reviewer's supervision, the IOM or IOM revision is issued, as follows:

- o DCC logs review as complete and files original review sheet.
- o Appropriate number of copies of the IOM Review Form are made.
- o A copy of the review form shall be inserted in front of the applicable IOM(s). This certifies the revision, manual or correction as approved.
- o DCC makes the appropriate controlled distribution.
- o DCC updates the Index of Approved IOM's by entering the appropriate manual identification and date of approval into the index.

Distribution of IOM's is controlled. IOM's are changed by issuing a properly reviewed and approved Drawing Change Notice.

PP&L has actively participated in the "NUTAC on NRC Generic Letter 83-28, Item 2.2.2." PP&L endorses the Vendor Equipment Technical Information Program (VETIP) concept proposed by the NUTAC. The NUTAC report is scheduled to be published in March. When the report is available the administrative program and procedures now in effect at PP&L will be reviewed against the recommended guidelines presented in the NUTAC Report. All variances will be analyzed and where appropriate procedures modified. Our goal for completion of this activity is January 1, 1985.

Section 3 - Post-Maintenance Testing

3.1 Reactor Trip System Components

3.1.1 A review was made 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 performed as required to demonstrate the equipment is capable of performing its safety function. No problems were identified.

A Maintenance Planners Guide was developed to provide clarification for the planners in assuring that all post-maintenance testing required by Technical Specifications is identified. In addition to the testing required by Tech Specs, the maintenance work groups are responsible for recommending operational testing on the Equipment Release Form (ERF) for a particular job. In many cases this testing is fulfilled by performing the applicable surveillance test for a system or component.

Shift supervision makes the final determination on the operational testing required to restore the system or component to operable status. We conclude that this system complies with NRC's position and plan no changes in this area.

3.1.2 Appropriate vendor and engineering recommendations have been incorporated in SSES test and maintenance procedures and Tech Specs for the reactor trip system. The process for ensuring that recommendations are incorporated is the same as for all other safety-related components described in section 3.2.2.

3.1.3 To date, we have identified no instances where post-maintenance testing required by Tech Specs degrades rather than enhances safety. However, we intend to continue to pursue this subject as addressed under item 3.2.3.

3.2 All Other Safety Related Components

3.2.1 A review of the SSES test and maintenance procedures and Technical Specifications has been completed. SSES procedures require post-maintenance functional testing and a review for and performance of appropriate operability testing for safety-related components. The required testing demonstrates that the equipment is capable of performing its safety functions before being returned to service.

Based on our review, we are in full compliance with the NRC position. No changes are planned.

3.2.2 Appropriate vendor and engineering recommendations have been reviewed for incorporation into SSES test and maintenance

procedures and Tech Specs for all safety-related equipment. Incorporation is accomplished by appropriate handling of vendor supplied Installation and Operating Manual (IOM) information and vendor bulletins such as described in section 2.2 above.

The plant work authorization system includes identification of required instructions and procedures in planning of work activities. Each work authorization includes instructions written specifically for it. Appropriate portions of IOM's are incorporated verbatim, paraphrased, or by reference as appropriate. The work authorization is subject to review and approval before use.

3.2.3 To date, we have identified no instances where post-maintenance testing required by Technical Specifications degrades rather than enhances safety. However, we share a BWR industry wide concern that the current frequency of surveillance tests required by some technical specifications may degrade rather than enhance safety. We have joined with other industry members in the formation of a Technical Specification Review Committee as part of the BWR Owners Group. We intend to pursue the revision of tech spec requirements which may be shown to degrade safety through this committee.

One example of a tech spec requirement that has the potential to degrade rather than enhance safety is the LCO Action Statements for tech spec 3.8.1, AC Power Systems. The Action Statements require the operable diesels to be started within 2-4 hours of entering the LCO's and once per 8 hours thereafter. The continual starting and shutting down of the diesels has a potential to reduce their reliability by continually subjecting them to transient conditions. We offer this only as an example. We will pursue justified revisions to tech spec requirements through the BWR Owners Group.

Section 4 - Reactor Trip System Reliability Improvements

4.1 Vendor Related Modification

Not applicable to SSES

4.2 Preventive Maintenance and Surveillance Program for Reactor Trip Breakers

Not applicable to SSES

4.3 Automatic Actuation of Shunt Trip Attachment for Westinghouse and B&W Plants

Not applicable to SSES

4.4 Improvements in Maintenance and Test Procedures for B&W Plants

Not applicable to SSES

4.5 System Functional Testing

4.5.1 and 4.5.2 The SSES reactor protection (trip) system includes the motor-generator power supplies, sensors, relays, bypass circuitry, and switches that cause rapid insertion of control rods (scram) to shutdown the reactor (FSAR 7.2.1.1.1). SSES Tech Specs (NUREG 0931) 4.3.1.1, 4.3.1.2 and 4.3.1.3 require periodic functional tests of each instrumentation channel, functional tests of the logic system and demonstration of the system response time respectively. Testing of the system response time includes de-energization of the scram pilot valves. Taken together, these tech specs require, and SSES performs, on-line functional testing of the reactor trip system, including independent testing of the scram pilot valves.

Generic Letter 83-28 section 4.5 also recommends on-line functional testing of the backup scram valves at GE plants. The backup scram valves are implied to be "diverse trip features" comparable to the breaker shunt trip features on other plants. As on-line functional testing of other plant diverse trip features is required by the Generic Letter, these requirements are also extended to the backup scram valves on GE plants. The difference between the GE reactor trip system and the one that initiated the Generic Letter makes this extension of on-line functional testing unwarranted.

Each Susquehanna SES unit has 185 control rods. Each control rod is activated by an independent scram pilot valve. Reference 1 concluded that reactor shutdown can be achieved if at least 50% of the control rods in checkerboard pattern or 69% in a random pattern are inserted in the core. Clearly, only a fraction of these 185 control rods must successfully function to shutdown the

reactor. The probability of independent failure of enough rods to prevent shutdown is negligible.

Two redundant backup scram valves are provided in GE plants to assure that the control rods do actuate should any of the pilot scram valves fail to function. No explicit credit is taken for these valves in plant safety analyses or system reliability analyses nor are they required by applicable regulatory requirements. Functional testing of these valves during plant operation would require a plant scram, a significant challenge to plant safety systems and therefore a degradation of plant safety. The backup scram valves are non-safety related additions that can only enhance the reliability of the safety-related reactor trip system.

In contrast, other designs include only two redundant reactor trip breakers, one of which is required to successfully function to scram the reactor. Each of these breakers has an undervoltage actuation device and a diverse, functionally redundant shunt actuation device. The successful function of this system requires the operation of 1 of these 4 actuation devices. In the event of an initiating event (loss of AC power) followed by single active failure (breaker failure), the shunt devices are rendered useless, therefore successful system operation depends on breaker operation by the remaining actuation device. Obviously, improper functioning of the diverse, safety-related trip devices on other plants would degrade the reliability of the safety-related reactor trip system.

In summary, the functions performed by and the safety-related reliability dependence on shunt actuation devices in one design are considerably different from the reliability enhancement afforded by backup scram valves in the GE design. Similarly, testing requirements on shunt actuation devices should not necessarily be requirements for backup scram valves. We therefore request exemption from this requirement for Susquehanna SES.

4.5.3 PP&L is participating in the BWR Owners' Group Technical Specification Improvements Committee program. This program includes development of a methodology for application to the review of intervals for on-line functional testing required by Technical Specifications. A generic methodology will be developed to show the sensitivity of system unavailability to changes in

a. Component failure rates
b. Common mode failure rates
c. Reduced redundancy during testing
d. Human error rates during testing, and
e. Component "wear-out" rates caused by testing.

PP&L will then utilize the results for specific application to Susquehanna SES. The schedule for the above generic approach is currently being prepared by the Technical Specification Improvements Committee of the BWR Owners' Group.

Reference: 1) "BWR Scram System Reliability Analysis", W. P. Sullivan, et al, September 30, 1976 (transmitted in a letter from E. A. Hughes (GE) to D. F. Ross (NRC), "General Electric Company ATWS Reliability Report", September 30, 1976).

rlw/msb200258o

Attachment 1

NSSS Post-trip Review Log Data Points

Data Point	Description
MAP03	Contnm. Press. A LOCA Range
NFL01	Rx. Narrow Range Lvl. A
NFL04	Rx. Upset Range Lvl.
NAR01	Mn. Steam Line Rad. A
NAR05	Offgas Pretreat Rad. A
NFP02	Rx. Press. Wide Range
NM501	APRM Flux Pwr. Lvl. A Chan.
NM503	APRM Flux Pwr. Lvl. C Chan.
NM505	APRM Flux Pwr. Lvl. E Chan.
NN101	SRM A Log Count Rate
NN103	SRM C Log Count Rate
NN109	IRM A Flux Reading
NN111	IRM C Flux Reading
NN113	IRM E Flux Reading
NN115	IRM G Flux Reading
NFF08	Mn. Turb. Steam Flow
NFF05	FW Flow A
NFF06	FW Flow B
NFF07	FW Flow C
NFF01	Rx. Steam Flow A
NFF02	Rx. Steam Flow B
NFF03	Rx. Steam Flow C
NFF04	Rx. Steam Flow D
NLT01	Rx. Bottom Head Drn. Temp.

DATE MO/DY/YR TIME 10:30:10

NSS POST TRIP REVIEW LOG

PRE-TRIP DATA

	PPPPPPPP	PPPPPPPP	PPPPPPPP	PPPPPPPP	PPPPPPPP	PPPPPPPP	PPPPPPPP	PPPPPPPP	PPPPPPPP	PPPPPPPP	PPPPPPPP	PPPPPPPP
IR:												
10:20:10	SXXXX.XXA	SXXXX.XXA	SXXXX.XXA	SXXXX.XXA	SXXXX.XXA	SXXXX.XXA	SXXXX.XXA	SXXXX.XXA	SXXXX.XXA	SXXXX.XXA	SXXXX.XXA	SXXXX.XXA
10:20:15	SXXXX.XXA	SXXXX.XXA	SXXXX.XXA	SXXXX.XXA	SXXXX.XXA	SXXXX.XXA	SXXXX.XXA	SXXXX.XXA	SXXXX.XXA	SXXXX.XXA	SXXXX.XXA	SXXXX.XXA
10:20:20												
10:20:25												
10:20:30												

POST-TRIP DATA

	PPPPPPPP	PPPPPPPP	PPPPPPPP	PPPPPPPP	PPPPPPPP	PPPPPPPP	PPPPPPPP	PPPPPPPP	PPPPPPPP	PPPPPPPP	PPPPPPPP	PPPPPPPP
10:25:10	SXXXX.XXA	SXXXX.XXA	SXXXX.XXA	SXXXX.XXA	SXXXX.XXA	SXXXX.XXA	SXXXX.XXA	SXXXX.XXA	SXXXX.XXA	SXXXX.XXA	SXXXX.XXA	SXXXX.XXA
10:25:15	SXXXX.XXA	SXXXX.XXA	SXXXX.XXA	SXXXX.XXA	SXXXX.XXA	SXXXX.XXA	SXXXX.XXA	SXXXX.XXA	SXXXX.XXA	SXXXX.XXA	SXXXX.XXA	SXXXX.XXA
10:25:20												
10:25:25												
10:25:30												

DATA RECORDED FOR 24 POINTS AT 5-SECOND INTERVALS FOR 5 MINUTES

NOTE: BOP Post Trip Review Log is same as NSS Log except, BOP log data is recorded for 48 points at 15 second intervals for 30 minutes before and 30 minutes after trip.

ATTACHMENT 2

Attachment 3

BOP Post-trip Review Log Data Points

Data Point	Description	Data Point	Description
GNE02	Gen. Phase A-B Voltage	THV09	Gen. BRG. #9 Vibration
GNE03	Gen. Phase B-C Voltage	THV10	Gen. BRG. #10 Vibration
GNE04	Gen. Phase C-A Voltage	THV11	Alt. BRG. #11 Vibration
GNJ01	Generator Power	THV12	Alt. BRG. #12 Vibration
GNU02	Gen. Reactive Power	TLP01	Turb. L-O Brg. HDR Press.
GNU03	Gen. Frequency	TLP02	Turb. L-O OPER Oil Press.
GST02	Gen. Stator Clg. Wtr. Out	TLP03	Turb. HYD Fluid Press.
TAP03	Turb. Seal Stm. Press.	TLT15	Turb L-O CLR. in Temp.
TCY01	Turb. CV Position	TLT16	Turb L-O CLR. DSCH. Temp.
TCY02	Turb. BYP. Vlv. Position	TNP01	Condenser A Press.
TEL01	MSEP A Drain Tank Level	TNP02	Condenser B Press.
TEL02	MSEP B Drain Tank Level	TNP03	Condenser C Press.
TET07	LPT A Exhaust Hood Temp.	YBB01	Sync. BKR. 230 KV Grid
TET08	LPT B Exhaust Hood Temp.	THX02	Turb. Diff. Expansion.
TET09	LPT C Exhaust Hood Temp.	THX03	Turb. Shell Expansion
THS01	Turbine Speed	THX04	Turb. Rotor Expansion
THV01	Turb. BRG. #1 Vibration	TNL01	Condenser Hotwell Level
THV02	Turb. BRG. #2 Vibration	FPT01	RFP A Discharge Temp.
THV03	Turb. BRG. #3 Vibration	FPT02	RFP B Discharge Temp.
THV04	Turb. BRG. #4 Vibration	FPT03	RFP C Discharge Temp.
THV05	Turb. BRG. #5 Vibration	FTS01	RFPT A Speed
THV06	Turb. BRG. #6 Vibration	FTS02	RFPT B Speed
THV07	Turb. BRG. #7 Vibration	FTS03	RFPT C Speed
THV08	Turb. BRG. #8 Vibration	GSC02	Gen. Stator CLG. CNVTY.

SEQUENCE OF EVENTS INPUTS

POINT IDENT	ENGLISH DESCRIPTION
CPZ03	COND PP A ABNORMAL TRIP
CPZ04	COND PP B ABNORMAL TRIP
CPZ05	COND PP C ABNORMAL TRIP
CPZ06	COND PP D ABNORMAL TRIP
EBZ01	T BUS 0A106-BUS 11A BKR
EBZ02	T BUS 0A106-BUS 11B BKR
EBZ10	BUS 10-T BUS 0A106 BKR
EBZ11	T BUS 0A106-0A107 BKR
EBZ20	BUS 20-T BUS 0A107 BKR
EBZ26	FDR TO SU BUS 10 UNDERVOLT
EBZ51	T BUS 0A 107-BUS 2A BKR
EBZ52	TIE BUS 0A107-BUS 2B BKR
EBZ76	FDR TO BUS 20 UNDERVOLT
EKY01	SU XFMR 10-BUS 10 BKR
EKY02	SU XFMR 20-BUS 20 BKR
EKY07	AUX XFMR 11-BUS 11A BKR
EKY08	AUX XFMR 11-BUS 11B BKR
ETZ01	UNIT AUX XFMR DIFF
ETZ02	UNIT AUX XFMR GRD DIFF
ETZ03	UNIT AUX XFMR PHASE OC
ETZ04	UNIT AUX XFMR GRD OVOLT

SEQUENCE OF EVENTS INPUTS

POINT IDENT	ENGLISH DESCRIPTION
FTI01	RFPT A MASTER TRIP
FT103	RFPT A VACUUM TRIP
FT104	RFPT A ACT THR BRG WEAR
FT105	RFPT A INACT THR BRG
FT201	RFPT B MASTER TRIP
FT203	RFPT B VACUUM TRIP
FT204	RFPT B ACT THR BRG WEAR
FT205	RFPT B INACT THR BRG
FT301	RFPT C MASTER TRIP
FT303	RFPT C VACUUM TRIP
FT304	RFPT C ACT THR BRG WEAR
FT305	RFPT C INACT THR BRG
GEZ02	EXCITER DIFFERENTIAL
GNZ01	MAIN GEN DIFFERENTIAL
GNZ02	MAIN GEN NEUTRAL OVOLTS
GNZ03	MAIN GEN LOSS FIELD A
GNZ04	MAIN GEN UNDER FREQUENCY
GNZ05	UNIT PRI LOCKOUT A/C
GNZ07	GEN OUT OF STEP
GNZ08	MAIN GEN GRD OVERVOLTAGE
GNZ09	MAIN GEN NEUT OVOLT STRT

Attachment 4

SEQUENCE OF EVENTS INPUTS

POINT IDENT	ENGLISH DESCRIPTION
GNZ10	UNIT DIFFERENTIAL
GNZ11	MAIN GEN LOSS FIELD B
GNZ12	UNIT BKUP LKOUT B-D
GNZ15	MAIN GEN NEG SEQUENCE
GNZ18	MAIN GEN VOLT/AERTZ
GNZ20	MN GEN BKUP VOLT/HZ
GNZ22	BKUP ANTI-MOTORING RELAY
GNZ23	GEN SPAN PROTECTION A
GNZ24	GEN SPAN PROTECTION B
GNZ38	GEN LOAD UNBALANCE
GNZ44	UNIT 1 SYNC BKR LO AIR PRS
GNZ45	230 KV GEN BKR FAILURE
GNZ46	OVERSPEED PROTEC TRIPPED
GNZ47	PRIMARY ANTI-MOTOR RELAY
GOZ01	GEN MN SEAL OIL PUMP
GOZ03	GEN EMERG SEAL OIL PP
NMQ51	UPSC NEUT TRIP APRM A
NMQ52	UPSC NEUT TRIP APRM B
NMQ53	UPSC NEUT TRIP APRM C
NMQ54	UPSC NEUT TRIP APRM D
NMQ55	UPSC NEUT TRIP APRM E

SEQUENCE OF EVENTS INPUTS

POINT IDENT	ENGLISH DESCRIPTION
NMQ56	UPSC NEUT TRIP APRM F
NMQ58	UPSC THERM TRIP APRM A
NMQ59	UPSC THERM TRIP APRM B
NMQ60	UPSC THERM TRIP APRM C
NMQ61	UPSC THERM TRIP APRM D
NMQ62	UPSC THERM TRIP APRM E
NMQ63	UPSC THERM TRIP APRM F
NNQ51	IRM UPSCL TRIP CHAN A
NNQ52	IRM UPSCL TRIP CHAN E
NNQ53	IRM UPSCL TRIP CHAN C
NNQ54	IRM UPSC TRIP CHAN G
NNQ55	IRM UPSC TRIP CHAN B
NNQ56	IRM UPSC TRIP CHAN F
NNQ57	IRM UPSC TRIP CHAN D
NNQ58	IRM UPSC TRIP CHAN H
NPQ51	DSCH VOL HI LVL TRIP A
NPQ52	DSCH VOL HI LVL TRIP B
NPQ53	DSCH VOL HI LVL TRIP A
NPQ54	DSCH VOL HI LVL TRIP B
NPQ55	MSIV NOT FL OPEN TRIP A1
NPQ56	MSIV NOT FL OPEN TRIP B1

SEQUENCE OF EVENTS INPUTS

POINT IDENT	ENGLISH DESCRIPTION
NPQ57	MSIV NOT FL OPEN TRIP A2
NPQ58	MSIV NOT FL OPEN TRIP B2
NPQ59	PRI CONTN TRIP A
NPQ60	PRI CONTN TRIP B
NPQ61	PRI CONTN TRIP A
NPQ62	PRI CONTN TRIP B
NPQ63	RPV HP TRIP A
NPQ64	RPV HP TRIP B
NQP65	RPV HP TRIP A
NPQ66	RPV HP TRIP B
NPQ67	RPV LOW WTR LVL TRIP A
NPQ68	RPV LOW WTR LVL TRIP B
NPQ69	RPV LOW WTR LVL TRIP A
NPQ70	RPV LOW WTR LVL TRIP B
NPQ71	MS LINE HI RAD TRIP A
NPQ72	MS LINE HI RAD TRIP B
NPQ73	MS LINE HI RAD TRIP A
NPQ74	MS LINE HI RAD TRIP B
NPQ75	NM SYS TRIP A
NPQ76	NM SYS TRIP B
NPQ77	NM SYS TRIP A

SEQUENCE OF EVENTS INPUTS

POINT IDENT	ENGLISH DESCRIPTION
NPQ78	NM SYS TRIP B
NPQ79	MAN SCRAM TRIP A OR C
NPQ80	MAN SCRAM TRIP B OR D
NPQ81	AUTO SCRAM TRIP A OR C
NPQ82	AUTO SCRAM TRIP B OR D
NPQ83	TURB STOP VLV CLS TRIP A
NPQ84	TURB STOP VLV CLS TRIP B
NPQ85	TURB STOP VLV CLS TRIP A
NPQ86	TURB STOP VLV CLS TRIP B
NPQ87	TURB CV FAST CLS TRIP A
NPQ88	TURB CV FAST CLS TRIP B
NPQ89	TURB CV FAST CLS TRIP A
NPQ90	TURB CV FAST CLS TRIP B
NPQ91	RECIRC PUM TRIP SYS A TRIP 13
NPQ92	RECIRC PUM TRIP SYS A TRIP 13
TAZ94	VACUUM PUMP
TBZ02	TURB BYPASS VLV #1
TBZ04	TURB BYPASS VLV #2
TBZ06	TURB BYPASS VLV #3
TBZ08	TURB BYPASS VLV #4
TBZ10	TURB BYPASS VLV #5

SEQUENCE OF EVENTS INPUTS

POINT IDENT	ENGLISH DESCRIPTION
TCZ06	TURB CONTROL VALVE 1
TCZ08	TURB CONTROL VALVE 2
TCZ10	TURB CONTROL VALVE 3
TCZ12	TURB CONTROL VALVE 4
TDZ01	TURB MASTER TRIP
TDZ07	TURB OVERSPEED TRIP
TDZ15	TURB BACKUP OSPD TRIP
TDZ16	TURB EXH HOOD TEMP TRIP
TDZ17	LOSS OF STATOR CLG TRIP
TDZ18	TURB SHAFT PP DSCH PRESS
TDZ19	TURB THR WEAR OR BRG OIL
TDZ21	TURB EHC 125DC POWER
TDZ22	TURB HYD PRESS LOW TRIP
TDZ23	TURB VACUUM TRIP
TDZ24	TURB MANUAL TRIP
TDZ35	TURB VIBRATION TRIP
TDZ36	TURB EHC SPD SIGNAL LOST
TDZ40	TURB MSEP HIGH LVL TRIP
TDZ51	TURB EHC POS VOLTS LOST
TDZ52	TURB EHC NEG VOLTS LOST
TDZ55	TURB QUILL SHAFT

Attachment 4

SEQUENCE OF EVENTS INPUTS

POINT IDENT	ENGLISH DESCRIPTION
WEZ29	CIRC WTR PUMP A
WCZ30	CIRC WTR PUMP B
WCZ31	CIRC WTR PUMP C
WCZ32	CIRC WTR PUMP D
YTZ01	MN XFMR LEAD DIFF
YTZ02	MN XFMR A DIFF
YTZ03	MN XFMR A SUDDEN PRESS
YTZ04	MN XFMR B DIFF
YTZ05	MN XFMR B SUDDEN PRESS
YTZ07	SU XFMR 10 PRI LKOUT RLY
YTZ08	SU XFMR 10 PRI BKUP RLY
YTZ30	TRF 10 MTR OPER AIR BRKE
YTZ31	SU XFMR 10 HSGS 1R106
YTZ57	SU XFMR 20 PRI LKOUT RLY
YTZ58	SU XFMR 20 PRI BKUP RLY
YTZ80	TRF 20 MTR OPER AIR BRKE
YTZ81	SU XFMR 20 HSGS 2R106

Attachment 5

UNIT I PAGE 1 06-24-83 11:16

SEQUENCE OF EVENTS LOG

TIME	PT ID	NAME	STATUS
11:16:31.584	YTZ07	SU XFMR 10 PRI LKOUT RLY	STARTED
11:16:31.650	EKY01	SU XFMR 10-BUS 10 BKR	TRIPPED
11:16:31.695	YTZ08	SU XFMR 10 PRI BKUP RLY	STARTED
11:16:31.775	YTZ31	SU XFMR 10 HSGS 1R106	CLOSED
11:16:31.902	YTZ31	SU XFMR 10 HSGS 1R106	OPEN
11:16:35.696	YTZ30	TRF 10 MTR OPER AIR BRKR	OPEN
11:16:43.908	EBZ26	FDR TO SU BUS 10 UNDERVG	YES
11:16:43.975	EBZ11	T BUS 0A106-0A107 BKR	CLOSED
11:16:44. 47	EBZ26	FDR TO SU BUS 10 UNDERVG	NO
11:16:50.257	NMQ55	UPSC TRIP APRM CHAN E	YES
11:16:50.258	NMQ51	UPSC TRIP APRM CHAN A	YES
11:16:50.259	NMQ53	UPSC TRIP APRM CHAN C	YES
11:16:50.262	NPQ53	DSCH VOL HI LVL TRIP A	YES
11:16:50.262	NPQ79	MAN SCRAM TRIP A OR C	YES
11:16:50.262	NPQ73	MS LINE HI RAD TRIP A	YES
11:16:50.262	NPQ59	PRI CONTN TRIP A	YES
11:16:50.263	NPQ51	DSCH VOL HI LVL TRIP A	YES
11:16:50.263	NPQ67	RPV LOW WTR LVL TRIP A	YES
11:16:50.263	NPQ61	PRI CONTN TRIP A	YES
11:16:50.263	NPQ55	MSIV NOT FL OPEN TRIP AI	NOT OPEN

Attachment 6

UNIT 1, PAGE 1 6/14/83 13:39:12

HISTORICAL DATA RETRIEVAL AND REVIEW SERVICES(MAG TAPE I

BETWEEN 9:45 ON 6/14/83 AND 10: 0 ON 6/14/83

ALARM ACTIVITY (CONTACT)

TIME	POINT ID	POINT DESCRIPTION	STATUS	VALUE	UNITS	LMT EXC
9:46:1	NPQ05	AUTO SCRAM TRIP A OR C	OK			YES
9:46:1	NPQ06	AUTO SCRAM TRIP B OR D	OK			YES
9:46:12	NPQ58	MSIV NOT FL OPEN TRIP B2	OK			NOT OPEN
9:46:12	NPQ56	MSIV NOT FL OPEN TRIP B1	OK			NOT OPEN
9:46:14	NND10	PERIOD B NEGATIVE	OK			NO
9:46:14	NND11	PERIOD C NEGATIVE	OK			NO
9:46:16	TCZ12	TURB CONTROL VALVE 4	OK			CLOSED
9:46:16	TCZ10	TURB CONTROL VALVE 3	OK			CLOSED
9:46:16	TCZ08	TURB CONTROL VALVE 2	OK			CLOSED
9:46:16	TCZ06	TURB CONTROL VALVE 1	OK			CLOSED
9:46:16	TDZ11	TURB CV COMMAND CLOSED	OK			YES
9:46:18	MVZ33	HVAC DIV 2 COMMON SYSTEM	OK			TRBLE
9:46:18	TCD01	TURB CV #1 INTER POSN	ALM			NO
9:46:18	TCD02	TURB CV #2 INTER POSN	ALM			NO
9:46:18	TCD03	TURB CV #3 INTER POSN	ALM			NO
9:46:18	TCD04	TURB CV #4 INTER POSN	ALM			NO
9:46:18	TDD01	TURB TRIP/MODE SW RUN/SU	ALM			NO
9:46:18	TAD03	SPE B STM EXH VLV/RUN/SU	ALM			NO
9:46:18	TAD05	SPE A STM EXH VLV/RUN/SU	ALM			NO
9:46:2	FV 74	FWHTR IA SU VENT VLV	ALM			CLOSED
9:46:20	MVZ32	HVAC DIV 1 COMMON SYSTEM	OK			TRBLE
9:46:22	TEZ06	TURB STM LEAD 3 DRN VLV	ALM			NOT CLSD
9:46:22	CRD01	COND REJECT CV A INTER	ALM			NO

SUSQUEHANNA STEAM ELECTRIC STATION
SALEM ATWS EVENT
NRC GL 83-28 RESPONSE - ATTACHMENT 7
1.2: Post-Trip Review Variable List

Explanation of column entries are as follows:

NAME: Description of variable or event

USE: Principle reason for recording.
IE means to determined initiating event
SSP means to analyze safety-related system performance.

SOURCES: Devices by which data is recorded, save and/or displayed.

G is GETARS

R is Recorder (Strip chart-part of plant instrumentation). #R indicates more than one (the#) such recorders exist.

S is Sequence of Events Log

LOG is the operations (manual) log.

Post Trip Log points are operator defined and therefore not listed here. The Post Trip Log allows for 24 NSSS and 48 BOP real or pseudo analog points. Additional data is also retrievable from the historical records processing function of the Plant Computer System.

NAME	USE	SOURCES
REACTOR		
Scram Discharge Vol. Level	IE	S
Total Scram	IE	G
Channel A Scram	IE	G,S
Channel B Scram	IE	G,S
Manual Scram	IE	G,S
Total Isolation	IE	G
Channel A Isolation	IE	G
Channel B Isolation	IE	G
SRM Signals (4)	IE	R
IRM Signals (8)	IE	R,S
APRM A	IE	G,R,S
APRM B	IE	G,R,S
APRM C	IE	G,R,S
APRM D	IE	G,R,S
APRM E	IE	G,R,S
APRM F	IE	G,R,S
LPRM A	IE	G
LPRM B	IE	G
LPRM C	IE	G
LPRM D	IE	G
LPRM E	IE	G
LPRM F	IE	G
LPRM Group A	IE	G
LPRM Group B	IE	G
NM Sys Trip A	IE	S
NM Sys Trip B	IE	S
NM Sys Trip A	IE	S
NM Sys Trip B	IE	S
Flow Biased Rod Block Setpoint	IE	G
Flow Biased Scram Setpoint	IE	G
Rx Vessel Shut Down Level	IE	R
Rx Vessel Upset Level Alternate	IE	R
Reactor Water Level Setpoint	IE	G
Narrow Range Level (0 - 60)	IE	G,R
Upset (Wide) Range Level (0 - 180)	IE	G,R,S
Low Range Level (-150 to 60)	IE	G,R
Bottom Drain Temperature	IE	G,R
Narrow Range Pressure (850 - 1050)	IE	G,R
Wide Range Pressure (0-1200)	IE	G,R,S
Total Core Flow	IE	G
A Recirculation Loop Flow	IE	G
B Recirculation Loop Flow	IE	G
A Recirculation Loop Temperature	IE	G
B Recirculation Loop Temperature	IE	G
A Recirculation Pump Drive Flow	IE	G
B Recirculation Pump Drive Flow	IE	G
A Recirculation Motor Generator Set Speed	IE	G
B Recirculation Motor Generator Set Speed	IE	G
A Recirculation (RPT) Breaker	IE	G

NAME	USE	SOURCES
A Recirculation Trip #13	IE	S
B Recirculation (RPT) Breaker	IE	G
B Recirculation Pump Trip #13	IE	S

RCIC

NAME	USE	SOURCES
Reactor Core Isolation Cooling Initiation	SSP	G
Reactor Core Isolation Cooling Pump Flow	SSP	G
RCIC Turbine Speed	SSP	G
RCIC Pump Suction Pressure	SSP	G
RCIC Pump Discharge Pressure	SSP	G
RCIC Steam Supply Pressure	SSP	G
RCIC Turbine Exhaust Pressure	SSP	G
RCIC Steam Flow dp-A	SSP	G
RCIC Steam Flow dp-B	SSP	G
RCIC Trip Signal	SSP	G
RCIC Control Valve Position	SSP	G
Reactor Core Isolation Cooling Controller Output	SSP	G

HPCI

NAME	USE	SOURCES
High Pressure Coolant Injection Initiation	SSP	G
High Pressure Coolant Injection Pump Flow	SSP	G
High Pressure Coolant Injection Turbine Speed	SSP	G
HPCI Pump Discharge Pressure	SSP	G
HPCI Steam Supply Pressure	SSP	G
HPCI Turbine Exhaust Pressure	SSP	G
HPCI Steam Flow dp-A	SSP	G
HPCI Steam Flow dp-B	SSP	G
HPCI Control Valve Position	SSP	G
HPCI Vessel Injection Valve Position	SSP	G
High Pressure Coolant Injection Contr.Output	SSP	G

RHR/LPCI

NAME	USE	SOURCES
A Residual Heat Removal System Flow	SSP	G,R
B Residual Heat Removal System Flow	SSP	G,R

AUTO DEPRESSURIZATION/SAFETY RELIEF VALVES

NAME	USE	SOURCES
ADS Initiation	SSP	G
Relief Valve ABC Initiate	SSP	G
Relief Valve GJKL Initiate	SSP	G
Relief Valve DEHFP Initiate	SSP	G
Relief Valve RSMN Initiate	SSP	G
Relief Valve B Position	SSP	G
Relief Valve D Position	SSP	G
Relief Valve F Position	SSP	G
Relief Valve H Position	SSP	G
Relief Valve K Position	SSP	G

NAME	USE	SOURCES
Relief Valve L Position	SSP	G
Relief Valve N Position	SSP	G
Relief Valve R Position	SSP	G
Relief Valve A Position	SSP	G
Relief Valve C Position	SSP	G
Relief Valve E Position	SSP	G
Relief Valve G Position	SSP	G
Relief Valve J Position	SSP	G
Relief Valve M Position	SSP	G
Relief Valve P Position	SSP	G
Relief Valve S Position	SSP	G
<u>MAIN STM LINES</u>		
MSIV Initiation Signal	SSP	S
A Inboard Main Steam Isolation Vlv Position	SSP	G
B Inboard Main Steam Isolation Vlv Position	SSP	G
C Inboard Main Steam Isolation Vlv Position	SSP	G
D Inboard Main Steam Isolation Vlv Position	SSP	G
A Outboard Main Steam Isolation Vlv Position	SSP	G
B Outboard Main Steam Isolation Vlv Position	SSP	G
C Outboard Main Steam Isolation Vlv Position	SSP	G
D Outboard Main Steam Isolation Vlv Position	SSP	G
A Steam Line Flow	SSP	G
B Steam Line Flow	SSP	G
C Steam Line Flow	SSP	G
D Steam Line Flow	SSP	G
A Main Steam Line Pressure Near Relief Valves	SSP	G
B Main Steam Line Pressure Near Relief Valves	SSP	G
C Main Steam Line Pressure Near Relief Valves	SSP	G
D Main Steam Line Pressure Near Relief Valves	SSP	G
Main Stm. Line Tunnel Temp	IE	R
Main Stm. Line Tunnel Delta Temp	IE	R
Main Stm. Line Radiation	IE	R
Main Stm. Line A HI Rad.	IE	S
Main Stm. Line B HI Rad.	IE	S
Main Stm. Line C HI Rad.	IE	S
Main Stm. Line D HI Rad.	IE	S
<u>TURBINE/GENERATOR</u>		
First Stage Shell Press	IE	R
Main Stop Valve A Trip	IE	S
Main Stop Valve B Trip	IE	S
Main Stop Valve C Trip	IE	S
Main Stop Valve D Trip	IE	S
Main Control Valve A Trip	IE	S
Main Control Valve B Trip	IE	S
Main Control Valve C Trip	IE	S
Main Control Valve D Trip	IE	S
Generator Gross Megawatts Output	IE	G

NAME	USE	SOURCES
Grid Frequency	IE	G,R
Grid Voltage	IE	G,R
Main Turbine Speed	IE	G
Main Turbine Trip (Note 5)	IE	G,S
Generator Breaker Open (Note 3)	IE	G
Control Valve #1 Position	IE	G,S
Control Valve #2 Position	IE	G,S
Control Valve #3 Position	IE	G,S
Control Valve #4 Position	IE	G,S
Total Bypass Valve Position	IE	G

FEEDWATER

A Loop Feedwater Flow	IE	G,R
B Loop Feedwater Flow	IE	G,R
C Loop Feedwater Flow	IE	G,R
A Feedwater Line Temperature	IE	G,R
B Feedwater Line Temperature	IE	G,R
C Feedwater Line Temperature	IE	G,R
A Condenser Vacuum	IE	G
B Condenser Vacuum	IE	G
C Condenser Vacuum	IE	G
Feedwater Pump Suction Header Pressure	IE	G,R
A Feedwater Pump Turbine Trip (Note 2)	IE	G,S
B Feedwater Pump Turbine Trip (Note 2)	IE	G,S
C Feedwater Pump Turbine Trip (Note 2)	IE	G,S
Startup Level Controller Output	IE	G
Feedwater Turbidity	IE	R

CONDENSATE

CST Level	SSP	R
CST Temp	IE	LOG

CONTAINMENT

Sup. Pool Temp	SSP	R
Sup. Pool Level	SSP	R
Drywell Temp	SSP	R
Drywell Pressure	SSP	G,2R
Containment Area Rad. Monitor	SSP	R
Suppression Pool Activity	SSP	Note 1
Primary Containment Trip (4 Signals)	IE	S

STANDBY GAS TREATMENT

SGTS Stack Monitor Flow	SSP	R
SGTS Air Flow	SSP	R
Area Radiation Including Refueling Floor and Railroad Access Area	SSP	R

NAME	USE	SOURCES
Vent Stack Radiation including SGTS	SSP	R
<u>STANDBY LIQUID CONTROL</u>		
SBLC Flow (Tank Level & Time)	SSP	LOG
SBLC Initiation	SSP	LOG

NOTES

1. Not required as function of time. Lab. Analysis of sample.

2. Seq. of Events has four separate trip initiation inputs per turbine.

3. See Seq.of Events Inputs list for twenty-five Generator/Electrical System trip initiation inputs.

4. All five bypass valves are input to seq. of Events log.

5. See Seq. of Events log for twenty-seven individual trip initiation event inputs.

6. Also see Seq. of Events log for various electrical system trip initiation event inputs.

ad/mej057i:del

GETARS SIGNAL LIST (TMS).

CH.#	NAME	UNITS	DESCRIPTION
1	SCRAM	DIG	TOTAL SCRAM
2	SCRMA	DIG	CHANNEL 'A' SCRAM
3	SCRMB	DIG	CHANNEL 'B' SCRAM
4	MSCRM	DIG	MANUAL SCRAM
5	ISOLT	DIG	TOTAL ISOLATION
6	ISOLA	DIG	CHANNEL 'A' ISOLATION
7	ISOLB	DIG	CHANNEL 'B' ISOLATION
8	APRMA	%	APRM 'A'
9	APRMB	%	APRM 'B'
10	APRMC	%	APRM 'C'
11	APRMD	%	APRM 'D'
12	APRME	%	APRM 'E'
13	APRMF	%	APRM 'F'
14	LPRMA	%	LPRM 'A'
15	LPRMB	%	LPRM 'B'
16	LPRMC	%	LPRM 'C'
17	LPRMD	%	LPRM 'D'
18	LPRME	%	LPRM 'E'
19	LPRMF	%	LPRM 'F'
20	LPRM1	%	LPRM GROUP 'A'
21	LPRM2	%	LPRM GROUP 'B'
22	HFLX1	%	HEAT FLUX #1
23	HFLX2	%	HEAT FLUX #2
24	FBRBS	%	FLOW BIASED ROD BLOCK SETPOINT
25	FBSCM	%	FLOW BIASED SCRAM SETPOINT
26	TCFLO	MLBH	TOTAL CORE FLOW
27	CPDP	PSID	CORE PLATE DIFFERENTIAL PRESSURE
28	RSLFA	MLBH	'A' RECIRCULATION LOOP FLOW
29	RSLFB	MLBH	'B' RECIRCULATION LOOP FLOW
30	RSLTA	DEGF	'A' RECIRCULATION LOOP TEMPERATURE
31	RSLTB	DEGF	'B' RECIRCULATION LOOP TEMPERATURE
32	RSDFA	KGPM	'A' RECIRCULATION PUMP DRIVE FLOW
33	RSDFB	KGPM	'B' RECIRCULATION PUMP DRIVE FLOW
34	RSDPA	PSID	'A' RECIRCULATION PUMP DIFFERENTIAL PRESSURE
35	RSDPB	PSID	'B' RECIRCULATION PUMP DIFFERENTIAL PRESSURE
36	MGSDA	%SPD	'A' RECIRCULATION MOTOR GENERATOR SET SPEED
37	MGSDB	%SPD	'B' RECIRCULATION MOTOR GENERATOR SET SPEED
38	ALF	%	AUTOMATIC LOAD FOLLOWING
39	RSMCO	%	MASTER RECIRCULATION CONTROLLER OUTPUT
40	MGSCOA	%	'A' M/G SET CONTROLLER OUTPUT
41	MGSCOB	%	'B' M/G SET CONTROLLER OUTPUT
42	MGMAA	%	'A' M/G SET M/A STATION OUTPUT
43	MGMAB	%	'B' M/G SET M/A STATION OUTPUT
44	MGSTA	%	'A' M/G SET SCOOP TUBE POSITION
45	MGSTB	%	'B' M/G SET SCOOP TUBE POSITION
46	RPT-A	DIG	'A' RECIRCULATION (RPT) BREAKER

47	RPT-B	DIB	'B' RECIRCULATION (RPT) BREAKER
48	MGFBA	DIG	'A' RECIRC. MOTOR-GENERATOR SET DRIVE / FIELD BKR.
49	MGFBB	DIG	'B' RECIRC. MOTOR-GENERATOR SET DRIVE / FIELD BKR.
50	MGV-A	VOLT	'A' RECIRCULATION MOTOR-GENERATOR SET VOLTAGE
51	MGV-B	VOLT	'B' RECIRCULATION MOTOR-GENERATOR SET VOLTAGE
52	RCICI	DIG	REACTOR CORE ISOLATION COOLING INITIATION
53	RCFLO	GPM	REACTOR CORE ISOLATION COOLING PUMP FLOW
54	RCTSP	RPM	RCIC TURBINE SPEED
55	RCPSP	PSIG	REACTOR CORE ISOLATION COOLING SUCTION PRESSURE
56	RCPDP	PSIG	RCIC PUMP DISCHARGE PRESSURE
57	RCTIP	PSIG	RCIC STEAM SUPPLY PRESSURE
58	RCTEP	PSIG	RCIC TURBINE EXHAUST PRESSURE
59	RCETA	INCH	RCIC STEAM FLOW DELTA - P "A"
60	RCETB	INCH	RCIC STEAM FLOW DELTA - P "B"
61	RCVSA	%	RCIC STEAM ADMISSION VALVE POSITION
62	RCVTT	%	RCIC TRIP / THROTTLE VALVE POSITION
63	RCVCO	%	RCIC CONTROL VALVE POSITION
64	RCVVI	%	RCIC VESSEL INJECTION VALVE POSITION
65	RCFCO	%	REACTOR CORE ISOLATION COOLING CONTROLLER OUTPUT
66	RCPSC	%	RCIC SUCTION PRESSURE CONTROLLER OUTPUT
67	RCEGM	%	RCIC TURBINE EGM OUTPUT
68	RCRGS	%	RCIC RAMP GENERATOR SIGNAL / CONVERTER OUTPUT
69	HPCII	DIG	HIGH PRESSURE COOLANT INJECTION INITIATION
70	HPFLO	GPM	HIGH PRESSURE COOLANT INJECTION PUMP FLOW
71	HPSPD	RPM	HIGH PRESSURE COOLANT INJECTION TURBINE SPEED
72	HPPDP	PSIG	HPCI PUMP DISCHARGE PRESSURE
73	HPTIP	PSIG	HPCI STEAM SUPPLY PRESSURE
74	HPTEP	PSIG	HPCI TURBINE EXHAUST PRESSURE
75	HPETA	INCH	HPCI STEAM FLOW DELTA - P "A"
76	HPETB	INCH	HPCI STEAM FLOW DELTA - P "B"
77	HPVSA	%	HPCI STEAM ADMISSION VALVE POSITION
78	HPVST	%	HPCI STOP VALVE POSITION
79	HPVCO	%	HPCI CONTROL VALVE POSITION
80	HPVVI	%	HPCI VESSEL INJECTION VALVE POSITION
81	HPCO	%	HIGH PRESSURE COOLANT INJECTION CONTROLLER OUTPUT
82	HPEGM	%	HPCI TURBINE EGM OUTPUT
83	HPRGS	%	HPCI RAMP GENERATOR / SIGNAL CONVERTOR OUTPUT
84	RHF-A	KGPM	'A' RESIDUAL HEAT REMOVAL SYSTEM FLOW
85	RHF-B	KGPM	'B' RESIDUAL HEAT REMOVAL SYSTEM FLOW
86	SWLFA	KGPM	'A' RHR SYSTEM SERVICE WATER FLOW
87	SWLFB	KGPM	'B' RHR SYSTEM SERVICE WATER FLOW
88	RHP-A	PSIG	'A' RESIDUAL HEAT REMOVAL Hx PRESSURE
89	RHP-B	PSIG	'B' RESIDUAL HEAT REMOVAL Hx PRESSURE
90	RHPCA	%	'A' RHR Hx PRESSURE CONTROLLER OUTPUT
91	RHPCB	%	'B' RHR Hx PRESSURE CONTROLLER OUTPUT
92	RHL-A	%	'A' RESIDUAL HEAT REMOVAL Hx LEVEL
93	RHL-B	%	'B' RESIDUAL HEAT REMOVAL Hx LEVEL
94	RHLCA	%	'A' RHR Hx LEVEL CONTROLLER OUTPUT
95	RHLCB	%	'B' RHR Hx LEVEL CONTROLLER OUTPUT
96	MISMA	%	STEAM FLOW / FEED FLOW MISMATCH
97	MSFLO	MLBH	TOTAL STEAM FLOW
98	FWFLO	MLBH	TOTAL FEEDWATER FLOW
99	FWLFA	MLBH	'A' LOOP FEEDWATER FLOW
100	FWLFB	MLBH	'B' LOOP FEEDWATER FLOW

101	FWLFC	MLBH	'C' LOOP FEEDWATER FLOW
102	FWLTA	DEGF	'A' FEEDWATER LINE TEMPERATURE
103	FWLTB	DEGF	'B' FEEDWATER LINE TEMPERATURE
104	FWLTC	DEGF	'C' FEEDWATER LINE TEMPERATURE
105	FWPFA	GPM	'A' FEEDWATER PUMP FLOW
106	FWPFB	GPM	'B' FEEDWATER PUMP FLOW
107	FWPFC	GPM	'C' FEEDWATER PUMP FLOW
108	FWPDA	PSIG	'A' FEEDWATER PUMP DISCHARGE PRESSURE
109	FWPDB	PSIG	'B' FEEDWATER PUMP DISCHARGE PRESSURE
110	FWPDC	PSIG	'C' FEEDWATER PUMP DISCHARGE PRESSURE
111	VAC-A	INCH	'A' CONDENSER VACUUM
112	VAC-B	INCH	'B' CONDENSER VACUUM
113	VAC-C	INCH	'C' CONDENSER VACUUM
114	CPDHP	PSIG	CONDENSATE PUMP DISCHARGE HEADER PRESSURE
115	FWPSA	PSIG	FEEDWATER PUMP SUCTION HEADER PRESSURE
116	FWVSU	%	FEEDWATER STARTUP VALVE POSITION
117	FWVCA	%	FEEDWATER FLOW CONTROL VALVE 'A' POSITION
118	FWVCB	%	FEEDWATER FLOW CONTROL VALVE 'B' POSITION
119	FWVCC	%	FEEDWATER FLOW CONTROL VALVE 'C' POSITION
120	FWTSA	RPM	'A' FEEDWATER PUMP TURBINE SPEED
121	FWTSB	RPM	'B' FEEDWATER PUMP TURBINE SPEED
122	FWTSC	RPM	'C' FEEDWATER PUMP TURBINE SPEED
123	FWTRP	DIG	ALL FEEDWATER PUMP TURBINE TRIP
124	FWTRA	DIG	'A' FEEDWATER PUMP TURBINE TRIP
125	FWTRB	DIG	'B' FEEDWATER PUMP TURBINE TRIP
126	FWTRC	DIG	'C' FEEDWATER PUMP TURBINE TRIP
127	FWCVA	%	'A' FEEDWATER PUMP TURBINE CONTROL VALVE POSITION
128	FWCVB	%	'B' FEEDWATER PUMP TURBINE CONTROL VALVE POSITION
129	FWCVC	%	'C' FEEDWATER PUMP TURBINE CONTROL VALVE POSITION
130	FWSCO	%	STARTUP LEVEL CONTROLLER OUTPUT
131	FWMCO	%	FEEDWATER MASTER CONTROLLER OUTPUT
132	FWMAA	%	'A' FEEDWATER TURBINE BIAS M / A STATION OUTPUT
134	FWMAB	%	'B' FEEDWATER TURBINE BIAS M / A STATION OUTPUT
134	FWMAC	%	'C' FEEDWATER TURBINE BIAS M / A STATION OUTPUT
135	FWFGA	%	'A' FEEDWATER TURBINE FUNCTION GENERATOR OUTPUT
136	FWFGB	%	'B' FEEDWATER TURBINE FUNCTION GENERATOR OUTPUT
137	FWFGC	%	'C' FEEDWATER TURBINE FUNCTION GENERATOR OUTPUT
138	FWMFA	%	'A' FEEDWATER PUMP MINIMUM FLOW CONTROLLER OUTPUT
139	FWMFB	%	'B' FEEDWATER PUMP MINIMUM FLOW CONTROLLER OUTPUT
140	FWMFC	%	'C' FEEDWATER PUMP MINIMUM FLOW CONTROLLER OUTPUT
141	WLSET	INCH	REACTOR WATER LEVEL SETPOINT
142	NRWL	INCH	NARROW RANGE LEVEL (0 - 60)
143	WRWL	INCH	WIDE LEVEL (0 - 180)
144	LRWL	INCH	LOW RANGE LEVEL (-150 TO 60)
145	MSIA	DIG	'A' INBOARD MAIN STEAM ISOLATION VALVE POSITION
146	MSIB	DIG	'B' INBOARD MAIN STEAM ISOLATION VALVE POSITION
147	MSIC	DIG	'C' INBOARD MAIN STEAM ISOLATION VALVE POSITION
148	MSID	DIG	'D' INBOARD MAIN STEAM ISOLATION VALVE POSITION
149	MSOA	DIG	'A' OUTBOARD MAIN STEAM ISOLATION VALVE POSITION
150	MSOB	DIG	'B' OUTBOARD MAIN STEAM ISOLATION VALVE POSITION
151	MSOC	DIG	'C' OUTBOARD MAIN STEAM ISOLATION VALVE POSITION
152	MSOD	DIG	'D' OUTBOARD MAIN STEAM ISOLATION VALVE POSITION
153	MSLFA	MLBH	'A' STEAM LINE FLOW
154	MSLFB	MLBH	'B' STEAM LINE FLOW

155	MSLFC	MLBH	'C' STEAM LINE FLOW
156	MSLFD	MLBH	'D' STEAM LINE FLOW
157	ADSI	DIG	ADS INITIATION
158	SRVI1	DIG	RELIEF VALVE ABC INITIATE
159	SRVI2	DIG	RELIEF VALVE GJKL INITIATE
160	SRVI3	DIG	RELIEF VALVE DEHFP INITIATE
161	SRVI4	DIG	RELIEF VALVE RSMN INITIATE
162	SRVPB	5VDC	RELIEF VALVE 'B' POSITION
163	SRVPD	5VDC	RELIEF VALVE 'D' POSITION
164	SRVPF	5VDC	RELIEF VALVE 'F' POSITION
165	SRVPH	5VDC	RELIEF VALVE 'H' POSITION
166	SRVPK	5VDC	RELIEF VALVE 'K' POSITION
167	SRVPL	5VDC	RELIEF VALVE 'L' POSITION
168	SRVPN	5VDC	RELIEF VALVE 'N' POSITION
169	SRVPR	5VDC	RELIEF VALVE 'R' POSITION
170	SRVPA	5VDC	RELIEF VALVE 'A' POSITION
171	SRVPC	5VDC	RELIEF VALVE 'C' POSITION
172	SRVPE	5VDC	RELIEF VALVE 'E' POSITION
173	SRVPG	5VDC	RELIEF VALVE 'G' POSITION
174	SRVPJ	5VDC	RELIEF VALVE 'J' POSITION
175	SRVPM	5VDC	RELIEF VALVE 'M' POSITION
176	SRVPP	5VDC	RELIEF VALVE 'P' POSITION
177	SRVPS	5VDC	RELIEF VALVE 'S' POSITION
178	MWE	MWE	GENERATOR GROSS MEGAWATTS OUTPUT
179	GFREQ	HZ	GRID FREQUENCY
180	GVOLT	KV	GRID VOLTAGE
181	MTSPD	RPM	MAIN TURBINE SPEED
182	TBSF	MLBH	TURBINE STEAM FLOW
183	MTT	DIG	MAIN TURBINE TRIP
184	MGB	DIG	GENERATOR BREAKER OPEN
185	PLUNB	DIG	POWER / LOAD UNBALANCE
186	LDSET	%	LOAD SET
187	TPSET	%	TRANSIENT PRESSURE SETPOINT
188	PRO	%	PRESSURE REGULATOR OUTPUT
189	CVAO	%	CONTROL VALVE AMPLIFIER OUTPUT
190	PRSPA	PSIG	'A' PRESSURE REGULATOR SETPOINT
191	PRSPB	PSIG	'B' PRESSURE REGULATOR SETPOINT
192	TIPA	PSIG	'A' PRESSURE REGULATOR SENSED PRESSURE
193	TIPB	PSIG	'B' PRESSURE REGULATOR SENSED PRESSURE
194	CVPT	%	TOTAL CONTROL VALVE POSITION
195	CVP1	%	CONTROL VALVE #1 POSITION
196	CVP2	%	CONTROL VALVE #2 POSITION
197	CVP3	%	CONTROL VALVE #3 POSITION
198	CVP4	%	CONTROL VALVE #4 POSITION
199	SVP1	%	STOP VALVE #1 POSITION
200	SVP2	%	STOP VALVE #2 POSITION
201	SVP3	%	STOP VALVE #3 POSITION
202	SVP4	%	STOP VALVE #4 POSITION
203	BPVT	%	TOTAL BYPASS VALVE POSITION
204	BPVP1	%	#1 BYPASS VALVE POSITION
205	BPVP2	%	#2 BYPASS VALVE POSITION
206	BHDT	DEGF	BOTTOM DRAIN TEMPERATURE
207	DWP	PSIA	DRYWELL PRESSURE
208	NRDP	PSIG	NARROW RANGE PRESSURE (850 - 1050)

209	[illegible]DP	PSIG	WIDE RANGE PRESSURE (0 - 1200)	21	1	4	30
210	1CBUS	DIG	*1C BUS POWER	12	1	6	20
211	1DBUS	DIG	*1D BUS POWER	13	1	6	31
212	1EBUS	DIG	*1E BUS POWER	14	1	6	32
213	1FBUS	DIG	*1F BUS POWER	15	1	7	1
214	EVENT	DIG	EVENT MARKER	1	1	7	18
215	MSLPA	PSIG	'A' MAIN STEAM LINE PRESSURE NEAR RELIEF VALVES	193	1	4	5
216	MSLPB	PSIG	'B' MAIN STEAM LINE PRESSURE NEAR RELIEF VALVES	194	1	4	6
217	MSLPC	PSIG	'C' MAIN STEAM LINE PRESSURE NEAR RELIEF VALVES	195	1	4	7
218	MSLPD	PSIG	'D' MAIN STEAM LINE PRESSURE NEAR RELIEF VALVES	196	1	4	8

SCRAM TIMING SIGNALS.

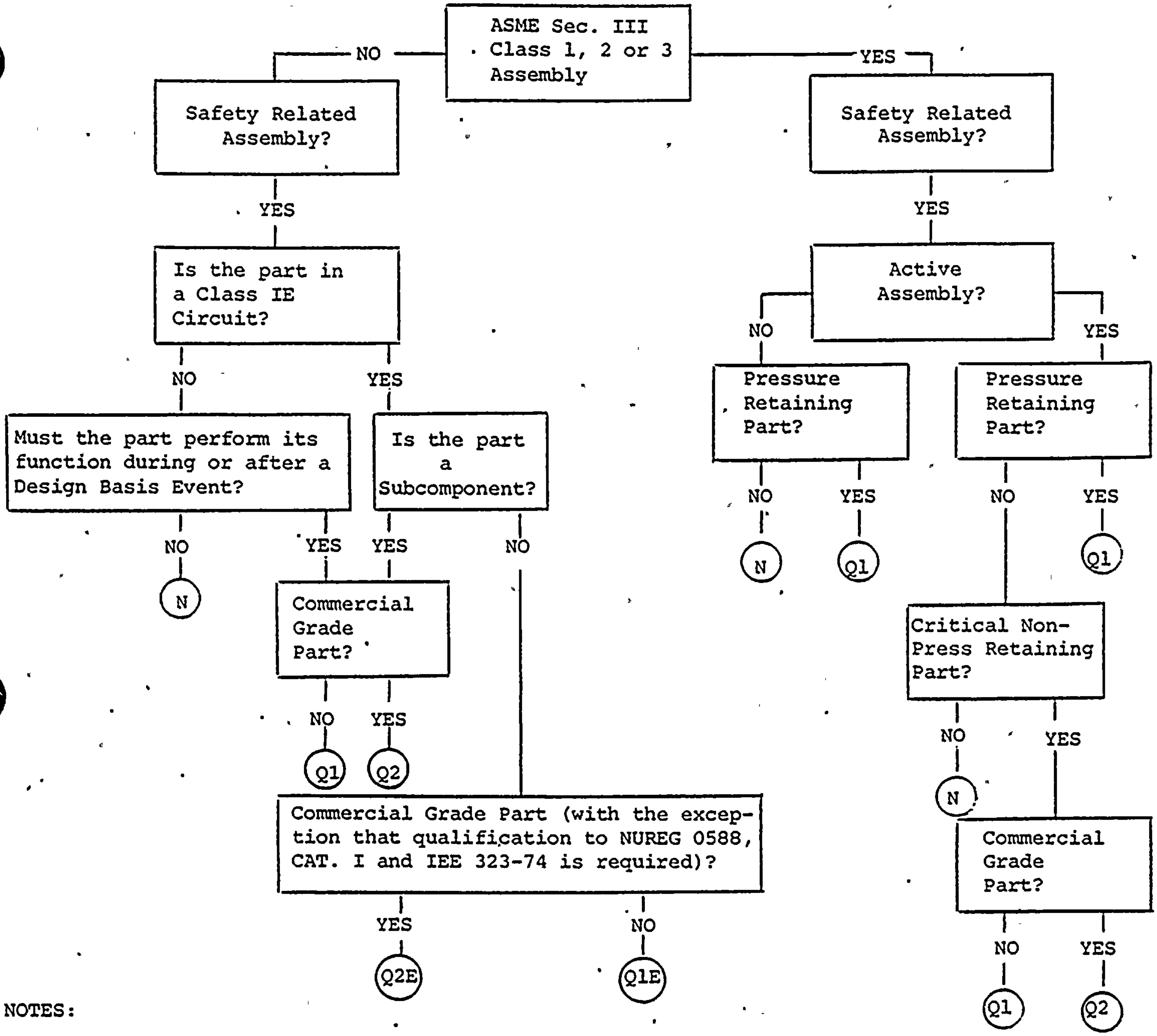
CH.#	NAME	UNITS		SUB-CHANNELS #1	#2	#3	#4	LINK	SB LNK	PORT
483	Spare CRD Channel									
484	CRD02	DIG4	CRD'S	34-03,	38-03,	42-03,	14-07.	13	1	2
485										
486	Spare CRD Channels									
487										
488	CRD06	DIG4	CRD'S	26-11,	30-11,	34-11,	38-11.	13	1	6
489	CRD07	DIG4	CRD'S	42-11,	46-11,	50-11,	6-15.	13	1	7
490										
491	Spare CRD Channels									
492										
493	CRD11	DIG4	CRD'S	2-19,	6-19,	10-19,	14-19.	13	1	11
494	CRD12	DIG4	CRD'S	18-19,	22-19,	26-19,	30-19.	13	1	12
495	CRD13	DIG4	CRD'S	34-19,	38-19,	42-19,	46-19.	13	1	13
496										
497										
498										
499	Spare CRD Channels									
500										
501										
502	CRD20	DIG4	CRD'S	26-27,	30-27,	34-27,	38-27.	13	1	20
503	CRD21	DIG4	CRD'S	42-27,	46-27,	50-27,	54-27.	13	1	21
504										
505	Spare CRD Channels									
506										
507										
508	CRD26	DIG4	CRD'S	2-35,	6-35,	10-35,	14-35.	14	1	1
509	CRD27	DIG4	CRD'S	18-35,	22-35,	26-35,	30-35.	14	1	2
510	CRD28	DIG4	CRD'S	34-35,	38-35,	42-35,	46-35.	14	1	3

0539	51[illegible]	[illegible]RD29	DIG4	CRD'S 50-35, 54-35, 58-35, 2-39.	14	1	4
0540	512						
0541	513						
0542	514		Spare CRD Channels				
0543	515						
0544	516	CRD34	DIG4	CRD'S 10-43, 14-43, 18-43, 22-43.	14	1	9
0545	517	CRD35	DIG4	CRD'S 26-43, 30-43, 34-43, 38-43.	14	1	10
0546	518	CRD36	DIG4	CRD'S 42-43, 46-43, 50-43, 54-43.	14	1	11
0547	519						
0548	520						
0549	521		Spare CRD Channels				
0550	522						
0551	523	CRD41	DIG4	CRD'S 18-51, 22-51, 26-51, 30-51.	14	1	16
0552	524						
0553	525						
0554	526		Spare CRD Channels				
0555	527						
0556	528						
0557	529						
0558	530						
0559	531						

Attachment 9 - SSES Unit 1 GETARS Points Presently Used to Trigger Recording

Channel	Description
1	Total Scram
4	Manual Scram
46	A Recirc RPT Breaker
47	B Recirc RPT Breaker
183	Safety Relief Valve "S" Position
214	Narrow Range Reactor Vessel Pressure

ITEM CLASSIFICATION LOGIC FOR PARTS OF ASSEMBLIES



NOTES:

1. Commercial Grade Item - An item that:
   a. is not subject to design or specification requirements unique to facilities/activities licensed by the NRC.
   b. is used in applications in addition to facilities licensed by the NRC.
   c. is listed in a manufacturer's or distributor's published product description, e.g.: catalog.
2. Critical Non-pressure Boundary parts are non-pressure retaining parts or appurtenances whose failure would compromise the proper operation of an active engineered safety component.

6.0 CONDITIONS OF SERVICE

6.1 Elevation

The Cable Penetrations will be installed indoors between elevations of approximately 707 to 712 feet above sea level.

6.2 Service

Each electrical penetration assembly shall be suitable for installation into a containment structure penetration nozzle which is part of the containment structure.

6.3 Design Life

The assemblies shall have a minimum design life of 40 years. At the end of 40 year life, they shall be capable of meeting the requirements for normal and emergency operation in accordance with the following sections.

6.4 Normal Environment - (For Penetrations No. 2W104E and 2W100H)

The outboard end of Cable Penetrations will be subject to the following conditions during normal, continuous operation.

a.	Pressure	-0.375" wg
b.	Ambient temperature	115°F (maximum continuous) 60°F (minimum)
c.	Relative Humidity	10% to 90%
d.	Radiation dose (Gamma)	0.100 Rads per hour (average) 3.5×10^4 rads over 40 years

The inboard end of the Cable Penetrations will be subject to the following conditions during normal, continous operation. (The values inside bracket are for wetwell penetrations. The other values are for drywell penetrations.)

a.	Pressure	0.1 to + 1.5 psig (0.1 to + 1.5 psig)
b.	Ambient Temperature	150°F Max. Cont.
c.	Relative Humidity	20% to 90% (100% continuous)
d.	Radiation dose (Gamma)	4 rads per hour (0.1 rads per hour) Average 1.1×10^6 rads (3.5×10^4 rads) over 40 years

The above data is without margins. Margins shall be added for qualifications as required by IEEE Std. 323-1974 per NUREG-0588 Category I.

6.5 Emergency Environment - (For Penetrations No. 2W104E and 2W100H)

The outboard end of the cable penetrations will be subject to the following conditions during abnormal condition:

a.	Pressure	2.2 psig	
b.	Temperature	300°F	0-60 seconds
		130°F	60-seconds-100 days
c.	Relative Humidity	100%	1-12 Hrs.
		90%	12 Hrs - 100 days
d.	Radiation dose (Gamma)	8.3×10^5 rads per hour (average)	
		1.7×10^7 rads (TID)	
e.	Radiation dose (Beta)	5.0×10^3 rads per hour (average)	
		1.1×10^6 rads (TID).	

TID shown in (d) and (e) includes normal 40 years dose plus accident dose.

All Cable Penetrations shall maintain their pressure seal and all electrical circuits shall remain operable when the interior ends of the penetrations are subject to the following emergency conditions at 100% relative humidity: (The values inside brackets are for wetwell penetrations. The other values are for drywell penetrations.)

a. First 45 seconds

Pressure	44 psig (29 psig)
Temperature	340°F (130°F)

The time of rise from normal to the above temperature and pressure will be within 10 seconds.

b. Next 3 hours

Pressure	35 psig (30 psig)
Temperature	340°F (200°F)

c. Next 3 hours

Pressure	35 psig (30 psig)
Temperature	320°F (210°F)

d. Next 18 hours

Pressure — 20 psig (15 psig)

Temperature — 250°F (210°F)

e. Next 99 days

Pressure — 10 psig (10 psig)

Temperature — 200°F (200°F–next 120 hrs., 140°F–120 hrs.–99 days)

f. Radiation Dose (Gamma) 1.6×10^7 Rads/hr. (1.9×10^6 Rads/hr)(average)
7.1×10^7 Rads(2.7×10^7 Rads)(TID)

g. Radiation Dose (Beta) 2.3×10^8 Rads/hr. (2.3×10^8 Rads/hr)(average)
1.9×10^9 Rads (1.9×10^9 Rads) (TID)

All above data is without margins. The margins shall be added for qualification as required by IEEE Std. 323-1974 per [NUREG-0588, Category I. TID shown in (f) and (g) includes 40 year dose plus accident dose.

6.6 Design Data

The cable penetrations shall be suited for the following conditions:

a) Radiation Streaming along axis of cable penetration ... 15–100 mr/hr.

b) External differential pressure (drywell to interior of the terminal boxes) 5 psig

c) Containment internal design pressure of 53 psig
Seller shall qualify cable penetrations to 62 psig in order to match rating of existing cable penetrations. 4

7.0 SEISMIC AND HYDRODYNAMIC REQUIREMENTS

7.1 The Cable Penetrations shall meet the requirements of Specification 8856-G-22, "General Project Requirements for Design Assessment and Qualification of Seismic Category I Equipment and Equipment Supports for Seismic and Hydrodynamic Loads". In addition, the cable penetrations shall be in compliance with IEEE Standard No. 344, "Recommended Practice for Seismic Qualification of Class I Electric Equipment for Nuclear Power Generating Stations."

7.2 During and after all DBE's penetrations must maintain pressure seal integrity. In addition, the penetrations indicated in Table I, Note 1.2 (see Material Requisition) must remain functional.

10.0. FINISH

10.1 Surface preparation, prime coat application, shop priming system, material, dry film thickness, inspection and tests shall meet requirements of Specification 8856-G-4 "General Project Requirements for Shop Priming of Mechanical and Electrical Equipment".

10.2 All machined surfaces of carbon steel which will not be welded upon shall be temporarily protected against oxidation during shipment and storage by liberal application of a grease coating conforming to Military Specification MIL-C-16173, Grade I: "Corrosion Preventive Compound, Solvent Cutback, Cold-Application".

10.3 All machined surfaces of carbon steel to be welded, and all carbon steel surfaces within 2 inches of field welds shall be protected against corrosion during shipment and storage by a coat of DE-OX Aluminate or Buyer approved equal, which can readily be removed prior to field welding.

10.4 Seller shall not finish paint the equipment.

10.5 Deleted.

10.6 The cable penetration design shall permit field cleaning, priming and painting in those areas that will require it after field welding.

11.0 DEFECTIVE MATERIAL

Warranty and limitation of liability shall be as stated in the Purchase Order.

12.0 QUALITY ASSURANCE

12.1 The Cable Penetrations listed in the Material Requisition shall perform critical safety related functions in a nuclear power plant and will require a high level of quality control and documentation of design and manufacture. For this equipment the Seller shall maintain a Quality Assurance Program which complies with Specification 8856-G-9, "General Project Requirements for Quality Assurance on Purchase Orders for 'Q' Designated items, and the requirements of the applicable specifications.

The Seller shall furnish the documentation as specified below and on forms G-321-C and 8856-QA1, in accordance with the applicable requirements of Specification 8856-G-9.

12.2 Each Cable Penetration shall be qualified by analysis and successful use under similar conditions or by actual test to demonstrate its ability to perform its function under normal and emergency conditions of service as defined in Section 6.0 of this Specification. Compliance with the following requirements shall be documented section by section. A general statement of conformance will not be sufficient. The analysis and testing program shall meet or exceed the requirements contained in:

12.2.1 IEEE No. 323-1974, "Qualifying Class I Electric Equipment for Nuclear Power Generating Stations," per NUREG-0588 Category I.

12.2.2 IEEE Standard 317, "Electrical Penetration Assemblies in Containment Structures for Nuclear Power Generating Stations," and as required in specification paragraph 13.1.

12.2.3 (DELETED)

12.3 Intentionally left blank.

12.4 Intentionally left blank.

12.5 Intentionally left blank.

12.6 Written documentation and/or certified test results shall be submitted verifying that the Cable Penetrations meet the following:

12.6.1 Factory Production Tests as required in specification paragraph 13.2.

12.6.2 Materials Manufacturer's or Materials Supplier's Certified Material Test Reports and/or Certificates of Compliance and ASME Design Reports as required in specification paragraph 13.4.

12.6.3 Non-destructive Test Results as required in Specification paragraph 13.5.2

12.7 Evidence of compliance to seismic requirements test procedures and test reports shall be furnished in accordance with specification paragraph 7.0 Seismic Requirements.

12.8 Certificate of compliance and test reports, indicating that the cable penetration has been qualified to the Section 12.2 requirements, shall be submitted.

12.9 Before the penetration is placed in service, copies of the appropriate Certificate Holder's Data Report shall be filled in by the owner with the Commonwealth of Pennsylvania which is the enforcement authority.

13.0 TESTS AND TEST REPORTS

13.1 Design Qualification

The prototype testing shall be met by one of the two following methods or a combination thereof:

Method I: Prior testing of prototypes of like design which in the judgement of the Purchaser are adequate to meet the requirements state herein. The buyer will use, as one basis for judgement, the IEEE Std No. 323, the guidance in NUREG-0588, Cat. I and IEEE Std No. 317, IEEE Std. No. 344 and Specification 8856-G-22.

Method II: Prototype testing performed after award of a purchase order on individual cable penetrations of the types, sizes and conductor groupings as specified herein.

If the Bidder proposes to qualify the cable penetration by Method I, he shall submit tests he has performed and adequately documented and which specific types of assemblies he believes are qualified under Method I above.

Documentation shall include complete description of the prototype cable penetrations, date and place of tests, test procedures, results and copies of certified test reports. Method I will be acceptable only if the prototype assemblies are clearly and unquestionably representative of the actual product supplied to the Buyer.

In the event the Buyer judges that Method I above is inadequate to properly qualify individual assemblies or all assemblies, it will be necessary that prototype testing under Method II be performed.

13.1.1 Prototype Tests

Prototype tests, if required, shall be performed on a prototype assembly of each type as defined in section 8.1 and each rating as defined in section 8.2. These tests shall be performed in accordance with section 6.0, "Conditions of Service", and section 8.0 "Type and Rating" of this Specification.

Existing qualification test data may be used to broaden the coverage of a particular test when it can be shown that the existing test data are valid for the penetration assembly being supplied.

13.1.2 The Seller shall submit documentation showing that each of the materials used in the fabricated Cable Penetration, including bushings, pigtails and/or connectors, is capable of withstanding the radiation environment described in Section 6.0, "Conditions of Service" without compromising its ability to remain operational.

13.1.3 It shall be demonstrated by test (using an equivalent sand, gravel or concrete mix) that the nozzle-concrete interface temperature will not exceed 150°F at any point under the most adverse conditions of normal operation stated in section 9.1.5 while all conductors carry rated current. This demonstration need only be applied to the low voltage or medium voltage assemblies with the largest calculated

13.1.4 After the tests it shall also be demonstrated that these circuits identified in Table I as required to operate in an emergency will remain functional and the remainder maintain leaktight integrity.

13.1.5 No Cable Penetration that has been subjected to the above prototype tests will be accepted for installation at the jobsite. However, the header plate may be reused if it meets the quality assurance requirements.

13.1.6 At the Buyer's option all of the prototype tests may be witnessed. The Buyer shall be notified 30 days in advance of the testing. Certified test reports are required for each cable penetration tested.

13.2 Factory Production Tests (Non-Destructive)

Production tests shall be performed on each penetration assembly prior to shipment as per Section 7 of IEEE 317. ~~At least two months~~ Before tests are to be performed, the Seller shall submit a written description of the tests and of the test procedures.

In addition to meeting the requirements of IEEE 317, the following supplementary requirements shall also apply:

13.2.1 Each assembly shall be proof tested across pressure barrier seals with ~~50 psig~~ 62 psig (plus the margins per IEEE-317) Helium to verify structural integrity, for field test. Pneumatic testing per Para 7.2 of IEEE-317 shall meet the requirements of ASME Sect. NE 6320

13.2.2 The leakage rate of each penetration assembly shall be determined and shall not exceed 1×10^{-6} std cc of Helium per second.

13.2.3 After the routine leak rate tests each cable or conductor and associated connectors, except coaxial and triaxial conductors and connectors, in all cable penetrations shall be tested per and shall meet the requirements of NEMA/ICEA Standards of Voltage Tests after Installation except that the cables need not be immersed in water if the type of construction shall be tested against every other conductor and also tested to the metallic end plates. ~~Coaxial and triaxial cable conductors and connectors shall be given a test at manufacturer's maximum rated voltage between center wire and inner, or first, shield for a period of one minute.~~

13.2.4 Coaxial and triaxial cables and connectors shall receive a dieletric withstand voltage test. The dielectric withstand test shall be at the manufacturer's maximum rated voltage between conductors and between conductors and inner, or first, shield for a period of one minute.

Largest motor HP to be started	2000 HP (1640 amps at 0.18 pf during starting)
Elevation at site	676 ft

5.2.2 Design Environmental Conditions

a) Temperature - Engine off (Min/Max) 72°/104°F

b) Temperature - Engine Running:

Max. At Generator	120°F
At Engine	160°F

c) Outside Air Temperature

Dry-bulb air temperature, minimum/maximum	-19°F/105°F
Wet-bulb air temperature, minimum/maximum	-13°F/82°F

d) Pressure Atmosphere

e) Radiation - Integrated Dose 1.8×10^2 Rads

f) Relative Humidity, Max/Min 90%/5%

g) Vendor shall supply a certificate of compliance for the listed environment conditions. 1

5.2.3 Cooling Water Available

a) Cooling water for the heat exchangers will be raw water from the spray pond receiving approximately 100 gpm of warm circulating water for deicing purposes.

Supply pressure	130 psig Max.
Max. Temperature	95°F
Min. Temperature	33°F

The cooling water will have the following analysis:

	Average	Max.
Calcium as Ca	121 ppm	240 ppm
Magnesium as Mg	35.5 ppm	81 ppm
Sodium as Na	9.8 ppm	34 ppm

5.2.4 Diesel Generator Building

The diesel generator unit will be housed in a reinforced concrete, Class I structure. The unit is completely enclosed in its own concrete and missile protected cell. The unit is also protected against other natural hazards such as tornadoes, floods, lightning, rain, ice or snow.

5.2.5 Seismic Requirements:

a) The diesel generator system is a Seismic Category I and safety related system. The diesel generator with auxiliary systems and supports shall be designed to function through and after five (5) Operational Basis Earthquake followed by a Safe Shutdown Earthquake (SSE) and shall meet the requirements of the General Specification for Design Assessment and Qualification of Seismic Category I Equipment and Equipment Supports for Seismic & Hydrodynamic Loads, G-1024. Use of the Seller's standard more stringent seismic qualification procedure for control devices may be substituted if the procedure is reviewed and approved by Buyer prior to testing. 1

b) The Seller shall design and submit seismic calculations for each diesel generator, control panels, control devices and all other accessories as required either in accordance with the attached General Specification G-1024, or to the pertinent manufacturer's procedures providing such procedures have prior approval by the Buyer.

c) The Seller shall also state the maximum allowable loads for the Seller's equipment nozzle interfacing with Buyer's supplied components or piping.

5.3 PERFORMANCE REQUIREMENTS:

5.3.1

The diesel generator will be installed as a backup to the four existing diesel generators, which are used in the event of loss of all normal sources of power. Upon loss of power the diesel generators shall start automatically and attain full operating voltage and rated frequency within ten seconds. Each diesel generator feeds an independent 4.16 kV bus arrangement for each reactor unit (2 units). Diesel-generator voltage setpoint control will be set by the Buyer to produce a full operating voltage within the range of 4160 volts plus or minus 5% as permitted by NEMA MG1-22.47 to suit plant operating conditions.

5.3.2

The diesel engine shall be capable of starting from cold. Reliability of automatic start and operation is of utmost importance and shall be of prime consideration in the design of the equipment.